Consumers Power
POWERING MICHIGAN'S PROGRESS
Big Rock Point Nuclear Plant, 10269 US-31 North, Charlevoix, MI 49720

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May 27, 1994

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES - REISSUE OF SECTION 3.0 - RESULTS AND SCREENING PROCESS (Generic Letter 88-20)

On May 5, 1994, a letter forwarding the Individual Plant Examination for Severe Accident Vulnerabilities (IPE) for the Big Rock Point Plant was submitted to the Nuclear Regulatory Commission. The purpose of this letter is to reissue Section 3.0, Results and Screening Process and Section 13.0, Uncertainties.

The IPE is an intricate, living document, that requires continuing maintenance. As issued, Section 3.0 (Attachment 1) was adequate for the required submittal. However, additional information and intrusive technical reviews have recently been performed with regard to Section 3.0. The result has enhanced the quality of the original submittal, therefore requiring the reissue of the IPE. Attachment 2 is a corrected copy of Section 13.0. The original version of Section 13.0 was erroneously printed such that the numeral 1 was omitted from the values reported in Tables 13.1-1, 13.1-2, 13.1-3, 13.1-4, 13.3-1, 13.3-2, 13.3-3 and 13.9-1. No textual or content changes have been made to the section.

Attached are the enhanced Section 3.0 and corrected Section 13.0. They should be placed immediately in your copy of the IPE; and the original Section 3.0 and 13.0 discarded. Thank you in advance for your cooperation in this matter.

Patrick M Donnelly

Patrick M Donnelly
Plant Manager

CC: Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point

ATTACHMENTS

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CONSUMERS POWER COMPANY

Big Rock Point Plant
Docket 50-155 License DPR-06

REVISED RESPONSE TO THE MAY 5, 1994
RESPONSE TO GENERIC LETTER No 88-20

At the request of the Commission and pursuant to the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, as amended, and the Commission's Rules and Regulations thereunder, Consumers Power Company submits our response to NRC letter dated November 23, 1988, entitled, "Individual Plant Examination for Severe Accident Vulnerabilities". Consumers Power Company's response is dated May 27, 1994.

CONSUMERS POWER COMPANY

To the best of my knowledge, information and belief, the contents of this submittal are truthful and complete.

By Robert A. Fenech

Robert A Fenech, Vice President
Nuclear Operations

Sworn and subscribed to before me this 27th day of May, 1994.

LeAnn Morse

LeAnn Morse, Notary Public
Berrien County, Michigan
(Acting in Van Buren County)

My commission expires February 4, 1997.

(SEAL)

ATTACHMENT 1

CONSUMERS POWER COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

RESPONSE TO GENERIC LETTER 88-20
REISSUE OF SECTION 3.0 FROM MAY 5, 1994 RESPONSE

May 27, 1994

51 Pages

3.0 RESULTS AND SCREENING PROCESS

3.1 Introduction

This section summarizes the overall results from the quantification of the Big Rock Point Level 1 PRA. Internal events are discussed, including internal flooding. The PRA analysis focused on plant design features and operating characteristics that are critical for prevention of core damage. Detailed descriptions of all dominant accident sequences are provided and are grouped by accident class. The specific items discussed for each sequence are:

1. Description of accident progression, event timing and containment failure mode, if applicable.
2. Specific assumptions to which the results are sensitive.
3. Significant initiating events, human actions and sensitive parameters.

The calculated CDF for the Big Rock Point internal events PRA is 5.4E-5/year. Unless noted otherwise, core damage is defined as those combinations of systems and equipment failures which would lead to core temperatures exceeding 2500°F.

3.1.1 Application of Generic Letter Screening Criteria

Appendix 2 from Generic Letter 88-20 identifies the screening criteria for reporting potentially important sequences which might lead to core damage or unusually poor containment performance. The criteria applicable to Big Rock Point are listed below:

1. Any functional sequence that contributes 1E-6 or more (per reactor year) to core damage.
2. Any functional sequence that contributes 5% or more to the total core damage frequency.
3. Any functional sequence that has a core damage frequency greater than, or equal to, 1E-6 per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than, or equal to, the BWR-3 release categories of WASH-1400.
4. Functional sequences that contribute to a containment bypass frequency in excess of 1E-7 per reactor year.

5. Any functional sequences that the plant determines (from previous applicable PRAs or by utility engineering judgment) are important contributors to core damage frequency or poor containment performance.

The reporting criteria used in this section exceeds the reporting criteria of Generic Letter 88-20. While the Big Rock Point event trees are systematic, the following discussion of accident sequences is categorized by the functional accident classes as described in Section 4.4.1. Each accident class consists of a set of front-line and support systems that are used to perform a specified safety function. Within each functional accident class, the individual core damage sequences that emerge have similar sets of components and initiating events that fail the safety functions. The accident classes discussed in this section are listed in Table 3.1-1 and include a definition of each class.

The results that follow are reported by accident class (from largest to lowest contributor) with characterizations for each accident class. Significant equipment failures, operator actions and a representative sequence for each accident class are also provided. The contributions to the specific class CDF from component failures, initiating events or operator actions are the percentage of the risk associated with the failure of the applicable component, initiator or operator action.

3.2 Accident Classes Initiated by Reactor Coolant System Pipe Rupture

The first three accident classes discussed are those initiated by a loss of coolant accident. While among the least frequent of initiating events, loss of coolant accidents constitute roughly 80% of the estimated internal events core damage frequency for Big Rock Point. This distribution in risk is largely due to the unique design characteristics of Big Rock Point. These characteristics provide multiple and diverse means of assuring adequate core cooling during initiators which do not involve a pipe rupture, thereby limiting the risk associated with non-LOCA events.

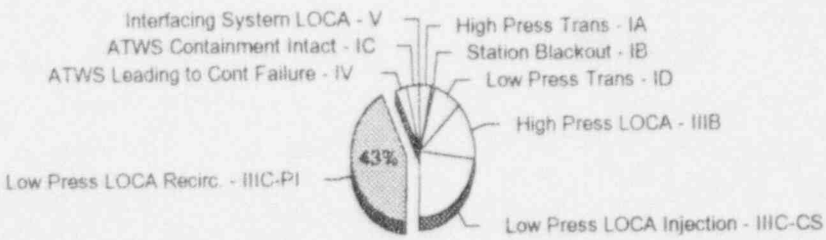
The total loss of coolant accident frequency for Big Rock Point is assumed to be 1E-2/year. This frequency is consistent with that used in other PRAs and is believed to very conservatively bound the overall potential for a loss of coolant accident (based on actual industry experience and the lack of such events in more than 1000 reactor years of operation).

The pipe rupture initiators are distributed in size among large, medium, small and very small pipe failures. Further, the location of the break in the primary system plays a role in plant system success criteria in responding to the LOCA initiator (i.e., whether the break is located below or above the core or in steam piping).

Key assumptions regarding system success criteria, as they relate to pipe rupture size and location, are discussed as a part of each accident class.

3.2.1 Accident Class IIIC-PI (LOCAs with Failure of Post Incident Recirculation) 2.3E-5/Year

This accident class is characterized by a pipe rupture in the primary coolant system with successful depressurization of the reactor as well as adequate core cooling either by core spray or the feedwater/condensate systems. Core damage is assumed to be a result of failures in post incident recirculation for this accident class and occurs very late in the accident progression (many hours to days). Accident Class IIIC-PI makes up 43% of the total internal events core damage frequency.



<u>Accident Class Characteristics (IIIC-PI)</u>

A rise in the water level in the containment occurs following the establishment of successful injection to the reactor with either core spray or feedwater/condensate. The rate at which the water level rises to the 587 foot elevation in containment depends on the size of the break and its location. Large breaks below the core result in reaching this elevation within about 4 hours, whereas small breaks, or breaks located well above the core in the steam drum on the main steam line can take many days.

When the containment level reaches 587 feet and PCS pressure is less than 65 psig, the operator is required to initiate the post incident system to recirculate water from the containment, pass it through the post incident heat exchangers, and return it to the core through one of two core spray lines. If recirculation to the reactor cannot be established by using the post incident system, emergency operating procedures require continued injection to the reactor by use of the fire system (through core spray lines) or the feedwater/condensate system. On reaching an elevation of 630 feet (corresponding to a level well above the top of the fuel in the reactor), injection is terminated in order to prevent stressing the containment sphere due to overfilling.

Core damage is postulated to occur, for this accident class, if recirculation is established (using the post incident system) but subsequently fails, or if recirculation cannot be established and injection is eventually terminated due to the level of water in containment.

Key Assumptions (Class IIIC-PI)

The following is a summary of assumptions made in the PRA which influence the quantification of sequences in the IIIC-PI Accident Class.

- All LOCAs (non-steam pipe breaks) eventually require initiation of post incident recirculation. This includes pipe break locations above the core as well as below. The EOPs instruct the operators to raise the water in the primary system to normal level in the steam drum with the available injection systems. It is therefore assumed that LOCAs above the core, but below the drum, lead to filling the containment (just as LOCAs below the core would).
- Steam line breaks are assumed not to require initiation of the post incident system. On returning the primary system level to normal, by use of core spray or feedwater/condensate, primary system level is maintained as opposed to overflowing the drum and circulating water through the break or RDS trains into containment. By maintaining drum level in this manner, steaming to containment occurs as opposed to water addition at fire pump flow rates. Makeup requirements and addition of water to containment are reduced to decay heat rates.
- For sequences in which recirculation is not established, heat removal through the post incident system heat exchanger is not available. However, long term overpressurization of containment (due to decay heat) is assumed not to occur. Passive heat removal through the large surface area of the containment steel shell and other passive heat sinks is sufficient to maintain containment pressure near design for days.
- In addition to recirculation, heat removal through the post incident heat exchanger is assumed to be required for LOCAs that are located below the elevation of the steam drum. While passive heat removal through the containment shell may be sufficient for these break locations, heat removal is assumed to be required in order to maintain adequate NPSH for the post incident system pumps. This assumption may be conservative for scenarios in which injection systems raise the water level in containment above the 595 foot elevation. The additional static head of water and compression of the atmosphere in containment, as the level is raised, could permit the

maintenance of adequate NPSH even without the post incident heat exchanger.

- LOCA size or the number of injection trains in service (including containment spray) have little influence on the operator's ability to initiate recirculation. This is due to the long time-frame available to initiate recirculation (approximately 4 hours, even for a large break) and the ability to continue in the injection mode almost indefinitely until recirculation can be initiated (the fire pumps take suction from Lake Michigan).
- Passive heat removal from the reactor to the water surrounding the vessel is conservatively ignored, even for sequences in which containment water level is raised to above the elevation of the core. Should injection to the reactor be terminated, once containment level is raised to the 630 foot elevation, boil-off of the water in the reactor is assumed to result in lowering the reactor level to below the top of the fuel.
- On establishing operation of post incident system recirculation, a return to injection systems (if post incident components subsequently fail) is conservatively ignored.

Contribution of Initiating Events (Class IIIC-PI)

A spectrum of pipe break initiators was developed for the Big Rock Point PRA with respect to both size and location of the postulated break. The following summarizes the contribution of these pipe break initiators to this accident class.

- Small and very small break sizes constitute the majority of the risk associated with failure of recirculation following a LOCA. Medium and large LOCAs contribute to only a fraction of the core damage frequency associated with this accident class. This distribution in risk is due principally to the potential for each of these break sizes (the frequency of the larger breaks being much less than small pipe break initiators).

LOCA Size	Contribution to Class IIIC-PI
Very Small	35%
Small	43%
Medium	18%
Large	4%

- Location of the pipe break also has an influence on the potential for this accident class. Only LOCAs below the core and above the core, but below the steam drum, contribute to this accident class. Steam line breaks do not require recirculation as primary system level can be returned to normal using injection systems alone. The EOPs do not require overfilling the steam drum in order to establish recirculation. As a result, steam line breaks do not contribute to this accident class.

Break Location	Contribution to Class IIIC-PI
Below Core	83%
Above Core	17%
Steam Line	-

Contribution of Hardware (Class IIIC-PI)

Random failures of components associated with the core spray and post incident systems constitute the entire risk associated with this accident class. There is little contribution from other systems (such as feedwater/condensate). The lack of importance of alternate injection systems is due to the assumption that, once containment level is raised to the 630 foot elevation, injection to the reactor must be terminated and only recirculation with the post incident system can maintain adequate core cooling.

Components	Contribution to Class IIIC-PI
Core Spray Valve (Common Cause Failures)	40%
Post Incident System Pumps (Common Cause Failures)	26%
Check Valve VPI 300 Fails to Open	7%
Fire Pumps	2%
Heat Exchanger MOVs	<1%

- The largest hardware-related contributor to failure to establish recirculation is the core spray valves. Their contribution suggests that between one-third and one-half of LOCAs leading to core damage during recirculation are due to the need to terminate feedwater/condensate in order to prevent overfilling the containment. These sequences take a significant amount of time to develop (approximately a day). There is a significant potential to mitigate these events by repair and recovery activities which have not been credited in the sequence quantification.

- Approximately one quarter of the sequences in this accident class are assumed to occur due to the inability to operate either of the post incident system pumps. The majority of pump failures are due to failure to start faults (with very limited contribution from failure of the pumps to run). If the pumps were to fail to start, core spray would continue in the injection mode. If the pumps were to fail to run, manual action to maintain adequate core cooling could be accomplished by returning to injection by use of the fire pumps or feedwater/condensate. Again, this recovery action (to support repair and eventual return to service of the pumps) has not been accounted for in the PRA.

- Check valve VPI 300 is a containment isolation valve in the post incident system pipe that enters containment and connects to the core spray lines. It is the only valve which must open in order to establish recirculation to the reactor (using the post incident system) other than the core spray valves. All other valves are pre-positioned to permit drawing water from the containment and returning it to the reactor. Should VPI 300 fail to open, adequate core cooling could be maintained for a significant period by continuing to use injection systems.

- Components associated with post incident heat removal do not contribute significantly to this accident class. These components include the fire pumps and motor operated valves to the post incident heat exchanger. The fire pumps have diverse drivers (one motor-driven and one diesel-driven) which limits the potential for common cause failure. Furthermore, operation of the heat exchanger is not required for heat removal purposes but to maintain adequate NPSH for the core spray pumps. Should containment water temperature be too high for post incident system pump operation, the operator must only return to the injection mode of core spray. Raising the water level in containment may restore pump NPSH.

<u>Contribution of Operator Actions (Class IIIC-PI)</u>

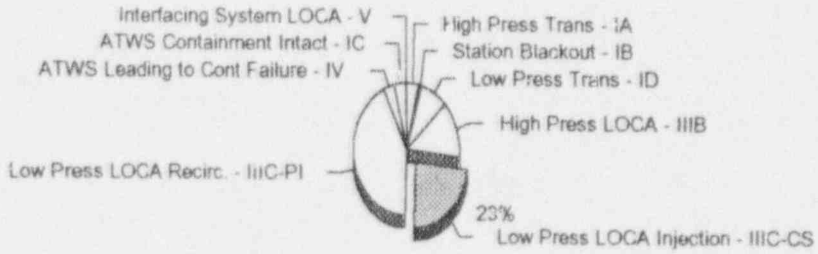
Although the post incident system is manually initiated, operator errors in establishing the system do not contribute significantly to this potential for core damage. This is due to the long time-frame available to the operator to align the system. Approximately 4 hours is required to fill the containment to the level at which the operator is instructed to initiate post incident recirculation, even for a large LOCA. As much as a day is required to raise the containment level to the elevation of the core. Injection to containment can continue during this period as the source of water to the fire pumps for injection to the reactor is unlimited.

The most significant human action related contributors to this accident class are therefore associated with pre-initiator alignment of components. The post incident system is flow tested during plant shutdown conditions. In addition, the heat exchanger tube bundle is leak tested on a monthly basis while the plant is at power. Closure of manual valves from the suction lines within containment and into containment is performed in order to facilitate these tests. Restoration of these valves is important for successful operation of the system. As most are located in the containment, they would not be recoverable after an accident. Recognizing this, each of these valves is provided with position locks. On restoring the system to service, the valves can only be locked if they are in their correct fully open or fully closed position.

Operator Action	Contribution to Class IIIC-PI
Restore VPI-004	6%
Restore System after TSD-07 (Core Spray Flow Test)	2%
Initiate Post Incident System	2%

3.2.2 Accident Class IIIC-CS (LOCAs with Failure of Reactor Injection) 1.3E-5/Year

Accident Class IIIC-CS is made up of primary system pipe break initiators in which the reactor is successfully depressurized but injection to the reactor by use of core spray or feedwater/condensate is insufficient to provide adequate core cooling. Core damage is assumed to occur relatively early in the event (within several hours). Accident Class IIIC-CS makes up 23% of the total core damage frequency for internal event initiators.



Accident Class Characteristics

At the onset of a primary system pipe break, a reactor trip will occur on high containment pressure, low reactor level or manual scram in response to annunciators and other indications of a LOCA. If the break is small enough, the operator will attempt to maintain drum level with the feedwater system. If the feedwater system is insufficient, reactor level will fall, possibly reaching the low drum level and low reactor level setpoints. At this time, a small LOCA would require RDS actuation to enable core cooling by a low pressure system. A medium to large LOCA will depressurize the reactor without RDS.

Two low-pressure systems are available to provide adequate core cooling: core spray and condensate. The core spray valves open automatically on low reactor level (2' 9" above the core) and low reactor pressure (200 psig). The distribution from either of the two spray lines is sufficient to provide adequate core cooling for the entire break spectrum, including a large break located below the core.

Even if core spray should fail, condensate may be sufficient to provide core cooling. The break must be sufficiently small such that flow diversion is less than condensate flow. Otherwise, the break must be above the core to permit reflood of the vessel. As the hotwell inventory is limited, operation of condensate may require makeup to the hotwell from the fire system. Conditions in which fire water makeup to the hotwell is required are assumed to be large breaks above the core as well as very small breaks below the core (which require continuous injection above decay heat loads).

Key Assumptions (Class IIIC-CS)

The following is a summary of assumptions which influence the quantification of accident sequences in Class IIIC-CS.

- Condensate makeup is adequate to provide core cooling for all steam line breaks. As the break location is well above the core, flow diversion through the break is not a concern in determining the condensate success criteria. Due to the depressurization rate, large steam line breaks are assumed to require fire water makeup to the hotwell to assure rapid reflood of the core. Only very small steam line breaks require depressurization by RDS or two loops of the emergency condenser to enable condensate flow.
- Condensate makeup is adequate for only a portion of LOCAs located above the core. Nearly 70% of the piping below the steam drum is assumed to result in diversion of condensate

flow before it reaches the core (such as the downcomers). Fire water makeup to the hotwell is assumed to be required for large breaks above the core to assure rapid reflood of the core following blowdown.

- Condensate makeup is not considered sufficient for LOCAs located below the core (due to the potential for flow diversion) with the exception of very small breaks. For very small breaks below the core, makeup to the hotwell from fire water is assumed to be required in order to sustain long term injection from the condensate system.
- Enclosure sprays are able to maintain containment conditions within the environmental qualification envelope for a limited set of core spray equipment during a medium steam line break. Reactor level and pressure instrumentation responsible for core spray valve operation may encounter temperatures in excess of their qualification if enclosure sprays are not successful. Other equipment associated with the ECCS remain within qualification temperatures even without the operation of enclosure spray (such as the core spray motor operators and RDS components). For these sequences, manual actuation of back-up enclosure sprays or the core spray valves themselves will ensure adequate core cooling. As these operator actions are assumed to be conditional, credit is taken only for manual initiation of the core spray valves in these sequences.

Contribution of Initiating Events (Class IIIC-CS)

The following summarizes the contribution of pipe break initiators to this accident class. Contribution by both break size and location is considered.

LOCA Size	Contribution to Class IIIC-CS
Very Small	10%
Medium	44%
Small	35%
Large	11%

- While potentially the more likely of all break sizes, very small breaks do not contribute as much to the IIIC-CS Accident Class as do small and medium breaks. As the break is very small, the normal means of reactor shutdown is possible without losing primary system level or requiring depressurization of the reactor with systems such as RDS.

Feedwater by itself is sufficient to make up for breaks of this size, thus minimizing the need for core spray.

- Small and medium size breaks constitute nearly 80% of the contribution to this accident class. This contribution is largely due to the assumed frequency of these pipe breaks as compared to that for large LOCAs. Medium size breaks contribute more than do the small break sizes. This is a result of the feedwater system's ability to provide makeup to the reactor for small breaks below the core, whereas medium breaks located below the core do not credit either feedwater or condensate.
- Large breaks constitute only 11% of the core damage frequency due to this accident class. This contribution is roughly in proportion to the relatively low potential for breaks of this size.

Location of Break	Contribution to Class IIIC-CS
LOCA Below Core	57%
LOCA Above Core	26%
SLB Inside Containment	16%
SLB Outside Containment	-

- Steam line breaks outside containment do not significantly contribute to Accident Class IIIC-CS for two reasons. First, the length of piping outside containment is only a small fraction of the total piping normally pressurized to reactor conditions. Second, the operators can take action to terminate reactor blowdown from this pipe rupture initiator simply by closure of the MSIV. Isolation of the break, in this manner, prevents further reactor inventory loss and allows the plant to be cooled in a more conventional manner (such as through the use of the emergency condenser).
- Steam line breaks inside containment constitute nearly half of the pipe break frequency, yet constitute only 16% of the total core damage frequency for this class. The ability to reflood the reactor with condensate (in addition to core spray) reduces the risk significance of these initiators. The contribution of the need for enclosure spray during medium breaks (to ensure maintenance of equipment qualification) raises the significance of steam line breaks slightly.

- The contribution of LOCAs above the core for this accident class is similar to the CDF from steam line breaks even though there is much less piping associated with the initiator. The potential for diversion of flow from condensate makeup results in a greater potential for core damage at this location than for breaks in the steam line.

- LOCAs located below the core constitute more than half of the core damage frequency for Class IIIC-CS. Condensate is not credited for reflooding the reactor for any breaks in this location with the exception of very small break sizes.

Contribution of Hardware Failures (Class IIIC-CS)

Failures of hardware associated with core spray and the fire protection system constitute the majority of the failures for reactor injection failure following a LOCA. Contributors to this accident class from feedwater and condensate are substantially less than for core spray and the fire system. The low contribution of condensate is largely due to the relatively restrictive assumptions regarding its capability (it can be effective in only a portion of LOCA break locations due to the potential for flow diversion) and because the system must be initiated manually in a relatively short time frame.

Components	Contribution to Class IIIC-CS
Core Spray Valves (Common Cause)	58%
Fire Pumps	20%
Condensate	2%
Enclosure Spray	2%
Firewater to Hotwell MOVs	1%

- The core spray valves include four MOVs: two dc-powered in the line to the core spray sparger and two ac-powered in the line to the core spray nozzle located in the vessel head. As all four MOVs are normally closed, there are four combinations of pairs of MOVs that would result in the failure of the core spray system. The majority of the contribution of the core spray valves to this accident class can be attributed to common cause failures in both trains.

- Loss of the fire pumps also contributes to this accident class, but is less significant given the diversity in the design of the pumps: one is motor driven, the other is diesel driven. However, the significance of the fire pumps is raised in this accident class in that they are also necessary for successful operation of feedwater and condensate in order to accomplish makeup to the hotwell.

- As noted above, condensate makeup to the reactor is a manually initiated system and is assumed to be effective in reflooding the reactor for only a portion of the pipe break spectrum. The risk significance of the hardware in this system, as well as the MOVs for makeup to the hotwell from the fire system, is small for these reasons.
- Enclosure sprays are required in order to maintain the environmental qualification envelope for a subset of instrumentation required for core spray operation. As the core spray valves do not require the operation of enclosure sprays and since the operator can manually initiate core spray valves should the automatic signal fail, the contribution of enclosure spray failure to the risk associated with this accident class diminishes.

Contribution of Operator Actions (Class IIIC-CS)

The core spray system is automatically actuated. Fire pumps start on low drum level and core spray valves open on low reactor level and pressure. The majority of operator actions in this accident class are associated with maintaining injection with the feedwater/condensate system.

Operator Action	Contribution to Class IIIC-CS
Trip Condensate Pumps	7%
Start Condensate Pumps	1%
Hotwell Makeup	1%
Manual Core Spray Initiation	1%

Key operator actions in this accident class are in response to the initiating event combined with failure of automatically activated components.

- Three actions are associated with condensate pump operation.

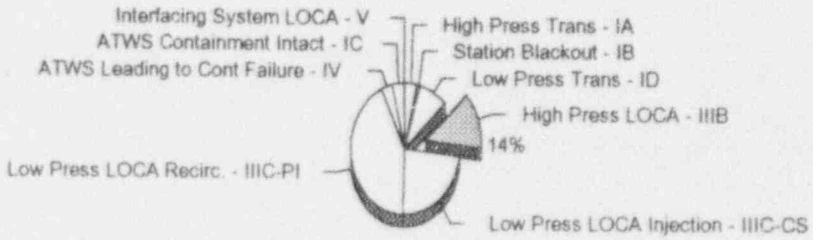
  The first occurs on depletion of the hotwell inventory. On low hotwell level, the condensate pumps trip. As makeup from the CST is provided, the permissive for pump breaker closure occurs. The pump will trip again, shortly thereafter, due to the depleted hotwell inventory. Cycling the pump and breaker in this manner is assumed to lead to failure of the breaker and inability to start the condensate pumps once makeup to the hotwell from the fire system is aligned. Operator action to trip the pumps on low hotwell level precludes this failure.

The second action is to establish firewater makeup to the hotwell (and the third is to restart the condensate pumps for injection to the reactor after depressurization). The significance of each of these actions is greater than that for hardware failures associated with the condensate system. However, the overall importance of these actions is diminished by the limited fraction of breaks over which condensate operation can be effective and due to the need for fire pump operation for the success of both core spray and condensate.

- The last operator action is manual initiation of the core spray valves due to exceeding the environmental qualification limits for reactor level and pressure instruments. As this action requires failure of automatic initiation of the enclosure spray system, its importance to risk is limited.

3.2.3 Accident Class IIIB (LOCAs with Failure of Reactor Depressurization)

Accidents initiated by pipe rupture, in which core damage occurs with the reactor at high pressure, constitute only 13% of the core damage frequency due to internal event initiators. RDS failure must occur coincident with a small or very small break in the primary coolant system to result in these conditions.



<u>Accident Class Characteristics</u>

At the onset of a break, in this classification, the reactor will trip either by the operator (small above core LOCA, very small and small below core LOCA or the very small steam line break) or automatically on low drum level (isolated small, medium and large breaks outside containment). Due to the break location (inside or outside containment), different accident characteristics occur.

For small breaks inside containment, the feedwater system must be unavailable as a source of makeup water before reactor water level falls. Only small breaks inside containment can lead to Class IIIB sequences as medium and large breaks will result in successful reactor depressurization. Without the feedwater system, reactor depressurization through the break must be augmented in order to allow use of the low pressure makeup systems. Depressurization can be accomplished by either the emergency condenser or by the automatic actuation of the RDS. Due to its capacity, the emergency condenser is only credited in the very small steam line break transient. Actuation of the emergency condenser (for this initiator) obviates the need for RDS. Regardless of whether the emergency condenser actuates, primary system inventory will gradually decrease to the automatic RDS setpoints (the final setpoint being a water level of 2' 9" above the top of active fuel). At this time, reactor pressure is above the core spray valve setpoint (or condensate pump shutoff head). With a failure of the RDS, the level will continue to fall as the reactor slowly depressurizes, the core will become uncovered while the reactor is at an elevated pressure (above 200 psig).

For pipe breaks outside containment, the plant response is different. Small, medium or large breaks may contribute to this accident class. As the operators become aware of the break (primarily for the small break), attempts would be made to isolate the primary system blowdown by closing the main steam isolation valve (MO-7050). For the medium and large breaks, isolation is automatic as reactor water level will quickly drop to the containment isolation setpoint of 2' 9" above the top of the fuel. The RDS actuation logic requires only the remaining signals from a successful fire pump start and the timing out of the 2 minute timers in order to open the depressurization valves. However, if the MSIV successfully closes, without RDS, the reactor will begin to repressurize and eventually exceed the fire pump shutoff head, thus curtailing core spray flow. Should the RDS fail to actuate, pressure control could be accomplished by the emergency condenser. Since the initial break was outside containment, normal operating conditions within containment are expected. For these sequences, the emergency condenser and the RDS must fail in order for core damage at high pressure to occur.

Key Assumptions (Class IIIB)

The following summarizes key assumptions which influence the quantification of accident sequences in Class IIIB.

- Feedwater is a viable means of providing makeup to the reactor. The ability to perform this function depends on the size and location of the pipe failure. Feedwater is assumed to be successful in making up to the reactor and avoiding RDS only if: 1) the break is very small, 2) the break is not located in a section of pipe that will cause flow to be diverted before it can reach the core (such as the downcomers), and 3) makeup to the hotwell with the fire system is established.

- Only small or very small breaks can lead to a situation in which RDS is necessary. Larger breaks result in depressurization of the reactor without the need for RDS.
- Success criteria for reactor depressurization for very small breaks assumes three of the four trains are necessary for the limiting break size. Breaks in the small LOCA ranges require only two of four trains, as the break assists in RPV depressurization.
- The very-small LOCA and steam line break initiators are assumed to result in the exceedance of the equipment qualification limits of the emergency condenser valves if the reactor is not tripped manually. This is due to the continuous ventilation of the recirc pump room which would prevent containment pressurization. Manual trip of the reactor would preclude heat-up of containment by closing the vent valves and allowing containment to pressurize to the enclosure spray setpoint.
- Manual actuation of both loops of the emergency condenser, with eventual shell side makeup, permits cool-down of the reactor for very small steam line breaks and small breaks above the core such that the need for RDS is avoided.

<u>Contribution of Initiating Events (Class IIIB)</u>

As in the case with other LOCA initiated accident classes, a spectrum of break sizes and locations is postulated based on their effect on system success criteria.

Break Size	Contribution to Class IIIB
Very Small Break	75%
Small Break	25%
Medium Break	-
Large Break	-

- Only small and very small breaks are contributors to this accident class. Larger breaks result in depressurization of the reactor without the need for RDS. Small and very small breaks constitute a large fraction of the pipe break frequency, however.

Break Location	Contribution to Class IIIB
LOCA Below Core	79%
Steam Line Break	14%
LOCA Above Core	7%

- Steam line breaks contribute less to core damage for this accident class than LOCAs as adequate core cooling and RPV depressurization can be more easily provided by feedwater or the emergency condenser in addition to RDS and core spray. Very small steam line breaks resemble transient (non-LOCA) initiators in this regard.

Contribution of Hardware Failı.res (Class IIIB)

As expected, most of the risk of this accident class is associated with RDS equipment. The majority of failures include isolation and Target Rock valve components. Electrical power or instrument and control contributions are limited due to a high degree of redundancy in the system.

Components	Contribution to Class IIIB
Target Rock Valves	49%
Isolation Valves	24%
Fire Pumps	1%

- Target Rock valve contribution to risk includes both random and common cause failures of the relief valves themselves as well as the pilot valves. Isolation valve contribution includes the air operated valve failure-to-open in addition to the solenoid valve failure-to-operate failure mode. Common cause, as well as random contribution, is included. Target Rock valve contribution to system failure is slightly greater than the isolation valves as a result of plant-specific operating experience with the two valve types.
- The fire pumps are limited in their contribution to this accident class due to the diversity in their design (one motor driven and one diesel driven pump).

Contribution of Operator Actions (Class IIIB)

The RDS is automatic and no operator actions contribute significantly to risk with respect to its operation. However, other manual interventions contribute to a large fraction of this accident class.

Operator Action	Contribution to Class IIIB
Condensate Pump Trip	31%
Makeup to Hotwell	6%
Restart Feedwater Pump	2%

- Feedwater pump operation is assumed to be necessary in order to prevent a demand on RDS during a very small break. Actions leading to a loss of feedwater contribute to this accident class as a result. These actions include trip of the condensate pumps on low hotwell level to prevent breaker cycling and possible failure, makeup to the hotwell, and restarting the feedwater pumps once makeup is established.

3.3 Accident Classes Initiated by Transients (Non-LOCA Initiators)

Several accident classes are made up of core damage sequences which begin with the primary system intact as opposed to a passive pipe rupture. Initiating events in these accident classes range from those which are relatively frequent (occurring as often as one or more times a year) to very infrequent events involving the potential loss of entire support systems required for plant operation. Given that the primary system is intact, systems important to the outcome of sequences in this accident class are required to remove and make up for decay heat at high reactor pressure as well as accomplish reactor depressurization and provide adequate core cooling with low pressure systems, if necessary.

Three accident classes constitute this portion of the core damage frequency for Big Rock Point. Their combined frequency is less than 8E-6/year or approximately 13% of the total for internal events.

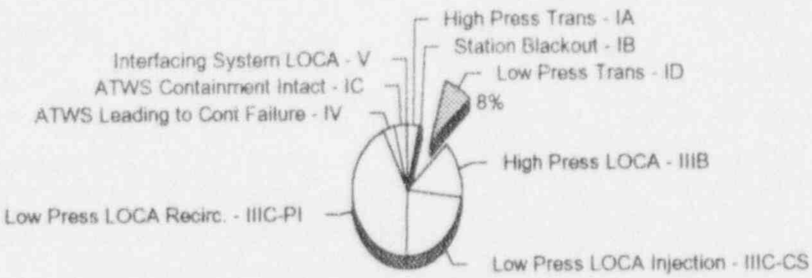
This section describes important features of the Big Rock Point design which result in the relatively low risk contribution associated with transient initiated events. Key assumptions regarding system success criteria (as related to plant response to transient initiators) are discussed as a part of each accident class.

3.3.1 Accident Class ID (Transients with Reactor Depressurization But No Inventory Makeup) 4.5E-6/Year

This accident class is characterized by transient initiators in which reactor trip is successful, but multiple failures occur which lead to either the inability to condense steam with the emergency condenser or, if heat is being removed by the main condenser, failure to make up for decay heat losses by use of high pressure systems such as feedwater or CRD. A gradual loss of reactor inventory would occur (under these conditions) until primary system level reached the RDS setpoint of 2' 9" above the core.

In Accident Class ID, reactor depressurization is accomplished successfully, but makeup with low pressure systems (such as condensate or core spray) is assumed to fail. Successful depressurization, without the ability to continue to remove decay heat or to provide makeup (by use of any of the normal or emergency core cooling systems), is assumed to result in core damage with the reactor at low pressure.

Accident Class ID constitutes approximately 8% of the total core damage frequency for internal events.



Accident Sequence Characteristics

Sequences in this accident class begin with an initiating event that causes a turbine trip or reactor scram. On reactor shutdown, steam may be condensed in the main condenser, or the reactor may be isolated from the main condenser. If the main condenser is still in service, steam will exit the primary system at decay heat levels through either the bypass valve or the bypass warm-up line. Makeup for decay heat losses would be accomplished with either the feedwater or CRD systems. If both of these makeup sources were to fail, reactor inventory would gradually fall to the RDS actuation setpoint.

If isolated from the main condenser, primary system pressure will rise until the setpoint for emergency condenser initiation is reached (1435 psig). In Accident Class ID, the emergency condenser is assumed to be unavailable. This may be due to one of two causes: either the outlet valves from the emergency condenser fail to open, or the emergency condenser is successfully placed in service but long term makeup to the shell is not initiated. Without the emergency condenser, pressure would rise to the steam drum safety relief valve setpoint of 1535 psig. Steam relief through a cycling SRV begins to lower the reactor inventory if feedwater is also unavailable. Gradual depletion of the primary system to 2' 9" above the core would occur, at which time RDS actuation would be required.

Timing of this event is driven by the large amount of inventory above the core at the time of reactor trip. Whether steam is being directed to the main condenser through the bypass line or to the containment through SRV operation, approximately 3 hours would elapse prior to reactor inventory depletion to the top of the fuel. If emergency condenser initiation is successful, approximately 6 hours of decay heat removal is available in the emergency condenser shell before reactor pressure would rise to the SRV setpoint. If emergency condenser makeup failure were the reason for loss of decay heat removal, more than 9 hours would elapse (following reactor trip) prior to reactor inventory being depleted to the RDS setpoint.

Key Assumptions (Class ID)

The following is a summary of assumptions which influence the quantification of sequences in the ID Accident Class:

- Events leading to the gradual loss of reactor inventory to the main condenser require failure of both feedwater and CRD pumps in order to deplete the primary system to the point at which RDS is initiated. A single CRD pump (by itself) is sufficient to make up for decay heat losses.
- CRD is not credited for sequences in which steam drum SRVs are cycling. The CRD pumps are located inside containment and would be inoperable in the steam environment or rising water level that would accompany SRV operation.
- Credit for use of the alternate shutdown portable pump has not been assumed in sequences in which emergency condenser makeup failure occurs. A significant fraction of emergency condenser makeup failures includes loss of the fire pumps. Crediting this mode of emergency condensate makeup would lower the contribution of this accident class to core damage.
- Makeup to the hotwell from the condensate storage tank is more than sufficient to accommodate decay heat losses if feedwater/condensate is used for injection to the reactor.

Should CST makeup fail, credit has not been taken for alignment of firewater to the hotwell.

- On actuation of RDS, the EOPs require that reactor level is maintained near normal by use of low pressure systems such as condensate or core spray. Recirculation is not required under these conditions as inventory addition to containment is limited to decay heat rates.
- Long term heat removal from containment does not require operation of the post incident system. Long term overpressurization of containment due to decay heat does not occur as a result of the passive heat sinks available for decay heat removal.

Contribution of Initiating Events (Class ID)

A series of potential transient initiating events was considered in the Big Rock Point PRA. The initiators range from those which may occur one or more times a year to very rare initiators which involve loss of entire support systems and have not occurred over the life of the plant. The contribution of each of these initiators to Class ID is as follows:

Initiating Event	Contribution to Class ID
Loss of Instrument Air	74%
Manual Shutdown	14%
Loss of Station Batteries	5%
Turbine Trip	4%
Loss of Off-site Power	2%

- Loss of instrument air is the largest single contributor to this accident class. Instrument air provides support for the operation of the main condenser, makeup to the emergency condenser, and makeup to the hotwell from the condensate storage tank. The loss of both of these heat sinks and the feedwater system for long term makeup results in the risk significance of this initiator. Without instrument air, the plant is dependent on the fire protection system to prevent SRV operation by maintaining emergency condenser makeup. Conservatively ignored in the quantification of the loss of instrument air accident sequences is the alternate shutdown portable pump, which can be used to maintain emergency condenser shell inventory, or firewater makeup to the hotwell. Credit for this source of decay heat removal would significantly lower the frequency of this transient's contribution to core damage.

- Together, manual shutdown and turbine trip contribute nearly 20% to this accident class frequency. Neither of these initiators impacts the operability of heat removal or makeup systems. However, their frequency of occurrence is greater than all other initiators which results in their contributing to transient initiated core damage frequency more than most other events.
- Loss of station batteries is a relatively rare event (at less than 5E-2/year) and contributes less than 1% to the total CDF. However, dc power has a pervasive effect on many systems. While the main condenser and feedwater will continue to operate without station batteries, an orderly shutdown is required as safety related equipment relies on dc power for operation. Once the reactor is cooled, eventual termination of feedwater is required. Restoration of these pumps cannot be performed without dc power. If the operating CRD pump becomes unavailable, the reactor must be isolated and the emergency condenser placed in service. Failure to activate the emergency condenser will lead to depletion of the inventory in the reactor and RDS actuation. One train of core spray valves and the diesel fire pump are available for core cooling.
- Loss of off-site power is more frequent than the preceding two initiators, yet has a very small contribution to Class ID. The ability of Big Rock Point to sustain a load rejection and continue operation (with the main generator providing house loads) reduces the significance of this initiator.

Contribution of Hardware (Class ID)

The single most important system to this accident class is the emergency condenser. Operation of this system will remove decay heat and prevent reactor inventory loss, regardless of the transient initiator, as long as the primary system is intact.

Components	Contribution to Class ID
Fire Pumps	83%
Emergency Condenser Makeup	5%

- The fire pumps constitute most of emergency condenser support system importance. The majority of the initiators in this category are loss of instrument air as demineralized water is not available. Fire water makeup is necessary to maintain long term operation of the emergency condenser. Fire water also supports core spray operation should the emergency condenser become unavailable to maintain reactor inventory.
- Emergency condenser makeup is dominated by SV-4947 failure, which is nearly 4% of the total contribution to Class ID. In fact, this valve can be operated manually inside the containment if it fails to energize.
- The failure of the emergency condenser outlet valves contributes only a small fraction to this accident class. With two emergency condenser loops, either one being sufficient to remove decay heat, the contribution of emergency condenser valve failure leading to core damage is limited.

Contribution of Operator Actions (Class ID)

The systems most important to reactor inventory control following a reactor trip are those that are already in service (such as the main condenser, feedwater and CRD) or are automatically actuated (emergency condenser). For this reason, few operator actions contribute to this accident class.

Operator Action	Contribution to Class ID
Manual Pressure Control with RDS	84%
Manual Initiation of Emergency Condenser	5%
Failure to Restore VEC-112 After Maintenance	4%
Align Demin Water Pump to Bus 2B	1%

- The largest single operator action contributing to Class ID is manual reactor pressure control using CV-4184 and a Target Rock valve. This action is required by the EOPs only after both heat sinks have failed. Using the RDS spool piece warming line as a means of decay heat removal prevents a demand on SRVs, thus limiting the potential for a stuck open SRV (which leads to a LOCA) and a subsequent heat-up of the containment (which would preclude CRD pump operability).

In fact, the probability of this operator error is conservatively high. The value (0.15) was developed as conditional on other operator actions to establish a heat sink, such as opening CV-4106 to the main condenser or initiating fire water makeup to the emergency condenser shell. As the operator is very likely to take these other actions, they do not contribute significantly to risk; hence, operator failure to initiate RDS for pressure control is over-estimated.

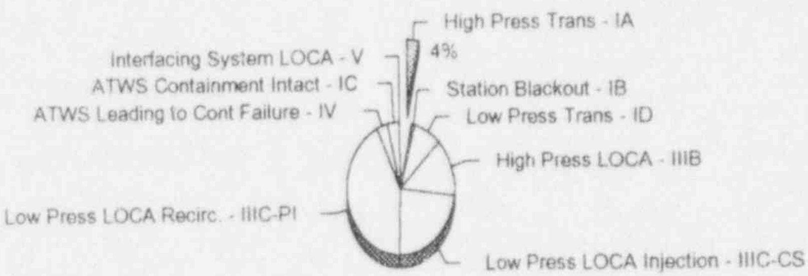
- Manual action to initiate the emergency condenser contributes to only a small fraction of this accident class. The emergency condenser is automatic on high reactor pressure (1435 psig). The sequences in which manual initiation is useful are those in which the main condenser remains in service and high reactor pressure does not occur. For the emergency condenser to be required in these sequences, makeup from both feedwater and CRD are assumed to fail, thereby reducing the significance of this operator action. Alternatively, the operator may manually initiate MSIV closure. Two to three hours is available to perform these actions at decay heat levels.
- One pre-initiator operator action that contributes to a limited portion of this accident class is failure to restore the emergency condenser shell level instrumentation after surveillance testing. This error would effectively disable the automatic makeup capabilities of the demineralized water system and eliminate indication of shell level in the Control Room.
- A minor contributor to this accident class is the alignment of the demin water pump to the emergency bus during an extended loss of off-site power condition. The importance of this action is limited as the operator can align other sources of makeup to the emergency condenser shell during loss of ac power conditions in the form of fire water makeup. Fire water addition can be accomplished from the Control Room.
- Manual initiation of fire water to the shell of the emergency condenser contributes less than 1% to this accident class. The time available for the operator to take this action is sufficiently long (over 6 hours) that it does not contribute to the core damage frequency.

3.3.2 Accident Class IA (Transients with Failure of Reactor Depressurization) 2.0E-6/Year

This accident class is characterized by transients in which a reactor trip occurs and a series of failures leads to the inability to condense steam with either the main condenser or emergency condenser. Reactor inventory is gradually depleted to the RDS actuation setpoint.

In Accident Class ID, the reactor depressurization system fails to operate. With the reactor remaining at an elevated pressure, inventory makeup with low pressure systems (such as condensate or core spray) cannot occur. The water level continues to drop until the active fuel region of the core is uncovered. Core damage is assumed to occur with the reactor at high pressure.

Accident Class IA contributes only 4% of the total internal events core damage frequency at Big Rock Point.



Accident Sequence Characteristics

Sequences in this accident class develop very much like those described for Class ID. Following a reactor and turbine trip, steam will flow to the main condenser through the turbine bypass line or, if the reactor is isolated, reactor pressure will rise to the emergency condenser valve setpoint (1435 psig). If steam is being condensed in the main condenser, this accident class requires that makeup from both feedwater and CRD fail. If the reactor is isolated from the main condenser, it is assumed that the emergency condenser outlet valves must fail to open or that makeup to the emergency condenser is unavailable.

In either scenario, steam generation due to decay heat causes a gradual loss of inventory in the primary system until reactor water level reaches 2' 9". For sequences in which the main condenser is in service and those in which the emergency condenser valves fail to open, 2 or more hours of decay heat boil-off is required in order to reach the RDS setpoint. For sequences in which the emergency condenser is successfully placed in service but makeup to the shell is not available, over 6 hours of decay heat removal will be provided before steaming and inventory loss through the steam drum safety relief valves would occur. More than 9 hours is required before inventory loss to the RDS setpoint exists in these sequences of events.

On depletion of reactor inventory to 2' 9" above the core, RDS operation would depressurize the reactor, thus allowing injection and reflood with core spray or the condensate system. In Accident Class ID, reactor depressurization fails and reactor inventory depletion continues without the ability for makeup to the reactor by use of low pressure systems.

Key Assumptions (Class IA)

The difference between Accident Class IA and ID is the assumption that RDS is unavailable for this class of events whereas, in Class ID, RDS actuation occurs successfully. Up to the point at which reactor inventory would reach the low reactor level setpoint, the timing of events and assumptions regarding this accident class are similar to those of Class ID.

- Both feedwater and CRD must fail in order to deplete reactor inventory with the main condenser in service. A single CRD pump is more than sufficient to make up for decay heat.
- Even though it is capable of makeup for decay heat, the CRD pump is assumed inoperable in sequences in which the steam drum SRVs are assumed to cycle. The CRD pumps are located inside containment.
- For sequences in which emergency condenser makeup does not occur due to fire pump failures, credit for the portable pump in the alternate shutdown building has been conservatively ignored. Considering this means of makeup further lowers the contribution of this accident class to core damage.
- Credit has not been taken for firewater makeup to the hotwell to maintain feedwater operation. Only makeup from the CST has been assumed.
- Because inventory addition to containment takes place at decay heat rates, initiation of recirculation with the post incident system is not required. Due to the rate at which water level rises in containment, it does not reach the 587 foot elevation for a significant period of time at decay heat levels.
- Pressurization of containment due to decay heat addition is limited because of passive heat removal through the containment shell and other passive heat sinks.
- The success criteria for RDS operation requires the operation of three of the four trains. This conservatism is based on the ECCS criteria for LOCAs. In fact, only one train of RDS is necessary to achieve depressurization for transient initiators with loss of both heat sinks.

Contribution of Initiating Events (Class IA)

Transient initiators and their contribution to this accident class are very similar to those in the ID Accident Class.

Initiating Event	Contribution to Class IA
Loss of Instrument Air	82%
Manual Shutdown	8%
Loss of Off-site Power	5%
Loss of DC	3%

- Loss of instrument air is the largest single contributor to this accident class. This initiating event affects the reliability of the two heat sinks; the main condenser and makeup to the emergency condenser are not available without instrument air. Also, makeup to the hotwell from the CST is assumed not to occur as CV-4009 requires air in order to open. This affects long term operation of feedwater for makeup to the reactor.
- Manual shutdown contributes somewhat to this accident class as a result of their frequency of occurrence. This initiator is assumed to occur several times each year.
- A loss of off-site power would lead to a loss of the main condenser and feedwater. This also affects emergency condenser makeup in that demineralized water must be loaded onto the emergency bus, as opposed to operating automatically. However, this initiator does not contribute significantly to the potential for core damage due to the ability of the plant to sustain a load rejection and continue operating to supply house loads (without a reactor trip) even if all off-site power is lost.
- The station batteries contribute slightly to this accident class because a manual shutdown is required on its occurrence. This affects restart of the feedwater and CRD pumps for high pressure makeup and operation of the electric fire pump for makeup to the emergency condenser. The UPS batteries supporting RDS actuation are independent of the station batteries; hence, this initiator does not contribute significantly to Class IA. The batteries which power the emergency condenser outlet valves and makeup to the emergency condenser are located in the alternate shutdown building and are also not affected by a loss of station batteries.

Contribution of Hardware (Class IA)

The largest single contributor to this accident class is the RDS. The emergency condenser and equipment which support makeup to the emergency condenser shell also play a significant role in preventing this accident class from occurring. Successful operation of the RDS or the emergency condenser assures reactor depressurization thus avoiding the potential for core damage at high reactor pressure.

Components	Contribution to Class IA
RDS Target Rock Valves	70%
RDS Isolation Valves	29%
SV-4947	37%
Fire Pumps	17%
Emergency Condenser Valves	4%

- The Target Rock depressurization and pilot valve common cause and random failures contribute to the largest fraction of this accident class. This is followed by the isolation valves and their solenoids. The Target Rock valves have a greater importance than the isolation valves as they can be used for manual pressure control thus avoiding demands on steam drum SRVs or the need for RPV blowdown and core spray.
- Makeup to the emergency condenser contributes to a large fraction of this accident class. Actuation of SV-4947 is required in order to establish fire water makeup to the emergency condenser shell during events in which the demineralized water system is not available (such as loss of instrument air).
- The fire pumps contribute to a relatively small fraction of this accident class. Loss of instrument air requires fire pump operability. Firewater is the only source of makeup to the emergency condenser shell for this accident class. Fire pump operation is also a permissive for automatic RDS actuation. Support for RDS actuation with the fire pumps is limited to their pressurization of the fire system.
- The emergency condenser valves contribute to a small portion of this accident class. Initiation of the emergency condenser requires only that one of the two dc operated MOVs open to permit condensation of steam and decay heat removal in one-of-two tube bundles.

Contribution of Operator Actions (Class IA)

Operation of the RDS is automatic. Therefore, no operator actions associated with its depressurization function contribute to this accident class. Operator actions considered in sequences associated with this accident class are similar to those discussed in Class ID. Use of the RDS to prevent a demand on steam drum SRVs and actions to assure main or emergency condenser operation constitute the operator action contributing most to Class IA.

Operator Action	Contribution to Class IA
Manual Pressure Control with RDS	85%
Failure to Restore VEC 112 After Calibration	26%
Manual Initiation of Emergency Condenser	7%
Manual Initiation of Fire Water Makeup to the EC	2%
Failure of Makeup from CST to Hotwell	1%

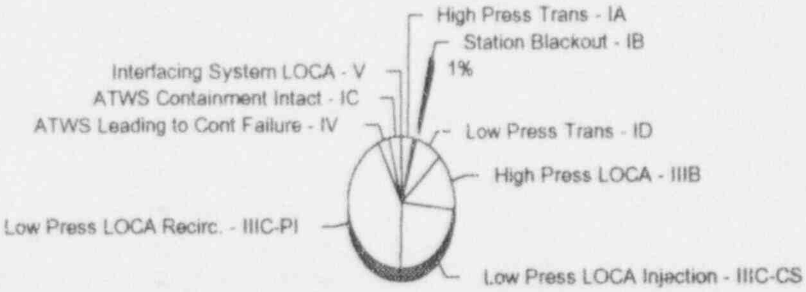
- The single largest contributor to Class IA from operator actions is manual pressure control with the RDS to prevent SRV operation after loss of both the main and emergency condensers. The probability of this operator action is conservatively high as it was developed assuming it would be conditional on the operator action to establish the main condenser or to initiate emergency condenser makeup.
- A pre-initiator operator action which appears in the results is associated with restoring emergency condenser level instrumentation after calibration and testing. This error is assumed to divert water from the emergency condenser (assuming makeup is required).
- Failure to initiate fire water makeup to the emergency condenser shell constitutes just over 2% of this accident class. The most significant initiator (loss of instrument air) raises the importance of fire water, as demineralized water makeup is not available.
- Manual initiation of the emergency condenser appears in a small fraction of Class IA sequences. Manual actuation is required for those events in which the main condenser is

available but makeup with both feedwater and CRD has been lost. Alternately the operator may initiate MSIV closure.

- Another pre-initiator operator action includes failure to place the makeup valve from the CST (CV-4009) to the hotwell in automatic. A few of the accident sequences lead to manual pressure control of the reactor with RDS using CV-4184 and a Target Rock depressurization valve. For these sequences, makeup to the reactor is required with the feedwater system. Without makeup to the CST, reactor inventory depletion may occur once the hotwell is empty. Credit for recovery of this error on falling hotwell level has been conservatively ignored.

3.3.3 Accident Class IB (Station Blackout) 5.1E-7/Year

An accident class often important to other PRAs involves accident sequences associated with station blackout conditions. In the Big Rock Point PRA, however, station blackout contributes less than 1% to the total internal events core damage frequency. The relatively low importance of this initiator can be attributed to a number of plant design features, several of which are unique to Big Rock Point.



Big Rock Point has two incoming transmission lines: a 138 kv line, to which the main generator is normally connected, and a 46 kv line, to which station loads can be transferred if both the main generator and the 138 kv line are unavailable. A loss of off-site power frequency of approximately 0.06/year is estimated for the Big Rock Point site, which is similar to the frequency of this event at other sites.

To minimize the potential impact of loss of the 138 kv line, the Big Rock Point Plant was designed with 100% turbine bypass capacity. On sensing decreasing voltage on the 138 kv line, a tone control relay sends an anticipatory signal to the bypass valve to open and the turbine admission valves ramp closed such that the main generator is producing 5% power or just over that required to carry house loads. With this load rejection capability, Big Rock Point can sustain a total loss of off-site power and continue operating without the occurrence of a reactor scram or turbine trip.

Even if the load rejection were unsuccessful and a reactor trip occurred, the emergency condenser will provide decay heat removal capability. The emergency condenser consists of two tube bundles, each capable of removing the equivalent of 5% reactor power with the inventory initially available in the shell at normal operating conditions. The size of each tube bundle is such that steam drum SRV operation is precluded by successful operation of the emergency condenser. The tube bundles are placed in service by the automatic opening of a dc powered MOV in the condensate return line from the emergency condenser to the steam drum. Once in service, heat is removed by natural circulation, steam entering the tubes of the emergency condenser and the condensate being returned to the steam drum. The inventory normally available in the emergency condenser shell is sufficient to remove more than 6 hours of decay heat before the primary system would reheat to the steam drum safety relief valve setpoint.

For long term operation of the emergency condenser (beyond its initial 6 hour inventory), makeup to the shell can be provided. Makeup rates allow heat removal from the reactor at 2% to 3% of reactor power. During a blackout, water can be supplied from the fire system by use of the diesel fire pump or by connecting a portable diesel-powered pump which is provided with the alternate shutdown building. The supply for both of these pumps is indefinite (each of them draws suction from Lake Michigan). Makeup is provided through a dc powered solenoid valve which connects the fire system to the emergency condenser shell. The solenoid valve can be remotely operated from the Control Room or the alternate shutdown panel.

Even with the capability of decay heat removal with no ac power, the plant is also provided with two diesel generators. Both diesels are approximately 250 kw in capacity. Because of the ability to remove decay heat for more than 6 hours, only one of the two diesels starts automatically. The standby diesel generator can be started manually should the emergency generator become unavailable. The function of the diesel generators during a loss of off-site power would be to provide power for ac powered instrumentation (useful in monitoring plant conditions) and to maintain battery charging.

Three separate sets of batteries are included in the BRP design. The station batteries have a capacity of 4 hours during blackout conditions and provide power for annunciation, various instrumentation and breaker control, and small emergency loads (such as the core spray valves). The UPS batteries supply power to RDS solenoid valves, instrumentation and control. The UPSA battery supplies power to the EDG output breaker (to close to bus 2B). These batteries last approximately 5 hours without charging. The alternate shutdown panel provides power for the MSIV, emergency condenser outlet valves, and a solenoid valve (which is operated to provide fire water to the emergency condenser shell). To meet Appendix R requirements, these batteries are required to have a 72 hour capacity and, under expected blackout loads, can last more than a week without battery chargers. The emergency diesel generator, the standby diesel generator and the diesel fire pump have their own batteries and are independent of station, UPS and alternate shutdown power supplies.

Because of the ability to remove decay heat and maintain reactor inventory with the emergency condenser, and because maintenance of emergency condenser heat removal is independent of ac power, the Big Rock Point Plant can effectively cope with a station blackout for about a week. Operator actions required during this period are minimal and are directed at providing makeup to the emergency condenser shell. For these reasons, station blackout is not a significant contributor to risk at Big Rock Point.

Key Assumptions

Quantification of the station blackout event tree credited many of the design features provided in the Big Rock Point plant design for coping with a blackout. A number of assumptions that were made in constructing and quantifying the event tree sequences and which impact the results include:

- The capability to ride out a total loss of off-site power, without requiring a reactor trip, was credited in the analysis. The ability to sustain a load rejection was assumed to have a reliability of 90%. Hence, failure of load rejection and a subsequent reactor and turbine trip was assumed to occur once in ten events.
- The RDS and core spray systems were assumed to play a limited role in providing adequate core cooling during a blackout. This assumption was based on the battery capacity for equipment required to operate in order to perform these system functions. UPS power supplies to support and maintain depressurization with the RDS were assumed to have a capacity of 5 hours. Core spray valves require the station batteries (which were assumed to have a 4 hour capacity). Both of these time frames are less than that which would occur if the emergency condenser were placed in service. Emergency condenser operation would result in more than 6 hours of successful decay heat removal before steam drum SRV operation would occur, even without makeup to the emergency condenser shell. Subsequently, another 3 hours would pass as reactor inventory gradually depleted (at decay heat rates) to low reactor water level. More than 9 hours would pass before the RDS and core spray systems would receive a signal to

actuate, at which time battery depletion would be expected, unless an ac power source had been restored.

- Little credit for the repair of failed equipment is considered in the station blackout sequences, in spite of the long time-period available for such activities. The only exception is credit for recovery of an off-site power ac source, which was based on plant specific experience with each of the off-site power sources at Big Rock Point (the 138 kv and 46 kv lines).

Contribution of Hardware Failures (Class IB)

The 138 kv and 46 kv lines and load rejection failure appear in 100% of the sequence cutsets for station blackout. The following examines the distribution of other plant systems and components to a station blackout.

Components	Contribution to Class IB
Emergency Diesel Generator	98%
Standby Diesel Generator	95%
SV-4947	57%
Steam Drum SRVs Fail to Close	37%
Diesel Fire Pump	16%
Portable Pump	5%
Emergency Condenser Valves	2%

- The emergency and standby diesel generators must fail (in addition to off-site power and load rejection) for a station blackout to occur. The majority of the contribution of the diesel generators is from failure to run, once successfully started. The diesel failure-to-run faults are more than 10% due to the long mission time assumed in the station blackout event.

- The emergency condenser is the single most important system which allows the plant to cope with a station blackout. Emergency condenser makeup contributes to the majority of the risk associated with this system. SV-4947 failures constitute approximately half of this contribution to emergency condenser failure. The diesel fire pump contributes significantly less as it is redundant to the Appendix R portable pump located in the alternate shutdown building. The emergency condenser outlet valves contribute only slightly to

CDF due to their reliability and because only one must open to be successful in removing decay heat.

- Steam drum relief valves play a role in approximately one-third of the core damage sequences associated with station blackout. It has been assumed that, subsequent to emergency condenser failure, there is a high potential for blowdown of the reactor due to cycling of an SRV and a failure to close. Depressurization of the reactor in this manner would minimize the time available to recover off-site power and has been accounted for in the quantification of the station blackout sequences.

Operator Action Contribution (Class IB)

The following operator actions were identified in examining the results of the station blackout quantification: the restoration of off-site power, establishing the standby diesel generator, and initiation of emergency condenser makeup.

- Off-site power recovery is evaluated based on the failure mode associated with the emergency condenser. More than 80% of the sequences for which off-site power is not recovered are associated with emergency condenser makeup failure. After 9 hours, (6 hours for emergency condenser to boil off and 3 hours to deplete the reactor to the top of the fuel) there is less than a 20% chance that off-site power cannot be recovered and thus prevent core damage. For emergency condenser valve failure, approximately 33% of accident sequences are assumed not to have recovered off-site power prior to reactor inventory depletion due to decay heat. Off-site power recovery factors are based on plant-specific experience with recovery of the 138 kv and 46 kv lines.

- Emergency condenser makeup requires manual initiation of SV-4947 to provide fire water makeup to the shell and to maintain long term operation of the emergency condenser. As this action is simple and can be taken from the Control Room, it accounts for only a few percent of the total risk of station blackout at Big Rock Point. Should the diesel fire pump become unavailable, operation of the portable pump in the alternate shutdown building can provide required makeup to the emergency condenser shell. This action contributes 11% to the risk of station blackout.

3.4 Accident Classes Initiated by Failure to Scram (ATWS)

Two accident classes have been defined for the purpose of examining Big Rock Point Plant response to a failure to scram. Initiating events associated with these accident classes include only the more frequent transient initiators:

Initiating Event	Frequency
Turbine Trip	1.1/year
Loss of Off-site Power	0.06/year
Loss of Feedwater	0.05/year
Spurious Bypass Valve	0.049/year
Loss of Main Condenser	0.045/year
Loss of Instrument Air	0.045/year
MSIV Closure	0.018/year

The reliability of the reactor protection system is such that the contribution of less frequent initiating events to the risk of core damage is small.

The total core damage frequency associated with ATWS for Big Rock Point is estimated at 3.8E-6/year, or less than 7% of the total for internally initiated events.

Several aspects of the Big Rock Point Plant design and operation result in a relatively low risk contribution due to ATWS initiators.

- A low annual reactor trip frequency: The transient initiators listed above total less than 1.4 trips/year. The period over which this initiating event data was collected includes almost 30 years of actual plant operation. Only six plant trips have occurred from any significant power level over the last 10 years, suggesting an even lower reactor trip frequency.
- Fast-acting poison injection system: The Big Rock Point Plant is equipped with a very fast-acting liquid poison injection system. The system is initiated passively by nitrogen pressurization and establishment of a siphon as opposed to the use of charging pumps. Subcriticality is achieved at hot operating conditions on the first pass of poison through the core, which occurs in slightly over a minute following actuation of the system under natural circulation conditions.
- Substantial primary coolant pressure relief capability: The Big Rock Point Plant was equipped with 100% turbine bypass capability. On the occurrence of a turbine trip, the bypass valve opens automatically to relieve all steam to the main condenser. A complete failure to scram can occur without a

demand on any primary system SRVs. Even if the main condenser were to be unavailable, the six steam drum safety relief valves have 200% relief capacity. Primary system pressure control is not a concern for Big Rock Point following an ATWS (even without recirc pump trip).

- Limited high pressure injection capability: The Big Rock Point design does not include any high pressure, high capacity ECCS makeup systems such as HPCI. For this reason, only the feedwater system is capable of maintaining the reactor at power with a high steam flow rate to containment following a failure to scram. However, feedwater addition is limited to the inventory available in the hotwell, which would last only a few minutes at full power operation. For these reasons, continued injection of high pressure systems to the vessel following an ATWS would not result in overpressurization of containment.

- Emergency Operating Procedures: The Big Rock Point EOPs provide appropriate instructions for reactivity control, reactor pressure control and reactor inventory control during ATWS conditions that will result in both rapid shutdown and adequate core cooling. The EOPs also provide instructions to the operator for protecting the containment from over-pressurization should power operation continue without the ability to achieve subcriticality.

Several additional capabilities are included in the Big Rock Point Plant design that have been given limited or no credit in the quantification of accident sequences associated with ATWS.

- Full load rejection capability: With 100% turbine bypass capability, the Big Rock Point Plant is equipped with the ability to sustain a failure to scram and continue operating for an indefinite period by condensing steam in the main condenser and returning it to the reactor with feedwater. This capability has been successfully demonstrated at power levels roughly half of full power. At higher power levels, operating experience has shown rejection of water from the hotwell to the condensate storage tank occurs, thus causing feedwater to trip on low suction pressure. For high pressure transients (i.e., sequences that result in an EC auto actuation signal) and load rejection scenarios, the recirc pumps will receive an auto trip signal. During low pressure scenarios, such as an open bypass valve sequence, manual action to trip the recirculation pumps may reduce reactor power sufficiently to maintain feedwater pump operability and allow the plant to sustain the evolution. Credit for this action in preventing feedwater pump trip has not been taken in the ATWS sequence quantification, even though the operators have demonstrated the ability to accomplish this action both in the simulator and during actual plant operation.

- Large, strong containment: The containment volume is very large for the small thermal power of the reactor. The ultimate capacity of the containment is estimated to be approximately 79 psig or approximately three times design pressure. Even though Big Rock Point does not have a suppression pool, power generation into containment at 20% to 30% power would take about an hour prior to exceed the ultimate capacity of containment. Credit for operator actions to provide alternate means of rod insertion or boron injection during the period following failure of liquid poison injection until the time at which containment failure is postulated has not been assumed in the accident sequence quantification.

- Infinite core cooling supply with low pressure core spray: Core damage has been assumed to occur if the initial attempts to inject liquid poison are not successful in providing reactor shutdown. In fact, if the reactor is depressurized and core spray operation occurs, only a limited amount of fuel damage may occur, if any. Core spray operation can continue indefinitely (as its suction source is from Lake Michigan) even if pressurization of containment to its ultimate capacity occurs. Credit for long term core cooling following containment failure has not been credited in the ATWS sequence quantification.

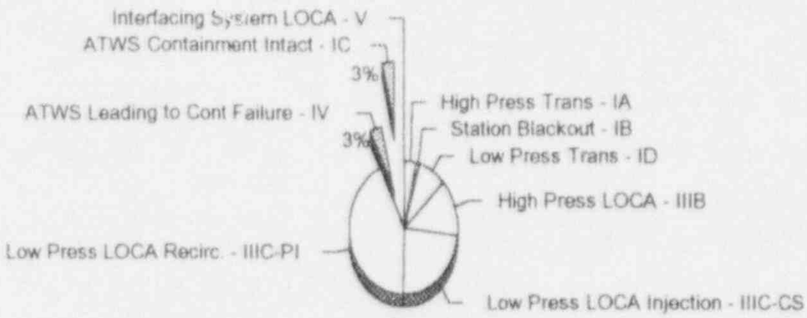
3.4.1 Accident Class IC (Failure to Scram, Without Liquid Poison Injection, Containment Intact)

This accident class begins with transients in which a failure to scram occurs and operator action to insert rods with a manual scram (moving the mode switch to shutdown or tripping the reactor protection system breakers) is unsuccessful. At this juncture, the EOPs require tripping of the recirc pumps (if this has not occurred automatically) and actuation of the liquid poison system.

In this accident class, poison is assumed to be unsuccessful in achieving reactor subcriticality. On termination of feedwater operation, reactor level falls to the RDS setpoint. On initiation of liquid poison, the operator is instructed to inhibit RDS. Without feedwater operation, the reactor level drops into the active fuel region. Subcriticality occurs due to voiding the reactor core, but core damage results with the reactor at high pressure.

Should RDS actuate, reflooding of the core with core spray would occur. Return to power at a rate equivalent to that required to boil off core spray flow would result. Gradual pressurization of containment from steam flow through the RDS trains would occur. To preserve the integrity of containment from this challenge, the EOPs instruct the operators to terminate injection to the reactor when containment pressure exceeds 10 psig and continues to rise following failure of the poison system. As is the case with inhibiting RDS, reactor power is terminated as the core is voided. Core damage is assumed to occur into an intact containment, this time with the reactor at low pressure.

Accident Class IC has an estimated frequency of occurrence of 2E-6/year, or less than 3% of the total core damage frequency due to internal events.



Accident Sequence Characteristics (Class IC)

While initiated by a transient, sequences in this accident class proceed more quickly because the reactor is at power throughout the event as opposed to decay heat levels. The timing and nature of plant response depends on the status of the main condenser, feedwater and the recirc pumps.

For events in which the main condenser is available, reactor pressure quickly rises following trip of the turbine. The bypass valve receives a signal to open at 10 psi above normal pressure (at 25 psi it is 100% open) and is assumed to respond sufficiently fast that peak reactor pressure is below 1435 psig (or less than a 100 psi pressure rise). Automatic operation of the emergency condenser or recirc pump trip does not occur at this pressure. Without recirc pump trip, a swell in the hotwell level is expected and the reject valve to the condensate storage tank would open. A trip of the feedwater pumps on low suction pressure is assumed to occur approximately 1 minute into the transient. Without feedwater, core inlet subcooling would drop and reactor power would reduce to about 50% of its original level. Reactor inventory would deplete at this rate for 2 to 3 minutes until low reactor water level is reached at 2' 9" above the core. At this juncture, RDS actuation is assumed to occur if it has not been inhibited. Initiation of liquid poison (within this time frame) permits reactor shutdown and restoration of reactor level with either feedwater or CRD.

For events in which the main condenser or continuous feedwater is not available, a high reactor pressure will occur with steam drum safety relief valve actuation approximately 8 seconds into the event. With the reactor pressure more than 200 psig above normal pressure, operation of the emergency condenser and recirc pump trip will occur automatically. Reactor power will drop to 60% of its former level with an additional 10% power being removed by the emergency condenser (for a short period of time). Feedwater operation will continue until the hotwell inventory is depleted which is expected to last about 3 minutes. At this time, feedwater pump trip occurs and reactor power level drops to 25% as a result of loss of the core inlet subcooling. Reactor power slowly coasts down as reactor level decreases. At this reduced power, and with the emergency condenser in service, an additional 8 to 10 minutes is available before reactor inventory drops to 2' 9" above the core. Initiation of liquid poison at any time in the first 12 minutes of this event would achieve reactor shutdown and avoid RDS.

The operator is instructed by the EOPs to inhibit RDS for sequences in which liquid poison injection is required. For both of these scenarios, however, RDS actuation is assumed which leads to depressurization of the reactor and shutdown of the core due to voiding. On depressurization to below 150 psig, core spray flow begins to reflood the reactor with cold water. A return to power is expected at a rate equivalent to that required to boil off the water being injected by core spray. At 300 to 500 gpm this power level is expected to be between 20% to 30% of full power.

The EOPs provide instructions to the operator to terminate injection with the containment greater than 10 psig during ATWS conditions in order to protect the containment from overpressure challenge. Securing core spray before substantial additional containment pressurization results in fuel damage but assures the integrity of containment. If injection is not terminated, gradual pressurization of containment occurs until containment reaches its ultimate capacity of 79 psig. Pressurization of containment in this manner is expected to take about an hour.

<u>Key Assumptions (Class IC)</u>

The following is a summary of assumptions which are important to the quantification of sequences in Accident Class IC.

- No automatic trip of the recirc pumps occurs if the main condenser and bypass valve are in operation. Manual recirc pump trip is not credited in time to prevent feedwater pump trip on low suction pressure once condensate rejection to the CST occurs, even though the capability to perform this action has been demonstrated.

- Poison injection is assumed to be required prior to RDS actuation. Limited fuel damage is assumed to result on return to power during reflood from core spray operation if RDS operation is successful. Core damage is not considered as is the case with other accident classes. Rather, the core remains in a configuration that would support criticality such that the potential for challenge to containment from continued power operation can be examined.

- The squib valves may self actuate on exposure to the steam environment from SRV or RDS operation. Self actuation is not assumed in the sequence quantification. While not formally a part of the EQ program, the squib valves are assumed to operate only if initiated by manual action to fire the explosive valves from the Control Room. This actuation is assumed to be required within a time frame which avoids RDS (approximately 2 to 12 minutes, depending on the accident scenario).

Contribution of Initiating Events (Class IC)

A series of potential initiating events was considered in the quantification of ATWS accident sequences. The contribution of these initiators to Class IC is as follows:

Initiating Event	Contribution to Class IV
Turbine Trip	92%
Loss of Feedwater	4%
Spurious Turbine Bypass Valve	4%
Loss of Instrument Air	<1%
Loss of Off-site Power	<1%
Loss of Main Condenser	<1%
MSIV Closure	<1%

- The initiators which contribute most significantly to this accident class are those with the shortest time-frames to RDS actuation. These initiators result in continued power generation at the original power level as a result of continued recirc pump operation as well as loss of feedwater due to rejection of the hotwell inventory to the CST. Turbine trip is the most significant of these initiators simply because of its frequency of occurrence (assumed at one event per year).

Contribution of Hardware (Class IC)

The risk associated with ATWS is largely driven by the initiating event and operator response to the accident sequence conditions. Hardware failures contribute very little to ATWS dominant accident sequences. The single most important system in terminating an ATWS event is the liquid poison system which, at less than 1E-3/demand, contributes very little to the core damage frequency of this accident class.

Contribution of Operator Action (Class IC)

The most important operator actions in this accident class are those specified in the EOPs to trip the recirc pumps and initiate poison. These events appear in nearly all of the failures associated with ATWS sequences.

Operator Action	Contribution to Class IV
Inhibit RDS	100%
Inject Poison	99.8%
Trip Recirc Pumps	99%

Manual initiation of poison occurs in 99.8% of ATWS accident sequences. The remainder of sequences include hardware failures associated with the poison system. Manual tripping of the recirc pumps shows up in slightly fewer sequences than injecting poison as it is automatic for some scenarios. Inhibiting RDS is assumed to be conditional on failure of LPS and therefore appears in all sequence cutsets.

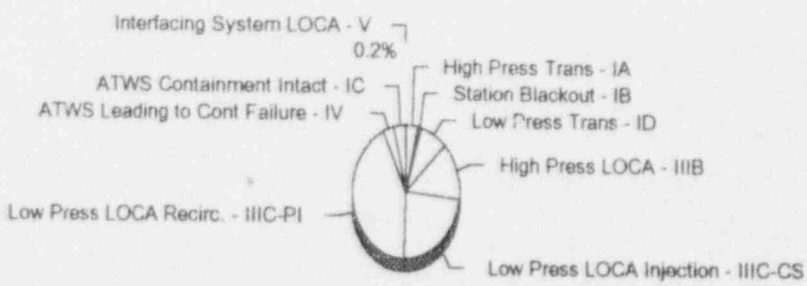
3.4.2 Accident Class IV (Failure to Scram, Without Liquid Poison Injection, Core Damage Leading to Containment Failure)

This accident class is nearly identical to Accident Class IC. It begins with a failure to scram and proceeds to the point of RDS actuation with the potential for limited core damage if poison is not successful. While EOPs require the RDS to be inhibited on initiation of poison, in Accident Class IV, RDS actuation and subsequent core spray operation is assumed. Injection of water by use of core spray results in reflooding of the reactor vessel and power generation (~20%) by the reactor core roughly equal to the rate of core spray flow to the core. Steam being generated in the reactor is directed through the four RDS trains to containment. Gradual pressurization of containment is assumed to occur until the ultimate capacity of 79 psig is achieved.

The contribution of initiating events and hardware to this accident class are the same as for Class IC. Operator actions are also the same, except that the operator action to terminate the event and preserve containment integrity by inhibiting RDS or securing core spray is not assumed. This action is assumed to be highly conditional on failure to initiate poison (0.5) and accounts for the even distribution of ATWS core damage sequences between Classes IC and IV.

3.5 Accident Class V - Associated with Containment Bypass

Accident Class V sequences are characterized by events in which the containment is directly bypassed. The consequences of this scenario are substantially higher than other postulated severe accidents as the containment is bypassed and an un-isolatable radionuclide release outside of containment occurs. This accident class contributes 0.2% to the total core damage frequency and is dominated by spurious bypass valve opening. Steam line breaks outside of containment and interfacing system LOCAs (ISLOCAs) are discussed in Sections 3.5.2 and 3.5.3, respectively.



Contribution of Initiating Events (Class V)

The breakdown of the initiating frequency for Class V sequences follows:

Initiator	Contribution to Class V
Spurious Bypass Valve Opening	81%
Core Spray Line ISLOCA	7.1%
Bypass/Fuel Pit Drain Line ISLOCA	4.7%
Medium Steam Line Break Outside	1.1%

Initiator	Contribution to Class V
Large Steam Line Break Outside	.16%

3.5.1 Accident Class V (Spurious Bypass Valve Operation)

Accidents initiated by spurious bypass valve operation constitute 81% of Accident Class V sequences. Typically, at other plants, interfacing system LOCAs account for the major contribution to Accident Class V. The contribution to Accident Class V from spurious bypass valve operation (8.9E-8) is principally due to the operator action required to terminate the blowdown prior to RDS actuation, combined with the relatively large initiator frequency of 4.9E-2/year. Categorizing this accident sequence as an Accident Class V is conservative as an automatic main steam isolation signal is required to be generated during the transient in order to be considered a Class V sequence. In this case, given manual failure to close the MSIV, an automatic MSIV closure signal would be generated at the same time that the RDS receives a signal to depressurize the primary system. Therefore, no credit was taken for automatic closure, as the blowdown would still have occurred through the RDS valves resulting in an assumed noble gas release to the turbine hall. If the sequence quantification had included the manual action and automatic closure, the resultant calculated unavailability would have been reduced by an order of magnitude. Therefore, the spurious bypass valve contribution to Class V conservatively over-estimates the importance of Class V sequences.

Key Assumptions (Spurious Bypass Valve Opening - Class V)

The following summarizes key assumptions which influence Class V accident sequence quantification:

- The bypass valve instantaneously opens to 100% of the full open position. This results in a hotwell level swell causing the condensate reject valve to open, and a subsequent drop in feedwater pump suction pressure, thus causing a feedwater pump trip.
- Condenser vacuum remains above the automatic closure setpoint of the turbine bypass, bypass isolation and bypass warming valves, therefore bypass line isolation is not automatic.
- Isolation of the main steam line requires manual closure of the main steam isolation valve (MO-7050).

Contribution of Hardware Failures (Spurious Bypass Valve Opening - Class V)

Components	Contribution to Class V
Common cause failure of the core spray valves to open	74%
Hotwell makeup valve fails to energize	30%
Core spray valves fail to open	13%
Fire pumps fail	5%
MSIV fails to close	4%

- Low pressure injection failures, including hotwell makeup from the condensate storage tank and core spray, are dominant component contributors.
- The automatic closure of the MSIV is not important to Class V sequences. This is because automatic closure of the MSIV is not credited as the closure signal and is the same signal that actuates the RDS. If automatic MSIV closure was assumed, the MSIV, as a component, would be as important as the contribution of the operator action to close the MSIV (listed below).

Contribution of Human Actions (Spurious Bypass Valve Opening - Class V)

Human Actions	Contribution to Class V
Operator fails to isolate the main steam line	99%
Operator leaves hand-switch HS-7039 for the hotwell makeup valve in the wrong position	65%
Operator fails to restart the feed-water pump following pump trip	9.4%

- In this sequence, feedwater is assumed to initially fail with a probability of 1.0. For injection to the reactor pressure vessel, following RDS, it is assumed the operator is required to restart the feedwater pump. Makeup to the hotwell from the CST is assumed required to permit feedwater addition.

- Because of the large human error failure rates associated with the first two manual actions contained in the table, these HEPs dominate the sequence frequency.

3.5.2 Accident Class V (Un-Isolated Steam Line Breaks Outside of Containment)

Accidents initiated by steam line pipe rupture (outside of containment), constitute 1.3% (1.4E-9/year) of the frequency for Accident Class V sequences. Automatic MSIV closure failure must occur coincident with the pipe break outside of containment in order to result in containment bypassed conditions.

Key Assumptions (Un-Isolated Steam Line Breaks Outside - Class V)

The following summarizes key assumptions which influence un-isolated steam line breaks outside sequence quantification:

- Successful automatic MSIV operation will not preclude RDS operation.
- The operators would be unable to close the MSIV in time to avert RDS operation for medium and large breaks outside of containment.
- MSIV auto failure and feedwater success will not result in an automatic RDS signal. Furthermore, the hotwell inventory and makeup from the CST are considered sufficient for decay heat removal.
- Fire water makeup to the hotwell is assumed to be required for sequences in which RDS actuates but automatic MSIV failure occurs. For sequences in which MSIV isolation is successful, terminating steam flow to the turbine building, makeup from the CST is sufficient to maintain feedwater makeup to the reactor.
- Only medium and large breaks contribute to this class. The un-isolated small steam line break outside is not a significant contributor to Class V, as manual closure of the MSIV can isolate the break.

Contribution of Hardware Failures (Class V)

Components	Contribution to Class V
MSIV fails to close	93%
RDS failures	19%
Condensate makeup failures	9.7%

Components	Contribution to Class V
Electric fire pump fails to run	5%
Core Spray valves	3.5%

- The steam line break outside containment sequences contributing to Accident Class V, are dominated by failure of the MSIV to close and terminate the blowdown. In these sequences, MSIV closure is automatic.
- Given failure to isolate, the reactor depressurization system is required for low pressure injection success.
- Given successful RDS, low pressure injection from the condensate system or core spray (fire pumps required) is important.

Contribution of Operator Actions (Class V)

Human Actions	Contribution to Class V
Operator fails to actuate both loops of the emergency condenser	75%
Operator leaves handswitch HS-7039 for the hotwell makeup valve in the wrong position	53%

- For the medium steam line break outside containment, manual emergency condenser actuation is required following RDS failure for low pressure injection success. Although this event does not appear in the large break failure summary, it is important as the medium break initiator is an order of magnitude larger than the large break initiator. In addition, the conditional probability of the manual action is 1.0 which substantially promotes the value of the importance measure.
- For medium and large steam line breaks outside of containment, MSIV closure is automatic, so the manual action to isolate the main steam line is not a contributor.
- The pre-initiator operator action of leaving the condensate fill valve handswitch in the wrong position dominates the human error contribution for condensate/feedwater as the condensate makeup is assumed necessary for feedwater operation. The manual action contribution is large compared to the

mechanical/electrical contributor due to the difference in failure probabilities.

3.5.3 Accident Class V (Initiated by Interfacing System LOCAs)

All core damage sequences which result from an interfacing system LOCA (ISLOCA) are grouped in Accident Class V. The initiating event frequencies in this Accident Class are among the lowest in the PRA, resulting in a small contribution to the core damage frequency. This sequence is characterized by the loss of isolation between the high pressure primary system and low pressure systems outside containment. This condition can potentially lead to the loss of reactor coolant while simultaneously disabling emergency core cooling systems, thus creating an unattenuated radionuclide release pathway outside containment. The potential consequences of this scenario are substantially higher than other postulated severe accidents because the containment is bypassed.

This section describes important features of the Big Rock Point design which result in the relatively low risk contribution associated with ISLOCA initiated events. Key assumptions regarding system success criteria (as related to plant response to ISLOCA initiators) are discussed.

Accidents initiated by ISLOCAs constitute approximately 12% (3.8E-8/year) of Accident Class V sequences. This accident class is characterized by ISLOCAs in which the low pressure piping connected to the high pressure primary system piping has ruptured outside of containment. Since it is assumed that the break cannot be isolated, reactor depressurization is required for low pressure injection if feedwater is unavailable.

In Accident Class V, the containment is bypassed, thus producing a potential direct path for radionuclide release outside of containment. No credit is taken for isolating the break as the break location is considered un-isolatable.

<u>Accident Sequence Characteristics</u>

Sequences in this accident class begin with an initiating event that causes the low pressure piping (which is connected to the primary system) to rupture outside of containment. It is assumed that the reactor scrams automatically on low steam drum level. Depressurization by RDS or the emergency condenser is required for all accident sequences for low pressure injection. Low pressure injection can be provided by the feedwater/condensate system or the core spray system. In some accident sequences, it is assumed that the ISLOCA has occurred in the core spray line; therefore, low pressure injection from this system is unavailable.

The timing of this event is driven by the size of the break. Two different segments of piping have met the criteria for ISLOCA-susceptible piping. For the larger break (core spray line), the dynamics are fairly rapid. The operator has time to verify and, if necessary, establish the expected system response. Assuming CST availability and the large tail end break area of 0.2 ft^2, the reactor water level would quickly collapse to low reactor water level setpoint within approximately 6 seconds after the break. The primary system would have depressurized to below the shutoff head of the fire pumps in less than 3 minutes after the break; however, the operator would still have approximately 51 minutes to establish primary system makeup in order to prevent peak clad temperatures from reaching 2500°F.

For the smaller break (bypass/fuel pit drain line), the transient occurs at a slower rate. For this size break, the loss of primary system mass out the break would cause an immediate drop in steam drum level. Although the feedwater control system would attempt to restore the water level by fully opening the feed reg valve, the additional feed flow supplied to the steam drum would be insufficient to avoid a scram on low level (i.e., centerline -8 inches). Following successful automatic control rod insertion and containment isolation, the desired plant response consists of automatic RDS actuation followed by core spray reflood.

Key Assumptions (Class V)

The following is a summary of assumptions which influence the quantification of interfacing system LOCA sequences:

- The pathway must be connected to the Primary Coolant System (PCS).
- The interfacing system must have a design pressure lower than that of the primary coolant system.
- The overpressurization event must be possible during power operation.
- The lower pressure interfacing system must extend outside containment.
- On actuation of RDS, EOPs require maintaining reactor level near normal by use of low pressure systems such as condensate or core spray.
- If an interfacing system has low pressure piping inside as well as outside of containment, the outside segment is assumed to be the location for the failure.
- If an interfacing system LOCA has low pressure piping segments inside and outside containment, the outside segment is assumed to fail.

Contribution of Hardware (Class V)

Components	Contribution to Class V
Common cause failure of core spray valves	8.4%
RDS failures	6.9%
Electric fire pump fails to run	4.3%
Condensate fill valve fails to open	4.3%

- Common cause failure of the core spray valves cause low pressure core spray system failures, which are important for core cooling.
- The RDS is required for depressurization for both ISLOCAs initiators. Failure results in a loss of the low pressure injection systems.
- The electric fire pump failing to run causes low pressure core spray failures, which are important for core cooling.
- The condensate fill valve failing to open causes low pressure condensate system failures, which are important for core cooling.

Contribution of Operator Actions (Class V)

The systems most important to this accident class are the reactor depressurization, core spray and condensate systems. Operation of these systems provide core cooling and prevent core damage. Because of the rapid nature of the events, and the automatic actuation of RDS and core spray, the operator actions (listed below) are not important contributors to interfacing system LOCA sequences.

Human Actions	Contribution to Class V
Operator fails to actuate both loops of the emergency condenser.	59%
Operator leaves handswitch HS-7039 for the hotwell makeup valve in the wrong position	28%

- A core spray line ISLOCA requires emergency condenser operability for additional depressurization for low pressure injection. This is a significant contributor, as no credit has been assumed for this event.

- Failure of the condensate system is important as the core spray ISLOCA does not take credit for low pressure injection from the core spray line due to the location of the break. Therefore, the condensate system is the only source of low pressure makeup.

TABLE 3.1-1

BIG ROCK POINT PRA
ACCIDENT CLASSES

Class	Definition	Frequency (yr^{-1})
IA	Transient initiated event with core damage at high reactor pressure	2.0E-6
IB	Station Blackout	5.1E-7
IC	ATWS events in which core damage results from failure to make up to reactor	1.9E-6
ID	Transient initiated events with core damage at low reactor pressure	4.5E-6
IIIB	LOCA initiated events with core damage at high reactor pressure	7.5E-6
IIIC-CS	LOCA initiated events with core damage at low reactor pressure due to core spray injection failure	1.3E-5
IIIC-PI	LOCA initiated events with core damage at low reactor pressure due to post incident recirculation failure	2.3E-5
IV	ATWS in which continued makeup to reactor results in containment over pressure	1.9E-6
V	Containment bypass	1.8E-8
Total		5.4E-5

ATTACHMENT 2

CONSUMERS POWER COMPANY
BIG ROCK POINT PLANT
DOCKET 50-155

RESPONSE TO GENERIC LETTER 88-20
REISSUE OF SECTION 13.0 FROM MAY 5, 1994 RESPONSE

May 27, 1994

125 Pages

13.0 UNCERTAINTIES

This section describes the key assumptions and uncertainties and their expected impact on the results of this study.

Rather than perform a detailed statistical uncertainty analysis, a qualitative approach was used. Level I sensitivity calculations were performed on the overall core damage frequency for specific events or assumptions. These sensitivities were chosen based upon the results of the calculated importance measures. The Level II sensitivity calculations focused on key phenomenological issues raised by the NRC. The Level II discussion is presented in Section 13.7. The following discussion centers on the Level I analysis.

13.1 Level I Sensitivity Evaluation

Previous risk and reliability related studies performed for Big Rock Point have identified a number plant modifications to the original plant design that have greatly improved the calculated risk of operating the plant. These modifications include: the addition of a second off-site power line to the substation, a redundant core spray line, an automatic reactor depressurization system, post incident valve position locks, automatic recirc pump trip during load rejections, the Appendix R alternate shutdown panel, SV-4947 for remote actuation of fire water to the emergency condenser and the addition of a portable pump for emergency condenser makeup. Implementation of these modifications has spanned the entire life of the plant. They have also addressed the risk significant issues at the plant.

Given the age of Big Rock Point, the remaining time left in the operating license and the current acceptable risk of operating the plant, rather than investing in risk reduction, investments should be directed to risk maintenance. Maintaining the risk involves keeping the existing plant components and systems important to risk at their current levels of performance.

To determine the systems and components that are the key contributors to risk, the Fussell-Vesely (risk reduction) and the Birnbaum (risk achievement) importance measures are used.

13.1.1 Fussell-Vesely Importance Measures

The Fussell-Vesely (FV) importance is an indication of the current contribution one event (or system) has on the overall core damage frequency (CDF) calculation, and is used as a measure of its risk reduction potential. The FV value is determined by dividing the change in the CDF (determined with an event equal to zero and the base CDF) by the base CDF (setting the event equal to zero is equivalent to assuming it never fails). The Fussell-Vesely value is basically the percent contribution one event (or system) has to the CDF. In equation form this is:

$$FV = \frac{(CDF_{BASE} - CDF_{I=0})}{CDF_{BASE}}$$

The CDF value, determined with an event equal to zero, represents the contribution of all cutsets that do not contain the event in question, thus the difference between this value and the base case CDF is the contribution to the CDF from just one event.

The FV values were generated for all the basic events that appear in the sequence cutsets from the event tree evaluation. Table 13.1-1 shows the events that have a FV value greater than 5E-2 (5% of CDF_{BASE}).

As an example, the Fussell Vesely value is calculated below for the electric fire pump failing to run. This is event FP-PM-P6-FTR, which has a FV value of 8.7E-2. This indicates that the electric fire pump failing to run contributes about 9% of the overall CDF.

$$CDF_{BASE} = 5.35E-5 \qquad CDF_{P6=0} = 4.88E-5$$

$$FV = \frac{(5.35E-5)-(4.88E-5)}{5.35E-5} = 8.7E-2$$

13.1.2 Birnbaum

The Birnbaum importance value is considered as a measure of an event's (or system's) risk achievement potential. The Birnbaum value is the slope of the line created by plotting the core damage frequency obtained for different probabilities of a single event. The steeper the slope of the CDF for a given basic event, the greater the potential for increasing or reducing risk were the reliability of the basic event to change. By evaluating the slope with the event probability set to 1 and 0, the denominator is equal to 1.0 and the numerator is the difference in core damage frequencies. The Birnbaum importance is then determined by simply calculating the difference between the CDF evaluated with the event set to one and then with the event set to zero. In equation form this is:

$$B = CDF_{I=1} - CDF_{I=0}$$

The Birnbaum values were generated for all the events appearing in the sequence cutsets. Table 13.1-2 shows the events with a Birnbaum greater than 5E-4 per year (one order of magnitude greater than CDF_{BASE}).

As a general guideline, if the CDF changes appreciably from the base calculation when a component (or system) is always assumed to fail (high Birnbaum), it is assumed that the current level of component (system) performance should be maintained and not allowed to significantly degrade. If the change in core damage frequency is minimal (low Birnbaum), the opportunity may exist to relax the performance levels without affecting the overall core damage frequency calculations.

13.1.3 System Importance Measures

The importance measures were also determined for the major plant systems, support systems and for several important sub-systems. Examples of sub-systems are the traveling screens or fire water to the emergency condenser. The events that were grouped into the various systems, support systems and sub-systems are shown in Tables 13.1-3 and 13.1-4. Also included in the tables are the importance measures of each group. The importance measures are plotted in Figures 13.1-1 and 13.1-2.

The Birnbaum importance measures for each system, sub-system or group are shown in Figure 13.1-1. The plot shows five groups of components that have Birnbaum values equal to or greater than 1.0. These are the systems (components) that have the greatest impact on the CDF should they degrade from current performance levels. The five groups are:

- FP Fire Protection System (1.7/yr)
- CS Core Spray Valves (1.7E-1/yr)
- Intake Traveling screens (1.1/yr)
- PI Post Incident Recirculation (1.4E-1/yr), and
- RD Reactor Depressurization System (2.2E-1/yr).

The Fussell-Vesely importance measures are shown on Figure 13.1-2. This plot shows the five systems that contribute significantly to the CDF. The five systems are:

- CS Core Spray Valves (4E-1)
- RPS Total ATWS contribution (3.8E-1)
- PI Post Incident Recirculation (1.7E-1), and
- RD Reactor Depressurization System (1.4E-1).
- FP Fire Protection System (1.2E-1)

It is interesting to note that four of the top five systems are shared between the two measures. This indicates that these systems are important to current risk, and that they have a big influence on risk if their performance degrades from current levels.

13.1.3.1 Four-Quadrant Plot

To rank the overall importance of a system or group of events, the Fussell-Vesely importance measure was plotted versus the Birnbaum importance measure for each system or group. This plot is shown in Figure 13.1-3. When lines are drawn through points representing a Birnbaum of 5E-4 (the NRC Regulator Review Group recommendation for high system performance) and a Fussell-Vesely of 5% (BWROG Risk Based Regulation guideline of 10 times the CDF for system importance), the systems become divided into three distinct groupings. Combining risk achievement and risk reduction measures in this manner assists in generating insights to areas where improving or maintaining system and equipment reliability would have the greatest effect.

The systems appearing in the upper right quadrant are those systems most important to the current CDF. These systems have a high Fussell-Vesely (the greater the FV the greater the contribution to CDF) and a large Birnbaum (a large Birnbaum implies that a reduction in current performance could have a big impact on CDF). Modifications aimed at reducing the importance of these systems (i.e. providing means to cope with the failure of these systems) would have the greatest benefit. Maintaining or improving system performance is also beneficial.

The systems in the upper left quadrant are those that have a moderate or small impact on the current CDF calculation, but performance degradation can have a large impact on the CDF value. Maintaining current system performance is more appropriate than providing modifications to reduce their contribution to CDF.

The systems in the lower left quadrant also have a moderate or small impact on the current CDF calculation; however, some degradation in their performance (even run to failure) can be tolerated with little increase in risk.

13.2 Level I Importance Measures

The following discussion centers on those systems or components that are influential in the current plant risk calculations.

13.2.1 Systems Most Influential to CDF

The four of the five systems that appear in the upper right quadrant of Figure 13.1-3 drive the results of the PRA. The end state distribution shows that about 80% of the CDF frequency is due to Class IIIB and IIIC sequences. These are sequences resulting from a LOCA in the primary system. The important mitigative systems for the various sized LOCA initiators are core spray and reactor depressurization for the initial response to the transient, and the post incident system for long term core cooling. The fire protection system is the common support system to each of these systems. The systems appearing in the upper right quadrant are:

- FP (Fire Protection)
- RD (Reactor Depressurization)
- PI (Post Incident)
- CS (Core Spray)
- RPS (ATWS)

The RPS grouping is the entire contribution to the CDF calculation from the ATWS event trees. Individually, no single ATWS sequence influences the result; together the ATWS sequences contribute about 3%.

The events with the greatest contribution for the remaining four systems are discussed below. Included are failures resulting from mechanical components and pre-event operator actions. The post event operator actions in response to an initiating event are included separately.

<u>Fire Protection System</u>

The fire protection system at BRP, in addition to providing fire suppression, is used to supply the core spray and the enclosure spray systems. The fire system is also capable of supplying water to either the emergency condenser, as a backup to demineralized makeup failure, and to the condenser hotwell for feedwater and/or condensate pump operability (in the event the normal condensate transfer from the storage tank is inadequate or unavailable). Fire water supply to the hotwell is only credited during LOCA sequences in which the normal CST transfer is insufficient to maintain feedwater and/or condensate pump operation. The events that have a high Fussell-Vesely and high Birnbaum that are included in the fire protection system are:

FP-PM-P6-FTR — This event is the electric fire pump failing to run.

If the electric fire pump is assumed to fail to run with a probability of 1.0, the CDF would be 6.1E-4/yr.

FP-PM-P7-FTR — This event is the diesel fire pump failing to run.

If the diesel fire pump is assumed to fail to run with a probability of 1.0, the CDF would be 3.9E-4/yr.

FP-PM-P7-FTS — This event is the diesel fire pump failing to start.

If the diesel fire pump is assumed to never start, the CDF would be 3.8E-04. Based upon this change in CDF, the current level of performance should be maintained.

FP-PM-P6-FTS — This event is the electric fire pump failing to start.

If the electric fire pump is assumed to never start, the CDF would be 5.8-04/yr.

FP-RV-5062-FTC — This event is the diesel fire discharge relief valve failing to reclose after pump start. This is included as a flow diversion from the diesel fire pump.

If this relief valve is assumed to always fail open after pump start, the CDF would be 3.8E-4/yr.

FP-KV-FSOS-FTE — This event is the fuel transfer solenoid valve for the diesel fire pump, which is a contributor to pump failure to run.

FP-KV-PNS1-FTE — This event is the diesel fire pump pinion solenoid which is a contributor to pump failure to run.

The change in CDF of these two relays, should they individually be assumed failed, is similar to that reported above for the diesel fire pump failing to run.

The fire system has the potential to be used as the ultimate source of decay heat removal for the plant for both LOCA and transient conditions. The pumps are capable of supplying core cooling following a break in the primary system (core spray or condensate) or without a break via makeup to the emergency condenser.

Because of the importance of these components, effort has been placed on providing diversity and minimizing the potential for failure of the system. The pumps though similar in design, differ in the main contributor to pump failure--the driver (one is motor driven and the other is diesel driven). The diesel fire pump, as well as the dc powered makeup systems (core spray and emergency condenser), will function to remove decay heat without the requirement for off-site or on-site ac power.

The fire pumps are required (by tech specs) to be operable when the plant is at power. Should either pump become inoperable, the plant staff must restore the pump to an operable status within 24 hours or commence an orderly shutdown. The pumps are subjected to weekly surveillance testing that ensures their operability. The electric fire pump is the required load for the diesel generator load testing and thus experiences a higher test frequency.

The events contributing to the emergency condenser and the hotwell makeup requirements of the fire protection system were not included in the system importance calculations; rather, they were treated as separate sub-systems.

The events representing fire water to the emergency condenser (identifier FH in Figure 13.1-3) have a combined Fussell-Vesely of 2.6E-3 and a Birnbaum of 1.3E-3/yr. The fire system components which support the emergency condenser operation are somewhat less important than the fire pumps, as the emergency condenser plays only a limited role during LOCA conditions. Based on the Birnbaum measure, however, the emergency condenser plots a significant role in assuring the CDF for transient initiated events is low.

The events included for fire water to the emergency condenser include the makeup valve (SV-4947) related circuitry components and the portable pump. The use of the portable pump was only credited in the internal flood and the station blackout event trees. Should the portable pump be credited in all the sequences with emergency condenser makeup failure, the importance of both fire pumps would be reduced.

The fire water to the emergency condenser is test operated during the performance of a monthly surveillance test. The portable pump is tested on a quarterly basis.

Those events representing fire water to the hotwell (identifier FM in Figure 13.1-3) have a combined Fussell-Vesely of 9.6E-3 and a Birnbaum value of 8E-5/yr. The two motor operated valves, MO-7073 and MO-7074, are the only events included in this sub-system. The Birnbaum measure suggests that the potential for increasing the CDF, due to the unavailability of these valves is smaller than for the emergency condenser components. This indicates that fire water makeup to the hotwell has a significant effect only for a limited portion of the pipe break spectrum and reflects that it is not credited at all during transients.

These two motor-operated valves are test operated during the performance of a quarterly surveillance test.

Reactor Depressurization System

The RDS is credited principally for automatically reducing reactor pressure to allow the low pressure makeup systems to function. Other uses credited in the PRA include manual actuation to either control PCS pressure or to avoid PCS overpressure. The importance of the mechanical components is discussed here. The events that contribute to the overall RDS importance measures are the modules representing the RDS valves failing to function. Each module represents one valve from one train of RDS.

RDS-I-10	Events included in this module are related to the failure of SV-4894, RDS train A depressurization valve.
RDS-I-12	Events included in this module are related to the failure of SV-4895, RDS train B depressurization valve.
RDS-I-16	Events included in this module are related to the failure of SV-4897, RDS train D depressurization valve.
RDS-I-14	Events included in this module are related to the failure of SV-4896, RDS train C depressurization valve.
RDS-I-15	Events included in this module are related to the failure of CV-4183, RDS train D isolation valve.
RDS-I-9	Events included in this module are related to the failure of CV-4180, RDS train A isolation valve.
RDS-I-11	Events included in this module are related to the failure of CV-4181, RDS train B isolation valve.
RDS-I-13	Events included in this module are related to the failure of CV-4182, RDS train C isolation valve.

These trains have a Birnbaum value reflecting that they are credited during a large fraction of the break spectrum (small and very small LOCAs contribute more than 65% of the potential for a LOCA). Thi ndicates that the current level of system performance should be maintained.

Based on the high Birnbaum value calculated for the RDS system, the change in CDF (should RDS be failed with a probability of 1.0) could be significant.

That the Fussell-Vesely value is also high is a result of the success criteria assumed in the PRA. The success criteria for RDS require that three of the four valve trains successfully open. This is an overly restrictive assumption of the blowdown capacity actually required. The three out of four blowdown path assumption was used in all but the small above and below core LOCAs and the below core interfacing system LOCA event trees. A two of four success criteria was used for these three initiators. his success criteria is similar to that used in the licensing design basis for the ECCS. A re-analysis of the success criteria for RDS is likely to show that a single train is effective for blowdown. In using the three of four success criteria for RDS rather than a more realistic two of four (or even one of four) the importance of the system is overstated.

The RDS valves and actuation circuitry is subjected to periodic surveillance testing to ensure proper operation. The RDS actuation channels are tested on a monthly basis and the four isolation valves are tested on a quarterly basis. In addition the RDS, UPS batteries are monitored on a weekly basis. Additional testing to the RDS is performed while the plant is shutdown.

Post Incident System

The post incident system is required for long term core cooling subsequent to a break in the primary system. For this study, based upon the plant emergency procedures and the 24 hour mission time criteria, the post incident system appears in only those transients that result in an inability to restore steam drum water level during post event recovery. These initiators are the above and below core LOCAs. For steam line breaks, RDS actuation (without an initial PCS break) and SRV actuation, steam drum water level would be recovered. The amount of primary system fluid steamed to containment plus enclosure spray flow will not raise containment water levels to the recirculation setpoint of 587 feet within the 24 hour mission time. The events that contribute to the post incident system importance are:

PI-PM-P2-CCFTS	This event models the common cause failure of the post incident pumps to start.

If the post incident pumps are assumed to fail, the CDF increases to 6.6E-3/yr.

PI-CV-PI300-FTO	This event is the failure to open of check valve VPI-300. This valve is located within containment in the common post incident pump discharge line.

The change in CDF, if this check valve is assumed failed, is similar to the common cause pump event discussed above.

PI-PM-P2-CCFTR	This event models the common cause failure to run of the post incident pumps.

The change in CDF, if the post incident pumps fail to run, is similar to the common cause pump event discussed above.

PI-PM-P2A-FTS and PI-PM-P2B-FTS	These events are the post incident pumps failing to start .

The post incident system pumps are started during the performance of surveillance test TSD-07. The test requires the re-alignment of several valves that have position locks to assure their proper configuration during plant operations. The pumps are flow tested during this surveillance and are operated for approximately 15 minutes each. The Big Rock Point Plant design is such that the failure of the post incident system would not result in core damage until very late in the event, if at all. The design of the system is such that the pumps used for recirculation (post incident) are independent of the pumps used for injection (fire protection or condensate-feedwater). If post incident recirculation could not be established, the core spray system would be maintained in an injection mode. If recirculation were initiated and subsequently failed, a return to injection mode would occur.

Additional water flow to containment can take place until the containment is filled to centerline (more than 9 feet above the top of fuel), at which time injection is terminated in order to prevent over-stressing the containment shell. Addition of this much water to containment takes several days at the core spray injection flow rate, providing significant time for repair and recovery of post incident recirculation (which has not been credited in the PRA).

Core Spray System

The core spray system consists of two independent trains of motor operated valves that are required to open to provide low pressure core cooling water flow following an accident in which fluid is lost from the primary system. Each train includes two motor operated valves separated by a check valve. The core spray valves are assumed to be required for all LOCAs, if not for the injection mode than for the recirculation. Its importance is included in many of the accident classes. The events that contribute to the core spray system importance are:

CS-MV-AC-CCFTO CS-MV-AD-CCFTO CS-MV-BC-CCFTO CS-MV-BD-CCFTO	These events are the common cause failure to open of one motor operated valve in each train. These events each have a Birnbaum value of 7.5E-3/yr.
CS-MV-7071-FTO	This event is the failure of MO-7071 to open. This valve is in the redundant core spray line. This event has a Birnbaum value of 1.1E-4/yr.
CS-MV-7051-FTO	This event is the failure of MO-7051 to open. This valve is in the primary core spray line. This event has a Birnbaum value of 2.4E-4/yr.
CS-MV-7061-FTO	This event is the failure of MO-7061 to open. This valve is in the primary core spray line. This event has a Birnbaum value of 2.4E-4/yr.
CS-MV-7070-FTO	This event is the failure of MO-7070 to open. This valve is in the redundant core spray line. This event has a Birnbaum value of 1.1E-4/yr.
CS-MV-ABCD-CCFTO	This event is the common cause failure to open of all four core spray valves. This event has a Birnbaum value of 7.4E-3/yr.

The core spray motor operated valves are timed during the performance of a monthly surveillance test. As part of this test, the pressure and level switches used to initiate valve opening are manually actuated to send open signals to the valves. The core spray valves are also subject to Generic Letter 89-10, Motor Operated Valve Performance Testing.

Maintenance Unavailability Importance

Contributing between 1 and 2% of core damage are the fire pump maintenance unavailability events. The plant is required (by tech specs) to have both fire pumps in service at all times. Failure to do so requires a plant shutdown within 24 hours.

FP-PM-P6-OOS	This event is the electric fire pump out for maintenance unavailability. This event has a Birnbaum value of 4.71E-4/yr.
FP-PM-P7-OOS	This is the electric fire pump out for maintenance unavailability. This event has a Birnbaum value of 2.69E-4/yr.

13.2.2 Upper Left

The systems that fall into this quadrant do not currently contribute to risk but could if system performance were to degrade significantly. The two systems that have the highest Birnbaum values are:

- Intake Traveling Screens (Birnbaum = 1.1/yr)
- SP Station Power (Birnbaum = 2.6E-2/yr)

Traveling Water Screens

The traveling water screens are located in the screenhouse structure. The condenser circulating water pumps, the service water pumps and both fire protection pumps draw untreated water directly from Lake Michigan through these screens. These screens function to remove the medium to small size material that may be drawn into the inlet bay due to pump operation. The inlet crib (located in Lake Michigan) and the trash racks serve to prohibit the larger sized material from entering the traveling screens. The low ranking of this sub-system based on its Fussell-Vesely of 1.7E-4, is an indication of the high quality water available to the plant for cooling. Plant operating history shows only several instances in which high differential pressures on the traveling screens (indication of clogging screens) have caused increased operator surveillance to remove the collected material from the screens. The automatic screen wash cycle has operated to keep the screens clear of material. The high Birnbaum for this sub-system (1.1/yr) is an indication of how pervasive the traveling screens are. The main condenser (circulating water pumps) and the emergency condenser (fire water makeup and service water cooling of the air compressors) require water from the traveling screens. For LOCAs the core spray system (fire protection system) is affected by the screens.

Station Power

The station power system consists of those components required to provide electrical power to the various equipment operated in response to a transient or LOCA. Included are such items as the different panels, buses, motor control centers and circuit breakers used in the station power distribution network. The dc and emergency power components are included as separate systems in Figure 13.1-3. The basic events grouped into station power have a Fussell-Vesely of 2.2E-2 and a Birnbaum of 2.6E-2. The fact that the plant has the capability to provide adequate decay heat removal (in the absence of ac power) is a contributing factor to the relatively low Fussell-Vesely value. With the emergency condenser and the diesel fire pump, the plant can effectively maintain decay heat removal and inventory control for all transients. For LOCAs, the diesel fire pump and the dc operated core spray valves provide inventory control and decay heat removal. The Birnbaum value indicates the importance of maintaining back-up systems for the diesel fire pump. Listed below are three of the basic events that contribute to the high Birnbaum:

SP-BS-PNL1Y-SCLG SP-BS-PNL3Y-SCLG	These two events have an effect on a majority of the plant instrument and control functions. Condensate system valve controls and manual pressure control are two of the functions served by these two power panels. Without power, the automatic transfer of water from the CST to the hotwell cannot occur, and the manual RDS function from the Control Room is disabled. Each of these basic events has a Birnbaum value of 7.5E-3/yr
SPS-I-6	This module of events represents the failure of the 2400 volt bus. This bus is required to provide power to the feedwater and condensate pumps. The unavailability of this bus places reliance upon the diesel generators for on-site ac power. This event has a Birnbaum value of 7E-3/yr.

Service Water

The service water system appears in the upper left quadrant due to its importance to instrument air. This study does not take credit for restoration of cooling water to the air compressors for faults not associated with a loss of station power. For loss of service water, the operators would provide cooling to the air compressors from either the well water system or the demin water system. Should this recovery be credited, it is expected that the service water system would be re-positioned to the lower left quadrant of Figure 13.1-3.

13.2.3 Lower Left Quadrant

Systems that appear in this quadrant of Figure 13.1-3 are those which do not have a significant impact on the current core damage frequency (low Fussell-Vesely) and would not cause a significant increase in the core damage frequency should their performance degrade. The emergency power system has the highest Fussell-Vesely (1.5E-2) of the systems in this group.

Emergency Power

The emergency power system consists of events contributing to the failure of power supply from either diesel generator. Emergency power is required as a back-up to normal power sources and is necessary to provide power to the electric fire pump, the redundant core spray valves, and the post incident pumps following a loss of station power transient (generated by off-site power loss or on-site by bus or breaker faults). The emergency power system has a Fussell-Vesely of 1.5E-2, or 1.5% of the current CDF. The basic events contributing to the emergency power system are:

SPS-I-29	This is a combination of events that result in a failure of the standby diesel generator. Included are the basic events for the failure of the generator to start and run and for the breaker failing to close. This event has a Birnbaum of 5.1E-6/yr.
SPS-I-7	This is a combination of events resulting in failure of the emergency diesel generator. Included are the basic events for the diesel start and run failure and for the output breaker failing to close. The Birnbaum value for this event is 3.3E-6/yr.
EP-GE-DG-CCFTR	This basic event is the common cause failure to run of the two diesel generators. This event has a Birnbaum of 2E-5/yr.
EP-GE-DG-CCFTS	This basic event is the common cause failure to start of the two diesel generators. This event has a Birnbaum of 9E-6/yr.

The emergency power system is underscored by the fact that, for transients, the Big Rock Point Plant can provide decay heat removal and inventory control without ac power being required. The emergency condenser provides initial decay heat removal via the stored water in the shell, and the dc powered fire water makeup valve allows the diesel fire pump to supply shell side makeup. In addition the portable fire pump can supply adequate makeup; this, however, was only credited for the station blackout and flooding initiators.

Instrument Air

The importance of the instrument air system is somewhat overstated in these results. The current results show that the instrument air system has a Fussell-Vesely value of 1E-2 and a Birnbaum of 5.3E-4/yr. The importance of instrument air comes from its use in providing water transfer from the condensate storage to the condenser hotwell, and in providing demin water makeup to the emergency condenser. These two functions are redundant for providing core spray flow from the fire pumps through the core spray valves and for providing fire water to the emergency condenser. No attempt was made to credit fire water to the hotwell for failures associated with the gravity drain of water to the hotwell. In addition, providing water to the emergency condenser (using the portable pump) has not generally been credited. Should these backup measures be credited, it is expected that the instrument air system will fall further to the left in Figure 13.1-3 (making it less important than currently shown).

13.3 Initiator Importance Measures

The calculated initiator importance measures are shown in Table 13.3-1. The Birnbaum value is the slope of the CDF, with respect to the initiating event. The value is unit-less, for initiating events, and reflects the reliability of systems and operator actions available to cope with the initiator.

TABLE 13.3-1
INITIATOR IMPORTANCE MEASURES

INITIATOR	DESCRIPTION	FREQUENCY	FUSSELL -VESELY	BIRNBAUM
S-1	Very Small Below Core LOCA	1.70E-03	2.54E-01	7.98E-03
S-2	Small Below Core LOCA	1.70E-03	1.96E-01	6.16E-03
S-3	Medium Below Core LOCA	1.30E-03	1.19E-01	4.89E-03
T-IA	Loss Of Instrument Air	4.50E-02	9.07E-02	1.08E-04
S-5	Small Above Core LOCA	8.80E-04	8.68E-02	5.27E-03

TABLE 13.3-1 INITIATOR IMPORTANCE MEASURES				
INITIATOR	**DESCRIPTION**	**FREQUENCY**	**FUSSELL -VESELY**	**BIRNBAUM**
A-TT	Turbine Trip with SCRAM Failure	1.10E-05	6.20E-02	3.02E-01
S-6	Medium Above Core LOCA	3.30E-04	4.68E-02	7.57E-03
S-4	Large Below Core LOCA	3.00E-04	2.74E-02	4.89E-03
S-8	Very Small Steamline Break Inside Containment	1.70E-03	2.51E-02	7.90E-04
S-9	Small Steamline Break Inside Containment	1.70E-03	1.99E-02	6.28E-04
S-10	Medium Steamline Break Inside Containment	1.30E-03	1.44E-02	5.94E-04
T-MS	Manual Plant Shutdown	5.60E+00	1.41E-02	1.35E-07
S-7	Large Above Core LOCA	7.70E-05	1.01E-02	6.98E-03
T-DC	Loss of DC Power	4.50E-02	4.06E-03	4.82E-06
S-11	Large Steamline Break Inside Containment	3.00E-04	3.58E-03	6.38E-04
T-T	Turbine Trip	1.10E+00	3.31E-03	1.61E-07
A-FW	Loss of Feedwater with SCRAM Failure	5.00E-07	2.83E-03	3.02E-01
A-BYP	Spurious Bypass Valve Opening with SCRAM Failure	4.90E-07	2.76E-03	3.01E-01
T-BYP	Spurious Bypass Valve Opening	4.90E-02	1.76E-03	1.92E-06
T-F	Loss of Feedwater	5.00E-02	6.46E-04	6.91E-07
T-MC	Loss of the Main Condenser	4.50E-02	4.51E-04	5.36E-07

TABLE 13.3-1 INITIATOR IMPORTANCE MEASURES				
INITIATOR	**DESCRIPTION**	**FREQUENCY**	**FUSSELL -VESELY**	**BIRNBAUM**
A-LOSP	Loss oF Station Power With SCRAM Failure	6.10E-07	3.47E-04	3.04E-02
T-MSIV	Spurious MSIV Closure	1.80E-02	1.80E-04	5.36E-07
T-RDS	Spurious RDS	6.40E-05	1.77E-04	1.48E-04
T-ILCS	Interfacing System LOCA Above the Core	1.20E-07	1.47E-04	6.48E-02
A-IA	Loss of Instrument Air with SCRAM Failure	4.50E-07	1.28E-04	1.52E-02
A-MC	Loss of the Main Condenser with SCRAM Failure	4.50E-07	1.13E-04	1.34E-02
T-ILBL	Interfacing System LOCA Below the Core	3.67E-06	9.78E-05	1.43E-03
S-14	Large Steamline Break Outside Containment	2.80E-05	8.66E-05	1.65E-04
A-MSIV	Spurious MSIV Closure with SCRAM Failure	1.80E-07	4.52E-05	1.34E-02
F-INT	Flooding	1.40E-06	2.10E-05	8.04E-04
S-12	Very Small Steamline Break Outside Containment	3.20E-04	3.79E-06	6.33E-07

Absent from the above list are the initiators for the loss of ac power. The ac power event tree (refer to Figure 7.1.4-1) was used to evaluate failed load rejection, loss of station power and station blackout. The initiating event for the loss of power event tree is the loss of the 138 kv transmission line. Failure to respond to this loss leads to the Load Rejection event tree, and includes the successful transfer of power to the 46 kv transmission line. The loss of station power event tree results from failure to transfer to the 46 kv line. For this analysis, the historical frequency of losing both power lines is used, rather than calculating the frequency based upon the conditional probability of losing the 46 kv line after the 138 is lost (as was done for all other event trees). Finally, the station blackout frequency was determined by quantifying the loss of the two diesel generators (in addition to the loss of station power). Table 13.3-2 shows how the various frequencies were determined.

TABLE 13.3-2 LOSS OF POWER INITIATING FREQUENCIES			
INITIATING EVENT	DESCRIPTION	FREQUENCY (per year)	CALCULATION
138	Loss Of 138 KV Transmission Line (Load Rejection)	0.29	historical data
LOSP	Loss of Offsite Power	0.06	0.29 * 0.20 [(1)]
SBLO	Station Blackout	0.01	0.29 * 0.20 * LR * EA [(2)]

(1) The 0.20 factor is the conditional probability of losing the 46 kv line with the 138 kv line, and is represented by event SP-OO-LOSP-FAIL, which appears in the sequence cutsets. Loss of the 138 kv and the 46 kv transmission lines does <u>not</u> necessarily result in a reactor scram.

(2) The loss of emergency ac power was evaluated using a value of .1 for LR and the results of emergency ac power fault tree, EA, which has a value of .0355.

The Birnbaum importance measures were calculated for the three loss of power transients. Since there is no specific initiator for the event trees, the failures that have to occur in order to put the plant into a specific condition were set equal to 1.0 and the sequences re-quantified. For example, to calculate the importance of the loss of off-site power initiator, the values for the 138 kv line and the 46 kv line were set to 1.0 and then 0.0; the Birnbaum value is the difference between these two core damage values. In addition, the load rejection term and events contributing to a failure to transfer were also changed. The importance measures for the loss of power initiators are shown in Table 13.3-3. The Fussell-Vesely value is the same for all three cases. Due to the calculation of the Fussell-Vesely importance value, the same term is subtracted from the baseline for each study. The Fussell-Vesely calculation requires setting the 138 kv term to zero for each calculation; with the 138 equal to zero, loss of power cannot occur.

TABLE 13.3-3 LOSS OF POWER IMPORTANCE MEASURES			
INITIATING EVENT	BASIC EVENTS MODIFIED	Fussell-Vesely	BIRNBAUM
Load Rejection	138 and LR	5.3E-5	2.4E-5
Loss Of Offsite Power	138, LR, SPS-I-8 and SP-OO-LOSP-FAIL	5.3E-5	1.7E-4
Station Blackout	138, LR, SP-OO-LOSP-FAIL, SPS-I-29, SPS-I-8 and SPS-I-7	5.3E-5	8.7E-3

13.4 Systems

The uncertainties and conservatism associated with the system models are included in Table 13.4-1.

13.5 Data

The sensitivity to the failure rate calculation for each basic event can be obtained by the Birnbaum importance calculation; events with a Birnbaum greater than ten times the baseline CDF are shown in Table 13.1-2. For the basic events, the Birnbaum value is an estimate of the core damage frequency should the component fail with a probability of 1.0. Comments regarding uncertainties in the data used are included in Table 13.4-1.

13.6 Human Action Uncertainties

13.6.1 Scope and Modeling

The human actions are separated into three basic categories: latent, dynamic, and recovery human actions. The human actions were analyzed in two stages. Initially, all human actions were analyzed using the ASEP methodology, which typically results in conservative HEPs. Important human actions were later reanalyzed using the Technique for Human Error Rate Prediction (THERP) from the Handbook (NUREG/CR-1278).

13.6.1.1 Latent Human Actions

The Accident Sequence Evaluation Program (ASEP) Human Reliability Procedure (Reference 10-1) was used for quantification of all latent human actions. The ASEP methodology quantifies latent human errors to include analysis of errors of commission and errors of omission. Potential recovery of the actions is credited when there is a second check, a periodic check, or a post-maintenance/ post-calibration test. ASEP also specifies that, if the error is evident in the Control Room, an upper bound of 1E-5 for the HEP should be used. The analysis using the ASEP methodology is consistent with standard HRA approaches to latent human error analysis.

Table 10-1 provides the results of the latent human error HRA calculations. This table also provides a summary of when each of the three potential recoveries was used, and when actions were evident in the Control Room. In general, the results using the ASEP methodology were conservative, due to the conservatively high error rates estimated for the errors of omission and commission, and conservatively high HEPs used for periodic check and post-maintenance/

post-calibration testing. The resulting HEPs are typically a factor of ten higher than what would be expected by use of a more detailed methodology such as the Handbook [Ref 10-2].

Based on the PRA results, none of the latent human errors were found to have a high contribution to the core damage frequency. Only one latent human error was found to have a potentially significant contribution to core damage, if failure was always assumed to occur: restoration of post incident system valves to their correct position after system testing. The restoration of these valves was also identified as potentially significant in the 1981 PRA. As a result of that analysis, position locks were placed on the valves to assure that they could only be left in their correct position. These position locks were credited in the evaluation of latent human errors in the current PRA, which demonstrates their effectiveness.

13.6.1.2 Dynamic and Recovery Human Actions

Dynamic and recovery human actions are actions which occur during or following an event. These include actions in response to an event, and recovery actions in response to system failures. All dynamic and recovery actions credited in this analysis were addressed in the plant procedures and are, therefore, mostly rule based in nature (i.e., the operator performs certain actions based upon the rules and guidance provided by the plant procedures).

Dynamic and Recovery Actions Calculated Using ASEP

The dynamic and recovery actions calculated by using the ASEP methodology were initially found to have low importance, and are considered conservative. The actions were analyzed to include operator errors to diagnose the problem, errors of omission and commission, and errors of recovery. In some cases, errors to diagnose the problem were considered time dependent, and the diagnosis HEPs were calculated using the nominal diagnosis model in ASEP. This is the same diagnosis model used in the Handbook methodology. In other cases, the diagnosis was dominated by failure to respond to an alarm, and was calculated using the annunciator response model. Errors of omission and commission were conservatively assigned based upon Table 8-5 in ASEP. Recovery was also assigned using these guidelines. In all cases, only a single operator recovery was credited for each action. This recovery model is considered very conservative, especially for actions in which multiple procedural checks are expected or a long time frame is available for the action.

Dynamic and Recovery Actions Calculated Using the Handbook

Actions initially found to be important for the PRA were reanalyzed using the Handbook methodology. This methodology requires a more detailed task analysis than the ASEP methodology, and includes more detailed HEP calculations to account for:

- Control Room and support crew size
- Types of procedures being used (i.e., with/without sign-offs) and verification steps
- Errors of commission/omission based upon the type and location of controls
- Expected stress
- Timing
- Dependence between operators and operator actions

This methodology is discussed in detail in Section 10.2. This methodology is consistent with other PRAs, and is consistent with past applications using the Handbook. The single difference between the applied methodology and past methodology is that human error event trees were not created for this analysis due to the time frame required to create graphical representations for each HEP. This does not affect the results since the total HEPs are calculated in the same manner as when supported by an event tree.

Dependencies between operator actions were analyzed based on the methodology discussed in Chapters 7 and 10 of the Handbook. This methodology accounts for levels of dependence, both positive and negative, with the levels of dependence calculated based upon the timing for each action, the procedures used for each action, the operators involved, the functional relationships between the tasks, and other factors. Initially, actions found to be potentially dependent on another action were set to 1.0. If found to be important, the actions were reanalyzed using the Handbook methodology. Only two actions were reanalyzed for determining dependence between HEPs.

13.6.2 Quantification and Data

Human error modeling produces relatively uncertain results, regardless of the HRA methodology used. There are a number of issues that affect the certainty of human error estimates, including both generic issues and applicability of HRA techniques to a particular application. Some generic issues that affect the certainty of the results include:

1. The availability of data on human performance in nuclear power plants in a form suitable to support HRA;

2. The development of HRA models that are applicable to all situations and conditions expected;

3. The identification of all relevant Performance Shaping Factors (PSFs) and their effect on HEPs;

4. The skill and knowledge of human reliability analysts;

5. The effect that plant management and emergency management structures have on human performance; and

6. The variability in performance of an individual and among the performances of different individuals and shifts.

Additionally, application of generic HRA techniques to a particular application introduces uncertainty. It is difficult for a methodology to account for all of the variability that is expected from plant to plant. For example, one plant may train extensively for a loss of instrument air procedure with both simulator and classroom training, while an identically designed plant may only have minimal training for this event. Factors such as the complexity of the plant, average experience of the operations crew, location of emergency support personnel, and other areas are difficult to factor into the HRA.

The approach used for the Big Rock Point HRA was conservative in order to minimize the affect that any uncertainty in the HRA would have on the final results. This process included:

1. Initial conservative estimates were used for all HEPs. These were either set to 1.0 or analyzed using the ASEP methodology.

2. HEPs considered potentially dependent on other HEPs, were initially set to 1.0. Only important dependent HEPs were analyzed using dependency methods established in the Handbook.

3. The important HEPs were analyzed in detail using the Handbook. Because only a few actions were reanalyzed in great detail, the HRA analyst could concentrate on these actions during the performance of interviews, walkdowns, and analysis. Where assumptions were required, conservative assumptions were used (i.e., timing).

13.6.2.1 Uncertainty in Latent Actions

The error factors provided for each of the latent action HEPs in Table 10-1 were derived from calculations on pages 5-16 and 5-17 of the ASEP report. In most cases, the resulting error factors were 5 for HEPs greater than 1E-2, and 10 for HEPs less than 1E-2. As discussed above, the latent human errors calculated are, on average, a factor of 10 (conservative due to the use of the ASEP methodology).

13.6.2.2 Uncertainty in Dynamic and Recovery Actions

The error factors provided for each of the ASEP dynamic and recovery HEPs in Table 10-2 were derived based on the dominant cause for the operator failure. For example, if the HEP was dominated by diagnosis failure, the error factor for the HEP was based on the error factor calculated from the nominal diagnosis curve. If the HEP was dominated by failure to perform a critical step, the error factor assigned was from the ASEP table for this action. This results in similar results to what would be calculated by combining error factors from the HEPs, except that extensive calculations are not required. This is considered adequate since the dynamic and recovery actions which were calculated using ASEP are considered unimportant. Additionally, ASEP assigns conservatively high error factors to the HEPs.

The error factors provided for the Handbook dynamic and recovery HEPs in Table 10-2 were derived based on the guidelines provided in the Handbook. In most cases, error factors were estimated from Table 7-2 of the Handbook. Most HEPs analyzed were assigned an error factor of 5 or 10 based on these guidelines.

The error factors assigned to the HEPs are nominally much higher than those assigned to mechanical failures or initiating events. An error factor of 10, for example, results in a 10 to 1 ratio between the upper confidence bound estimate and the median value, and a 100 to 1 ratio between the upper and lower confidence bounds. In comparison, a valve failure with an error factor of 3 has ratios of 3 and 9, respectively. For both the Handbook and ASEP, these upper and lower confidence bounds correspond to the 5th and 95th percentile, respectively, for a lognormal distribution for the HEP.

13.6.2.3 Operator Action Importance

The operator actions (in response to an initiating event) that appear in the sequence cutsets that: 1) were not set equal to one and 2) have a Birnbaum importance measure greater than 1E-6 are shown in Table 13.6-1. The post event operator actions with a Fussell-Vesely of 1E-2 or greater (1% of CDF) are discussed below.

TABLE 13.6-1 POST-EVENT OPERATOR ACTION IMPORTANCE MEASURES			
Operator Action	**Probability**	**Fussell-Vesely**	**Birnbaum**
RD-OO-PCNTL-POIC	1.50E-01	1.01E-01	3.59E-05
CD-HS-P9TRP-POIC	8.00E-02	7.23E-02	4.84E-05
LI-OO-INJ2-POIC	3.00E-01	6.72E-02	1.20E-05
RP-RX-VSSLB-POIC	6.00E-04	4.80E-02	4.28E-03
FP-OO-MAKUP-POIC	1.60E-02	1.45E-02	4.84E-05
PI-OO-PISYS-POIC	8.30E-05	1.01E-02	6.48E-03
CS-MV-CSVLV-POIC	6.20E-03	6.47E-03	5.58E-05
FW-PM-P8SRT-POIC	4.60E-03	5.89E-03	6.85E-05
EC-MV-ECOUT-POIC	1.00E-02	5.07E-03	2.71E-05
MS-OO-ISOLT-POIC	1.30E-02	2.65E-03	1.09E-05
EP-GE-SDG-POOC	8.80E-03	3.26E-04	1.98E-06
LI-OO-INJ12-POIC	1.00E-02	2.87E-04	1.54E-06
EM-KV-4947-POIC	8.50E-05	2.74E-04	1.73E-04
RP-OO-MANUL-POIC	5.00E-03	1.80E-05	1.50E-05

RD-OO-PCNTL-POIC

The most significant of the operator actions (based on Fussell-Vesely) is the operator failure to provide manual pressure control (RD-OO-PCNTL-POIC). The importance of this event is derived from its inclusion in all of the loss of instrument air event tree dominant cutsets. Manual primary system pressure control is required in these sequences because of a failure of the emergency condenser to remove decay heat, which is dominated by a failure to provide makeup to the shell by the fire system. The loss of instrument air event tree contributes about 10% of the base CDF. Listed below are several reasons the importance of this action are overstated in the current results:

- The portable fire pump is not credited in the loss of instrument air event.
- The event which models failure to reclose the EC level instrument drain valves is overly conservative in that emergency condenser failure is assumed, no attempt for discovery of this valve mispositioning is assumed prior to the initiating event.

- No credit for the recovery of instrument air is assumed to occur after the initial loss, this includes use of the newly installed service air compressor (fourth air compressor on-site).

The low Birnbaum value (3.59E-5) reflects that the CDF is somewhat insensitive to the value assigned for this event. This is primarily due to such a high probability being used in the current CDF (.15).

CD-HS-P9TRP-POIC

This event represents the operator failure to trip the condensate pump breakers after a pump trip on low hotwell level. This action is important, in that the condensate pumps can be utilized as a low pressure makeup source to the primary system during certain steam line breaks and LOCAs where fire water to the hotwell is required. It was assumed that, during these transients, uncontrolled feedwater-condensate flow into the primary system, before fire water is aligned to the hotwell, would cause the condensate pumps to drain the condenser hotwell below the low level trip point. If the operators do not place the condensate pump breaker control switch to trip, it was assumed that the breaker would cycle open and closed. The breaker control circuit is configured such that, if power is available, as soon as the breaker trips on low level, the auto close circuit is completed. If level has not been restored, the breaker will trip again. This cycle is expected to continue until a) hotwell level is maintained above the trip setpoint, 2) the operator trips the breaker with the hand switch, 3) power to the breaker is lost, or 4) the breaker fails.

This event does not take into consideration the potential of flashing in the feedwater steam drum level, Bailey instrumentation reference leg. If the reference leg flashes, the feedwater control system will interpret this as a false high drum level condition; coupled with the drop in steam flow through the main steam line, the control system will respond by closing the feed reg valve. With the valve closed, no additional depletion of the condenser hotwell will occur. If the condensate pumps are still operating (low hotwell level has not occurred), the pumps will automatically go to recycle (CV-4011 opens to maintain approximately 400 gpm condensate flow through the inter\after condenser) and maintain feedwater pump suction pressure. The condensate pump discharge flow returns to the hotwell after passing through the gland seal condenser and inter\after condenser.

If the condensate pumps had tripped due to low hotwell level (without depletion to the steam drum), the hotwell fill line will bring the level back to the point at which the pumps will restart. After restart, the pumps would then begin to recycle water back to the hotwell.

The low Birnbaum value (4.8E-5) reflects that the CDF is somewhat insensitive to the value assigned for this event. This is primarily due to such a high probability being used in the current CDF (8E-2).

LI-OO-INJ2-POIC

The liquid poison system is a manually initiated system; therefore, the amount of time available for the operator to diagnose and respond is critical for human error analysis. This event is operator failure to initiate poison injection within 2 minutes. The 2 minute time frame is based on the operator initiating poison injection prior to the automatic actuation of RDS. This action has a relatively high Fussell-Vesely value (6.7E-2), as operator action is the principal means of mitigating an ATWS. The dominant ATWS sequence is the turbine trip with bypass. For this sequence, the feedwater pumps are assumed to fail due to a low suction pressure caused by the opening of the condensate reject valve. With the open bypass line to the main condenser and limited high pressure makeup (CRD flow of 25 gpm), reactor water level quickly drops. The RDS 2-minute timers delay the initiation of RDS and provide the operator with a limited time to inject poison.

The low Birnbaum value (1.2E-5) indicates that the CDF is somewhat insensitive to the value assigned for this event. This is primarily due to such a high probability being used in the current CDF (.30).

RP-RX-VSSLB-POIC

This event is the operator failure to respond to a very small breech in the primary system by scramming the reactor. Because the containment is continuously vented during normal operation, pressurization of containment and automatic trip of the reactor is not expected for breaks of this size. While scram of the reactor is an appropriate action for the operator to take under these circumstances, a significant amount of time is available for the operator to take this action (that was not credited in the derivation of the HEP).

FP-OO-MAKUP-POIC

This HED includes those actions required to provide fire water to the hotwell for continued injection of water from the condensate system. Fire water to the hotwell is credited for supplying low pressure core cooling as a backup to core spray valve failures for un-isolated breaks outside of containment, large steam line breaks inside containment, very small and small LOCAs below the core, and for large above core LOCA. In addition, fire water to the hotwell is required for the condensate pumps to provide core cooling in the fill the ball sequences. The basic event for supplying fire water to the hotwell (FP-OO-MAKUP-POIC) has a Fussell-Vesely of 1.5E-2 and a Birnbaum of 4.8E-5/yr. This relatively low Birnbaum value is indicative of an event that has a minimal effect on the core damage frequency.

PI-OO-PISYS-POIC

This basic event represents the operator failure to respond to the increasing water level in containment (following a breech in the primary system), indicating the need to switch the core cooling mechanism from injection mode to a recirculation mode. The time frame associated with this action is on the order of hours before the water level reaches the recirculation setpoint. This time frame is reflected in the low Fussell-Vesely value (1E-2) for the event. In contrast, the Birnbaum for the event is 6.5E-3/yr, which is the highest Birnbaum for the events included in Table 13.6-1. This is a result of not taking credit for the continued injection of core cooling flow and filling containment.

13.7 Level II Sensitivity Evaluation

As part of the containment evaluation, there are phenomenological and probabilistic issues (e.g., system reliability, operator action) that can have a large impact on the course of the events or the radionuclide release magnitude and timing. Both types of issues become candidates for sensitivity analysis. The BRP CET provides a structure to perform sensitivity studies on issues for which a large uncertainty may exist.

Phenomenological uncertainties are addressed in this section. These uncertainties are addressed quantitatively using deterministic computer calculations to simulate alternative modeling assumptions. Any new accident management insights from these sensitivity evaluations are included in Section 2.8.

This section includes the following information:

- Approaches to sensitivity (Section 13.7.1),
- Issues for which an uncertainty or sensitivity study is desirable (Section 13.7.2), and
- Deterministic sensitivity studies (Section 13.7.3).

Table 13.7-1 (Table A.5 from NUREG-1335) identifies parameters that past studies indicate as prudent choices for sensitivity cases.

The sensitivity cases chosen for MAAP-BRP investigation in the BRP PRA were further influenced by recommendations contained in the NRC sponsored MAAP3.0B code evaluation. The MAAP3.0B code, in its third phase, reached a consensus on recommended sensitivity runs which were necessary to provide utility staff with an appreciation of the uncertainties surrounding containment performance during a severe accident [Ref 13-1].

Most of the resources of an IPE back-end analysis effort are devoted to treating uncertainties that could directly influence accident management strategies, in general, and containment failure time, in particular. Stated more narrowly (from the standpoint of accident management), the principal goal of performing sensitivity studies is to identify and understand physical phenomena that put a premium on specific operator actions. In addition, accident management actions have been identified as effective for controlling or preventing postulated phenomena under certain accident sequence conditions or assuming certain modeling conditions. These phenomena may not be physically possible or may behave differently than the modeling assumptions. In the future, it may also be prudent to investigate the impact of the accident management actions over a range of postulated physical models on phenomenological assumptions.

Fewer resources should be devoted to phenomena that, to varying degrees: 1) are generic rather than plant-specific; 2) are being studied elsewhere on a generic basis; or 3) which do not impact accident management strategies directly, even though they could affect the source term from a given sequence. For such phenomena, only best-estimate treatments are recommended for the purpose of developing the PRA.

The results of these sensitivity cases are described in Section 13.7.3. The following two sections identify possible approaches to performing the sensitivity analysis and the methods chosen for BRP.

13.7.1 Sensitivity Approaches

The approaches for investigating key sensitivities can take on a wide range of methods and cover a wide spectrum of breadth and depth of investigation. This section identifies optional approaches that could be used to satisfy different objectives as follows:

<u>Resource Intensive Approach</u>:

Identify all parameters or modeling assumptions that have uncertainties larger than an error factor of three, and include a sensitivity of varying these. In addition, identify coupled parameters that also need to be varied.

IPE Approach:

Satisfy the requirements of IPE Generic Letter 88-20 for the Level 2 portion of the IPE. Address the phenomenological issues posed by the NRC:

- Probabilistically, or
- Deterministically.

As a result, it is necessary to identify a limited sample of additional containment or plant-specific issues that should be addressed.

Big Rock focused on the containment phenomenological issues rather than the systematic results. This decision was based on the experiences garnered from the high containment isolation unavailability noted in the 1981 PRA, the application of the Appendix J leakage data, the availability of secondary-side leakage data, the availability of extensive plant data on containment bypass contributors, and the results of the detailed fault tree model of the containment isolation, bypass and spray function.

Accident Management Sensitivity Approach:

This group of sensitivities would be developed to support additional investigations in an attempt to optimize accident management actions or hardware use that could be implemented as part of an accident management response to severe accidents.

Conclusion

As part of this part of this study, Big Rock has selected the IPE approach and furthermore has focused on the phenomenological uncertainties rather than the systematic uncertainties. Therefore, the following sections will present the results of sensitivity assessments for a group of selected issues as affected by plant-specific features. The uncertainties are, in general, addressed quantitatively, using deterministic methods. In a few select cases, the uncertainties are discussed qualitatively in order to ascertain their impact on accident management actions. Big Rock Point has already included many of the industry identified accident management strategies and has incorporated them into the EOPs. This activity used the insights gained from the 1981 Level III PRA results and whose conclusions have not changed as a result of this study (see Section 2.5.3).

13.7.2 MAAP-BRP Sensitivity-Runs Overview

To ensure that a broad scope of possible severe accident progression is considered in the BRP PRA, several sensitivity analyses were performed using the MAAP-BRP code. The MAAP code has been widely used in the industry for Level I and II IPE quantification and has been independently reviewed for its suitability by Brookhaven National Laboratories (BNL) under an NRC contract. The BNL review concluded:

> *"...MAAP should be used to provide the utility with a framework for obtaining an understanding of containment failure modes, the impact of phenomena and plant features, as well as operator actions. In this role, MAAP analyses should be supplemented with sensitivity studies to ensure that the utility staff has an appreciation of the uncertainties surrounding containment performance during a severe accident. ."* [Ref 13-1]

MAAP-BRP cases were selected to evaluate the key functional events for mitigating radionuclide releases associated with severe accidents at the BRP plant. This nominal set of MAAP-BRP calculations represents a best estimate of how the plant will respond under severe accident conditions. However, it is recognized that considerable uncertainty exists in the modeling of the complex phenomena associated with such accidents. One should recognize that MAAP-BRP does not contain detailed models for all phenomena. Indeed, there are more mechanistic codes available, such as CONTAIN and SCDAP/RELAP. These are generally used in a research setting and are not considered suitable for use in IPEs due to long run-times and the much greater requirements they impose on the user for specialized knowledge of severe accident phenomena. An alternative code, whose scope is similar to MAAP-BRP, is MELCOR. However, less experience has been accumulated with the MELCOR code than with MAAP. Therefore, MAAP-BRP was chosen as the best available tool to perform the plant specific evaluation.

Table 13.7-2 summarizes an extensive list of possible sensitivity calculations that could be performed to support a full PRA. Within Table 13.7-2, the phenomena or items identified are those which are:

- Recommended by GL 88-20 or NUREG-1335 to be addressed as part of the IPE;
- Deemed sufficiently important to address; and
- Useful for consideration in an accident management program.

13.7.3 Deterministic Sensitivity Studies

As part of the evaluation of uncertainties through the performance of specific sensitivity cases, Big Rock has reviewed the status of the NRC position on the series of so-called "issue" papers. Those that are carried forward to the performance of specific sensitivities include the following:

- Core melt progression
- In-vessel hydrogen generation
- RPV pressure at vessel failure
- Late CsI revaporization from the RPV
- Debris spread in containment
- Amount of debris retained in RPV
- Ex-vessel debris coolability
- Containment failure location
- Containment failure area

In addition, the MAAP-BRP model parameters generally represent inputs to phenomenological models in which significant uncertainties exist. Variations in the values of these parameters can be made in order to assess the impact of uncertainties in important physical models. The best-estimate values used in the BRP PRA are provided in the BRP MAAP Parameter File. These best estimate values were taken from the "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," Gabor, Kenton and Associates, EPRI 1990. Sensitivity analyses were performed in accordance with the recommendations in the EPRI/GKA report [Ref 13-2] and the BNL/NRC report [Ref 13-1], as well as additional areas deemed important for BRP.

The resulting list of deterministic sensitivities performed for BRP is a combination of the NRC "open issues" and the GKA-recommended sensitivities and includes the following:

- Core Melt Progression: Amount of Residual Debris in RPV (Section 13.7.3.1)
- In-Vessel Hydrogen Generation: Core Blockage (Section 13.7.3.2)
- RPV Pressure at Vessel Failure (Section 13.7.3.3)
  - Non-Condensible Gas Generation

- Debris Cooling in the Sump
- Concrete Attack
- Containment Debris Spreading

* Debris Coolability (Section 13.7.3.4)

* Containment Failure Mode - Size and Location (Section 13.7.3.5)

* Containment Flooding Sensitivity Evaluation (Section 13.7.3.6)

13.7.3.1 Core Melt Progression: Amount of Residual Debris in RPV

The amount of core material remaining in the RPV is calculated by MAAP-BRP. The core begins to melt and then relocates into lower regions of the core. This continues until the lowest core node in any radial region becomes completely molten, at which time all molten core material exits the core region and moves into the lower plenum.

In past MAAP analyses, it has been observed that the amount of material molten at the onset of movement into the lower head is strongly dependent on the amount of in-vessel Zircaloy oxidation. More oxidation tends to heat up the core and results in a larger mass of molten material moving out of the core region. Due to various modeling assumptions and a general lack of detail in representing core melt progression, there is a spectrum of results, ranging from all the material exiting at vessel failure to cases in which a significant fraction of the core material remains behind in the RPV.

The BWRSAR model includes the fact that, after the lower plenum dry-out, the debris bed temperature would increase, causing thermal attack and failure of the control rod guide tube structure in the lower plenum. Since the control rod drive mechanism assemblies and the control rod guide tubes support the core, the remaining standing outer regions of the core would be expected to collapse into the vessel lower plenum when these support columns fail.

The BWRSAR model for core melt progression and RPV bottom head attack supports the idea that little residual debris would be retained in the RPV. With the exception of the small LOCA case, the MAAP-BRP results indicated that significant amounts of core material would remain in the vessel long term, even in cases with large LOCAs or depressurization without injection. The amount of core material remaining in the vessel (following RPV failure) can have a significant impact on the key results from an analysis.

If all of the core material exits the RPV, more mass will be available for core/concrete interaction, core/water interactions, and debris transfer to the enclosure node. If core material remains behind in the vessel, it may contribute to late fission product revaporization and containment heat-up due to radiative heat transfer from the RPV to the containment atmosphere. The MAAP-BRP parameter FMAXCP specifies the minimum amount of core material capable of supporting the remainder of the core. When the fractional amount of core material remaining in the core region is less than FMAXCP, the remaining core material is forced out of the core.

In the MAAP-BRP sensitivity analysis, two MAAP cases are rerun with a value of FMAXCP large enough to prevent debris from residing in the RPV after vessel failure. In these sensitivity cases, the vessel fails at low pressure (one case with containment sprays available, the second with sprays failed off). The base case assumption used in all of the MAAP-BRP analyses is that FMAZCP is 0.10. That is when 90% of the core has melted, and the remaining 10% is forced out of the vessel and into the CRD pump room sump. Core melt progression, in most BRP base cases, does not reach the 90% value so that base cases typically retain 50% of the original core mass in the vessel long term. The two MAAP-BRP sensitivity cases set FMAXCP equal to 0.60, so that only 40% of the core material is required to melt before the remainder is forced to exit the vessel. Table 13.7-3 summarized the key results from this sensitivity evaluation.

The moderate reduction in releases for the sensitivity Cases 1a and 2a are typical in situations where all the debris is allowed to exit the vessel). Table 13.7-3 shows that these cases have lower long term RPV gas temperatures.

Sensitivity Case 1a, with 100% of the core forced out of the vessel, has no containment sprays available during the sequence and takes 10 hours from vessel failure for the CRD room to dry out and the core debris to reheat and begin to attack the concrete basemat. This can be contrasted with Base Case 1, in which 52% debris retention in the RPV allows only a limited amount of corium to be deposited in the CRD sump. The smaller amount of debris is insufficient to both dry out the compartment and then reheat to the temperature required for CCI to initiate. In Case 2a, with containment sprays operating, the additional corium forced out of the vessel (at vessel failure) is covered with water for the duration of the sequence. The additional heat transfer to the water (from the balance of corium) increases the steam partial pressure. In both sensitivities (Case 1a and 2a), the increases in sphere temperatures and pressures, due to CCI and increased steaming, respectively, are marginal and are therefore insufficient to threaten the integrity of containment.

This set of cases was also used to examine the issue of late revaporization of fission products from the RPV. In Base Case 1, which had no containment sprays, approximately 2% of the total CsI is revaporized from the RPV in the last 10 hours of the run. This can be compared with the sensitivity cases with no core material in the vessel and Case 2 with containment sprays available to cool the vessel, one of which showed evidence of late-fission product revaporization. To illustrate a case with large amounts of revaporization, Table 13.7-3 includes Case 4, which is identical to Case 1, except that early failure of containment was forced at vessel failure. In Case 4, the lower containment pressures (after vessel failure) require higher RPV gas temperatures in order to dissipate the heat originating from the debris remaining in the vessel. These higher RPV gas temperatures result in approximately 10% of the total CsI revaporizing from the vessel in the last 10 hours of the run, which significantly contributes to the high source term for this early failure scenario.

13.7.3.2 In-Vessel Hydrogen Generation: Core Blockage

Uncovery of the core occurs in each of the core damage sequences investigated. As the core becomes uncovered, the clad begins to oxidize, producing hydrogen as a by-product. Eventually, melting and relocation of the core material ensues, with the potential for blocking steam flow and reducing additional clad oxidation. By setting the value of the MAAP-BRP parameter FCRBLK, different options are available for treating the resulting effects from melting and relocation of core material.

Base Case: Local Blockage (FCRBLK = 0)

Setting FCRBLK at 0 causes zircaloy oxidation to cease within a core node when the temperature of the core exceeds the eutectic melting temperature of 2500K.

This case assumes that the melting of an individual node does not affect the steam flow to nodes above that location so that, as long as those nodes are under 2500K, they will continue to contribute to hydrogen production.

Channel Blockage: (FCRBLK = 1)

In this case, a core node that reaches the eutectic melting temperature is assumed to melt and completely fill its associated channel, thus preventing steam from rising to nodes above the blockage. The plugged channel then pressurizes and forces any remaining water out the bottom, thus isolating the channel. This results in the immediate cessation of oxidation along all nodes in that radial channel, thereby greatly reducing the amount of hydrogen generated in the melting process.

No Blockage Case: (FCRBLK = -1)

In this case, a node reaching the eutectic melting temperature has no effect on steam flow within its channel and oxidation continues for all nodes at the same rate as before the onset of melting.

The increased core exit temperatures that typically occur with the no-blockage and local-blockage options will tend to result in early RPV fission products being released to containment either through the relief valves or by an early RDS. If it were not for the very large volume of the BRP containment and the high rates of mixing shown to be effected in Section 12.4.4, the increased amounts of hydrogen produced could possibly pose a threat to the BRP containment either through over-pressurization or by ignition in the non-inerted atmosphere.

The local blockage option (FCRBLK = 0.0) with double-sided oxidation (FUMIN = 1) was selected for the BRP bases cases as being the most realistic. Experience has shown that low pressure cases are less affected by the choice of FCRBLK, as the use of RDS to depressurize the vessel quickly leads to loss of inventory and core uncovery, with little water available to supply steam for oxidation of the cladding and the production of hydrogen. On the other hand, high pressure failure cases can be sensitive to the choice of core blockage model. Typically, a high pressure case requires a longer time period to lose its inventory, thus allowing significant amounts of cladding oxidation to take place.

Case 15, a high pressure failure with no containment sprays, was chosen as a base case to investigate the sensitivity of hydrogen generation to the choice of core blockage model. Case 15c was run with the no blockage model (FCRBLK = -1.0) and double-sided oxidation (FUMIN = 2) in order to compare hydrogen production and long term temperature and pressure conditions in containment with Base Case 15. Table 13.7-4 shows the release magnitude and key event timing for these runs.

As expected, the greatest amount of hydrogen is produced when there is no channel blockage and double-sided oxidation is assumed. The higher core exit temperatures in the sensitivity case allow for more melting of the core and a greater amount of debris being entrained in the steam jet following vessel failure. The additional debris that accumulates in the enclosure room leads to core concrete attack and the generation of additional hydrogen. However, the large containment volume is able to absorb the additional loading of non-condensibles and still maintain a sphere pressure of about 50 psia. In conformance with the analysis performed in Section 12.4.4, in which the use of the generalized containment model showed that internal circulation in the containment kept hydrogen from accumulating in any given node, the sensitivity Case 15c showed no evidence of hydrogen burning. Consequently, it is judged that the BRP containment is insensitive to the choice of core blockage model and that, in any event, in-vessel hydrogen generation will not contribute significantly to containment failure frequency. Therefore, the selection of the blockage model used in the BRP base cases was not dictated by which was the most "conservative," but rather which was considered more realistic. Therefore, FCRBLK - 0.0 (local blockage) was chosen.

13.7.3.3 RPV Pressure at Vessel Failure

The use of RDS to depressurize the vessel (before the core uncovers) can have a significant effect on the evolution of a sequence. When water level in the vessel falls to below 2' 9" from the top of active fuel, RDS is activated, thus causing a rapid depressurization of the vessel with a significant loss of the remaining inventory. There is enough water left in the vessel to cool the core for between 1 and 2 hours, at which point level has fallen so low that steam cooling is no longer sufficient to prevent the hottest nodes of the core from reaching the melting point. Once core melt begins, vessel failure follows within minutes, with corium debris flowing out of the vessel onto the CRD floor. This scenario is in contrast to high pressure failure cases in which RDS is not used. With the vessel at pressures above 1500 psia, the relief valves allow a gradual loss of inventory as steam is vented to containment. Boil-down takes longer, with core uncovery coming 4 to 5 hours later than in a low pressure case. This delay in vessel failure allows time for additional oxidation of the fuel rod cladding and generation of hydrogen, thus increasing gas exit temperatures and pressurizing the containment. Vessel failure, when it occurs, is much more dynamic; steam, with corium entrained, transports debris to the enclosure room as well as the CRD room while the containment is subject to a pressure spike that might prove severe enough to threaten its integrity.

Table 13.7-5 shows comparative results for vessel failure cases occurring at high and low pressures, with the additional effect of containment sprays.

Cases 15 and 15d, high pressure failure cases, are compared respectively with Cases 1 and 2, equivalent low pressure cases. In the first high/low pressure comparison, containment sprays operate, while in the second comparison, sprays are not available. It can be seen from Table 13.7-5 that, even without sprays, the BRP containment remains intact after both high and low pressure failures. Peak pressures in the sphere occur either at vessel failure (< 42 psia) or 36 hours later, at the termination of the MAAP-BRP sequence (< 46 psia). Without containment failure, the magnitude of the source term is determined by the assumed normal leakage rate of 0.5% per day and is insensitive to whether the vessel fails at high or low pressure. The use of containment sprays to scrub the containment atmosphere of fission product aerosols can reduce the source term by an order of magnitude as seen in the pair of high pressure Cases 15 and 15d in Table 13.7-5.

Case 15a also used sprays to cover the containment floor with water, but introduced them at ground level, precluding the scrubbing effect of water droplets falling through the containment atmosphere. The source term for Case 15a is similar in magnitude to the cases without sprays.

Variations on High Pressure Sequences: Forced Early Containment Failures

High pressure vessel failure cases appear to have little likelihood of failing containment at the time of vessel failure. Peak containment pressures reached in the seconds after vessel failure typically range between 38 and 44 psia, which corresponds to a cumulative containment failure probability of less than 1% (Figure 12.3-1). To examine the consequences of the containment pressure spike damaging containment integrity, Case 14 was run. In this case, a hole (2 feet in diameter) was forced open to the environment at vessel failure. A second variation assumed that containment sprays were operational before and after the early failure. The key results for these cases are summarized in Table 13.7-6.

Case 15 poses the most extreme test of containment integrity by a transient. Pressure is allowed to remain high in the RPV until core melt and vessel failure. Table 13.7-5 shows that the sphere pressure rises to 41.6 psia in the few moments in which blowdown occurs. This pressure spike is not expected to fail containment and the small source term that results comes from the normal leakage, assumed to be 0.5% per day. Without containment sprays to scrub the fission products aerosols, this low-low source term is typical of the MAAP-BRP "normal leakage" cases.

Case 14 was run with a 3.14 ft^2 containment break opened at the time of vessel failure, allowing the containment to quickly depressurize. With no sprays available, fission products have a direct path to the environment, thereby generating a high release. In Case 14a, enclosure sprays are activated 0.36 hours after the initiation of the transient. This delays vessel failure by 0.20 hours and reduces the magnitude of the associated pressure spike. Unlike Case 14 (where fission products were released for the entire 36 hours), it takes 24 hours for the active sprays in Case 14a to scrub the remaining aerosols in the containment atmosphere and cool the core debris in the vessel, thereby stopping further release of fission products to the environment. Case 14d shows that the use of containment sprays can reduce the source term in forced early cases by an order of magnitude.

13.7.3.4 Debris Coolability

Without continued water injection after vessel breach, the core debris will dry out and begin to heat up. Eventually, the debris will begin to interact with the concrete basemat. There is also the possibility that core concrete attack can occur in the presence of an overlying water pool. Prior to containment failure, any fission products that are evolved by core concrete attack or by long term revaporization will be deposited in the containment. At containment failure, the amount of fission product release will be dictated by the airborne mass of radionuclides at failure and the subsequent rate of their revaporization from the containment and RPV.

Four separate aspects of debris coolability will be discussed:

- Non-condensible gas generation
- Debris cooling in the sump
- Basemat attack
- Containment debris spreading

Non-Condensible Gas Generation

There has been a substantial amount of disagreement on the issue of debris coolability ex-vessel. Some analyses indicate that water will ingress into the debris and provide cooling. Others have pointed to the very limited experimental data base and concluded that an impermeable crust will form which isolates the water from the debris. The EPRI document on MAAP sensitivity analysis indicates that selected cases should be run assuming that the debris-to-water heat transfer is limited to approximately 300 kw/m^2. Later work (based on the results of the ANL MACE series of experiments) suggest that 100 kw/m^2 may be a better lower-bound for the debris to water heat transfer coefficient. The latter value was selected to use in the BRP sensitivity study for non-condensible gas generation.

The low pressure Cases 2 and 3 were selected as base cases on which to vary the coolability parameter FCHF. In both cases, a transient is followed by RDS activation and the initiation of containment sprays. The vessel fails at low pressure and the corium debris flows into the sump in the CRD room floor and is covered by water. The sump is sufficiently large to hold all the debris without overflow. In Case 2, the sprays switch to recirc mode when the water height reaches 8 feet in the CRD room, while in Case 3, the failure to switch to recirc allows the water level to rise. Both cases utilize the MAAP-BRP default value for FCHF of 0.09, or 900 kw/m^2 as the upward heat transfer rate between the debris and the overlying water pool. In sensitivity Cases 11 and 12, FCHF was fixed at 100 kw/m^2, while all other conditions remained identical to Cases 2 and 3, respectively.

In each of these base and sensitivity cases, a significant fraction of core material was retained in the vessel. As a result, Case 11a was run with FMAXCP = 0.6 to ensure 100% of the core was forced out of the vessel at vessel failure. With all of the corium on the floor of the CRD room and the coolability of the debris limited to 100 kw/m², Case 11a gives an upper limit to the amount of core concrete attack and hydrogen generation expected in the presence of an overlying pool of water.

Table 13.7-7 summarizes the results of the debris coolability sensitivity runs compared with the two base cases.

When the debris to water heat transfer coefficient is set to the MAAP-BRP default value of 900 ks (FCHF = 0.09), the debris in the CRD room valve pit sump is sufficiently cooled by the overlying water pool to prevent core concrete interaction. In the sensitivity Cases 11 and 12, where the coolability parameter is reduced to 100 kw/m², CCI erodes 1.5 feet of the concrete basemat, producing 63 pounds of hydrogen. In each of these four cases, approximately half of the core material remains in-vessel long term. Case 11a forced this residual debris out of the vessel and into the CRD room sump. It was found that some of the additional debris flowed out of the CRD sump and into the enclosure room through junction 9, which had failed earlier on debris contact. This allowed approximately 1.5 feet of concrete attack in both the CRD and enclosure sumps, thereby doubling the amount of hydrogen produced to 124 pounds.

In all the cases considered, the CsI release fraction was remarkably stable, varying between 6.32E-4 and 7.98E-4. It is therefore judged that the BRP containment is insensitive to the choice of FCHF.

These sensitivity studies show that the magnitude of the source term is independent of whether or not the containment sprays switch to recirculation mode. In cases where sprays switch to recirculation mode, containment pressures increase to approximately 40 psia at 36 hours after vessel failure. This value poses little threat to containment integrity and therefore the BRP containment can be judged insensitive to the use of sprays in recirculation mode. In cases where recirculation mode is not implemented, containment conditions are more benign with lower temperatures and pressures prevailing 36 hours after vessel failure. These scenarios will be examined further in Section 13.7.3.6 (Containment Flooding Sensitivities).

<u>Debris Cooling in the Sumps</u>

When vessel failure occurs, molten debris drops onto the CRD room floor and flows into the 2.25 foot deep CRD room valve pit. The sump is large enough to hold the entire core to a depth of 1.6 feet (Section 12.4.2.2). The MAAP-BRP model connects the CRD room to the enclosure room through several junctions (one of which is junction 9) originally representing a 2 inch line that runs horizontally between the nodes and lies 1.25 feet above the bottom of the CRD room sump. This line emerges 2 feet above the floor in the enclosure room, providing a convenient path for melted corium to flow between the two sumps. When implementing the MAAP-BRP code change to represent this junction, it was decided that the probability of corium freezing and plugging the passageway was sufficiently high enough for the junction to be considered a non-mechanistic pathway that would allow transport of debris between sumps in the event of a structural failure. Consequently, this line is assumed closed at the start of the transient, but will fail either on pressure differential at vessel failure or on temperature conditions after reaching the melting point of steel. Once opened, it can allow excess water in the enclosure room to drain into the CRD room, cooling the corium, or it may allow excess corium to transport to the adjacent node when the corium reaches its melting point and is at a level above the junction elevation.

Table 13.7-8 compares three low pressure cases that fail the vessel without the benefit of containment sprays. Cases 1 and 4 are typical of most MAAP-BRP cases in that approximately 50% of the core material remains in the vessel long term. In these cases, debris fills the CRD valve pit sump to a depth of 0.8 feet, somewhat below the opening to junction 9. Consequently, there can be no corium flow into the enclosure room unless more core material is forced from the vessel.

In Case 1, the low pressure failure base case, water remains on the floor of the CRD room for 8 hours after vessel failure. The containment pressurizes to 39 psia with the debris temperature peaking at 1600°F. This is below the melting point of steel and the initiation point for CCI, so junction 9 remains intact, precluding water flow into the CRD room from the enclosure.

In Case 4, the forced early containment failure reduces containment pressure to atmospheric levels. This reduces the effectiveness of the heat transfer from the debris to the CRD room atmosphere, causing the corium temperature to increase until junction 9 fails, allowing water from the enclosure to flow onto the debris. The corium temperature falls temporarily, then reheats, achieving a maximum temperature of just under 3200°F.

Case 1a was run with FMAXCP = 0.6 to ensure that 100% of the core material was forced out of the vessel at vessel failure. The volume of corium falling into the CRD room fills the sump to a level above the junction 9 elevation. When the junction fails on debris contact, approximately 33% of the core debris flows into the enclosure room. The debris remaining in the CRD room cools from 3600°F to

1800°F over the next 24 hours and gives rise to a negligible amount of basemat erosion.

In all three cases, the debris temperature in the CRD room peaks within 15 hours of vessel failure. In Cases 1 and 1a, where the containment remains intact, only small amounts of basemat penetration is observed. In early failure cases, where the CRD room pressure is low and debris temperatures are higher, 1.5 feet of concrete attack occurs during the first 36 hours after vessel failure.

Concrete Attack

Core concrete attack is a potential contributing failure mode to containment. In the absence of an overlying pool of water, molten core debris that remains long term in the CRD room valve pit sump will heat up and attack the concrete basemat. Similarly, in a high pressure vessel failure, debris entrained in the steam flow from the vessel can be transported to the enclosure sump and attack the enclosure floor. If a substantial fraction of the debris remains in one of the sumps long term, concrete erosion through the 7 foot thick basemat could eventually endanger containment integrity.

To investigate this possibility, MAAP-BRP cases were selected in which containment integrity was preserved after vessel failure and normal leakage was the mechanism for fission product release. In leakage cases with containment sprays operating and normal debris to water heat transfer rates, the debris was coolable and no CCI occurred. In most "dry" cases, the MAAP-BRP core melt progression model predicted that approximately 50% of the core material would exit the vessel. In such cases, with normal leakage operating, the pressurized containment (with its associated natural convection gas flows) proves sufficient to cool the limited amount of debris in the CRD room. In Case 1, with 48% of the core transported to the CRD sump, the debris proved coolable and CCI did not initiate.

If all the material was forced out of the vessel, as in Case 1a, the failure of junction 9 allowed approximately 30% of the corium flow into the enclosure, leaving 70% of the core material in the CRD room sump. In these cases, debris temperature increased sufficiently in the CRD room for CCI to initiate, but only 0.165 feet of concrete had eroded. In Case 1c, junction 9 was kept closed to prevent corium from flowing to the enclosure room so that 100% of the core material was retained in the CRD sump with the result that 2.15 feet of concrete erosion took place. This case gives an upper limit to concrete attack depth in a dry scenario. The total hydrogen generated was 170 pounds with the containment pressure 40.5 psia at 36 hours after vessel failure; there is little likelihood that concrete attack will fail containment due to overpressurization.

In the sensitivity cases where the debris to water heat transfer coefficient was limited to 100 kw/m², significant amounts of concrete erosion occurred. In Case 11a, all core material is forced out of the vessel and into the CRD room sump. With the limited cooling effect of the containment sprays, the connecting junction with the enclosure room fails 1.1 hour later, with the corium above the junction relocating after an additional 1.5 hours. In this case, the concrete erosion occurs in both sumps, but does not exceed 1.6 feet and finishes within 24 hours of vessel failure.

It is evident from the results summarized in Table 13.7-9 that concrete attack depths in "dry" cases, or in cases with very limited debris to water heat transfer coefficients, would be unlikely to lead to basemat failure in any reasonable length of time.

Containment Debris Spreading

The MAAP-BRP model assumes that all material that exits the vessel will either deposit in the CRD room sump or be entrained in the steam flow to the enclosure room where it will deposit in the sump. Consequently, normal BWR sensitivity studies which vary the drywell floor area for debris spreading have no meaning for the BRP model. Instead, it was decided to examine whether the accident progression was significantly influenced by the MAAP-BRP debris entrainment calculation. Therefore, high pressure failure Cases 14 and 15 were rerun as Cases 14b and 15b with entrainment turned off. Table 13.7-10 shows the key event timings and source terms associated with these runs.

In each of the base and sensitivity cases, 74% of the core material exits the vessel at vessel failure. With entrainment turned off, in Cases 14b and 15b, the debris falls into the CRD room sump. Some time later, junction 9 fails open and the corium (which has now dried out the CRD room compartment) melts and flows through the junction to the enclosure room sump. With entrainment turned on, in Cases 14 and 15, debris is transported to the enclosure room at vessel failure, with junction 9 allowing melted corium to flow back to the CRD room some time later. In each case, the sump that receives the initial deposit of debris retains more than 50% of the original core inventory and experiences concrete attack to depths up to 1.37 feet. The sump that takes the spill-over through junction 9 receives a small enough fraction of corium that little or no CCI ensues.

In conclusion, the release fractions calculated by MAAP-BRP are sensitive only to whether an early containment failure is forced. The containment response is insensitive to whether entrainment carries debris to the enclosure or lets it fall into the CRD room sump.

13.7.3.5 Containment Failure Mode (Size, Location and Type)

Containment Failure Area

A sensitivity case was run to investigate the impact of a medium (3.14 ft^2) containment failure versus a large containment failure (10.0 ft^2). Table 13.7-11 summarizes the containment failure size sensitivity cases. In general, increasing the assumed break area will slightly increase the calculated fission product release. It would be expected that cases dominated by late primary system revaporization will probably be less affected by the assumed break failure size. In that case, the revaporization rate will dictate the source term rather than the rate of efflux from containment.

Containment Failure Location

MAAP-BRP sequences assume that forced containment failures take place between the sphere and the environment, with Class V sequences the only exception. Accident Class V sequences are modeled as 0.2 ft^2 LOCA into the turbine building. The Class V sequence shown in Table 2.4-2 (Total class contribution to CDF of 1E-7/yr), had a high CsI release fraction of 0.546 (Table 12.7.1-1). The turbine building, which was modeled as a single node, retained 0.105 of the CsI inventory, giving it a DF of 1.19.

Containment Failure Type

Three types of failure or release mechanisms are modeled in the MAAP-BRP analysis: normal leakage, containment isolation failures, and early failures coincident with vessel failure. Normal containment leakage is assumed to be 0.5% per day and gives rise to source terms in the low-low to negligible range. Isolation failures are modeled as 3.14 ft^2 openings from the sphere to the environment, initiated at the beginning of the transient. Isolation failure releases generally fall in the low to moderate range, depending on whether or not containment sprays are available. Early failure cases assume that the pressure spike associated with vessel failure opens a 3.14 ft^2 junction with the environment. Early failure cases give rise to CsI release fractions between 0.03 and 0.19, within the moderate range. Table 13.7-12 characterizes the release magnitude associated with each failure type, given the availability of containment sprays.

13.7.3.6 Containment Flooding Sensitivity

The MAAP-BRP analysis shows that use of containment sprays can reduce the CsI release fraction significantly by scrubbing fission product aerosols from the containment atmosphere. Several analyses (with containment sprays) were run as "fill the ball" cases with sprays always supplied from an external source, rather than switching to recirculation mode when water reaches the 8 foot level in the enclosure room.

Table 13.7-13 contrasts the sphere temperatures and pressures for the two modes of containment spray usage. In MAAP-BRP sequences with RDS, the magnitude of the source term is insensitive to whether or not sprays switch to recirculation mode. However, a modest reduction in source term can be seen in Case 3b, where an increased spray flow rate is used. In general, containment conditions are more benign with lower temperatures and pressures long term in the "fill the ball" cases. The BRP containment volume is sufficiently large that many more hours of spray operation are required before containment pressure in "fill the ball" cases achieves levels equivalent to the recirculation case at 36 hours after vessel failure.

13.8 Core Damage Model Conservatisms

In the construction of the system fault tree models, the level of detail included leads to the potential double counting of component failures. For instance, the failure rate of the diesel fire pump failing to start or failing to run was calculated from the collected plant data. Minimal sorting of the raw data was performed to assign specific failure to the diesel fire pump sub components. All the failures were attributed to the pump failing to run. However, included in the fault tree for the diesel fire pump are the relays, batteries, fuel supply etc., with their appropriate generic failure rates. Since the plant specific value incorporates all the components required to start the engine and keep it running, including the specific components in the fault tree, this adds extra events to the sequence cutsets.

A second source of double counting occurs in the evaluation of each event tree. The code package which was used (IRRAS) does not allow for the combination of cutsets from an event tree into a single equation. This allows cutsets that would be subsumed, if the cutsets were combined, to be included in the total CDF.

No repair or recovery of equipment was included in the baseline model. The only exception for recovery was in the station blackout event tree. The recovery of off-site power was included in the event tree as top events. The restoration of the main condenser (after MSIV closure) was not credited, nor was the restoration of feedwater in the loss of feedwater event tree. In addition no recovery was included in the Level II analysis.

13.9 Other Sensitivity Studies

Based on the results of the system importance shown in Figure 13.1-3, sensitivities were performed for system combinations and their impact on core damage.

System combinations were based on similar functions (primary system makeup for instance). In addition to the system groupings, several other combinations of components were examined (common fuel oil supply for the different diesel engines used on site). These are discussed below.

13.9.1 Sensitivity to Diesel Fuel Oil Supply

There are four diesel motors at the plant site that may be utilized during any given initiating event:

- Diesel Driven Fire Pump - This is one of two fire protection system pumps at BRP. The DFP and the emergency diesel generator are located in the screen house structure (in completely separate rooms). The DFP has a separate fuel oil tank from the EDG.
- Emergency Diesel Generator - This is one of two equal capacity diesel generators on the plant site. This unit will automatically start and load when conditions exists indicative of a loss of station power or loss of normal power feed to MCC-2B.
- Standby Diesel Generator - Physically separated from the EDG is the standby diesel generator. This unit is trailer mounted and must be manually started and loaded onto MCC-2B during the conditions discussed above.
- Portable Pump - The portable pump is use primarily to supply emergency cooling to the emergency condenser in the event both the demin water system and the fire protection system are unavailable. The pump is stored within the alternate shutdown building and must be transported to the screenhouse area where it is piped to a dry hydrant. The pump will draft water from the discharge canal and can be discharged into a number of fire hydrant locations. The motor for the portable pump requires diesel fuel for operation.

The portable fire pump is the only motor that does not have a dedicated fuel supply. However, the existing fuel in the tanks would be used for pump operation. Under normal conditions, only the fuel tank that is low is refilled. However, there is currently no requirement prohibiting the refilling of all the fuel tanks with the same batch of fuel. This is the basis for this sensitivity.

The pump and generator failure to run values were all set equal to 1.0 for this study.

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{DG} = 2.3E-3$$

$$\Delta CDF = 2.25E-3$$

This large change in CDF is expected due to the impact the diesel engines have on plant systems. The diesel fire pump has a pervasive effect on a number of accident sequences, in that it can be used to maintain the emergency condenser as a heat sink during transients as well as makeup to the reactor during LOCAs. Without the diesel fire pump or generators, low pressure makeup can only be supplied with the electric fire pump or condensate pump, provided an off-site power source is available. Similarly, makeup to the emergency condenser is dependent on the demin water pump and off-site power.

The chemistry department currently collects samples from the different fuel oil tanks for off-site analysis as directed in surveillance procedure T90-10, Diesel Generators and Fire Pump Diesel Oil Storage Tank Sampling and Analysis. This provides assurance of diesel fuel quality and limits the potential for common fuel supply contribution to risk.

13.9.2 Sensitivity to the Load Rejection Assumption

For the loss of power transients, an assumption regarding the plant response to a loss of the 138 kv transmission line was made. The baseline model assumes that the plant will properly respond to a load rejection 90% of the time. A proper response is defined as: the bypass valve successfully opens and controls primary system pressure. The main condenser continues to operate to remove the excess power generation of the reactor, the turbine successfully reduces load to match station power requirements, the selected recirc pump trips reducing core power production, the feedwater system continues to operate and the operators successfully reduce power to house load by the control insertion of control rods.

While the plant was designed for a full load rejection, successful demonstration of this capability has only occurred at partial reactor power. This analysis addresses the sensitivity of the PRA results to the assumptions regarding the reliability of load rejection.

The event tree heading, LR (Load Rejection), in the loss of ac power event tree, was assigned a value of 1.0 and the core damage value was recalculated:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{LR} = 6.0E-5$$

$$\Delta CDF = 6.0E-6$$

This small change in the overall core damage is due to the availability of the emergency condenser to provide for decay heat removal and inventory control independent of ac power. The Big Rock Point Plant has the capability to cope with a station blackout for over a week.

13.9.3 Sensitivity to Hotwell Makeup

A sensitivity calculation was performed for the ability to use the condensate system as a low pressure water supply to the reactor. The condenser hotwell can be supplied with water from two different sources: 1) the normal, automatic gravity transfer of water from the condensate storage tank or 2) the fire protection system and the fire pumps. To perform this sensitivity analysis, the value used to quantify the cutsets for CV-4009 (condenser fill valve) and MO-7073 (fire water makeup to the hotwell) were altered from their random values to 1.0. The CDF equation was then re-quantified for a new CDF value. The modified CDF is:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{CD\text{-}MAKEUP} = 1.3E-4$$

$$\Delta CDF = 7.6E-5$$

This sensitivity analysis reduces the inventory supply capability to the design basis low pressure makeup system (the core sprays with fire water). This change in CDF by a factor of two, indicates the assumption which credits the condensate pumps with providing low pressure makeup is of, at least, marginal importance (with respect to CDF).

If the failure rates for CV-4009, MO-7073 and MO-7074 is increase by an order of magnitude, the following CDF is determined:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{CD\text{-}MAKEUP2} = 6.0E-5$$

$$\Delta CDF = 6.0E-6$$

Performance criteria for the reliability of these components can be relatively flexible and still allow the systems to provide a reliable source of makeup as back-up to the ECCS.

13.9.4 Sensitivity to Electrical Bus Failure Rate

The importance measures for the various electrical distribution panels and buses (with a Fussell-Vesely value greater than 1E-5) are shown in Table 13.9-1, below. None of the electrical panels contribute more than 1% to the base CDF value (all Fussell-Vesely values are less than 1E-2).

TABLE 13.9-1 ELECTRICAL DISTRIBUTION PANEL IMPORTANCES			
Event	Discription	Fussell-Vesely	Birnbaum
SP-BS-PNL1Y-SCLG	120 vac PANEL 1Y	1.7E-3	7.5E-3
SP-BS-PNL3Y-SCLG	120 vac PANEL 3Y	1.7E-3	7.5E-3
SPS-I-6	2400 vac BUS	5.8E-3	7.0E-3
SPS-I-11	480 vac MCC-2B	2.6E-4	5.7E-4
SPS-I-9	480 vac MCC-1A	5.0E-5	1.0E-4
SPS-I-1	125 vac PANEL D12 (ASD)	4.5E-5	1.0E-4
SPS-I-28	480 vac MCC-2A	4.0E-5	9.4E-5
SPS-I-23	480 vac PANEL 2P	1.2E-5	1.4E-5

Below is a brief description of each of the buses listed in Table 13.9-1 and the equipment affected by its failure.

SP-BS-PNL1Y-SCLG or SP-BS-PNL3Y-SCLG

These two 120 vac power panels distribute the majority of the power for the instrument and control functions outside containment, including the Control Room. Panel 3Y is fed from panel 1Y without an isolation breaker between them; therefore, a fault in 3Y was assumed to fail 1Y as well. Some of the equipment powered from these panels are:

Condensate Valve Control Scheme

Included in this circuit are valves that control the flow of the condensate. The important valves are:

- CV-4106 - Steam warming line for the turbine bypass valve fails closed on loss of power to solenoid; failure of one of two potential steam paths to the main condenser.
- CV-4104 - Steam to turbine steam seal regulator and air ejectors fails closed on loss of power to solenoid resulting a failure of the main condenser.
- CV-4009 - Condensate transfer line from condensate storage tank to condenser hotwell; failure of long term operation of condensate feedwater where makeup is required.

Manual Pressure Control

Without power to the RDS control cabinets (powered by panel 1Y) the operator cannot perform manual blowdown using the RDS isolation valve bypass line (CV-4184), nor can the operator perform a four train manual blowdown from inside the Control Room. These actions are important for the loss of instrument air event tree (about 9% of CDF).

Post Incident System

The containment water level instrumentation (used by the operator) is powered by panel 1Y. Without proper level instrumentation, the operators are assumed not to initiate the PIS.

These power panels each have a Birnbaum of 7.5E-3, which is a significant increase in CDF should they be assumed failed.

2400 Volt Bus

The 2400 volt bus is used to distribute the power from the station power transformer number 1 or number 7. Without this bus, the plant is essentially in a loss of off-site power transient and reliance is placed on the diesel generators for on-site ac power requirements. Directly powered from the 2400 volt bus are the two reactor recirc pumps and the two reactor feedwater pumps. The two condenser circulating water pumps and the two condensate pumps, which cannot be powered by the either diesel generator, are powered from 480 volt bus 1 and 2. Without the 2400 volt bus, one heat sink is lost (main condenser). The only source of high pressure-high volume makeup to the primary system is disabled (feedwater) as is a backup source of low pressure high volume makeup for the core sprays (condensate). This event has a Birnbaum value of 7E-3, which is a significant increase in CDF should the bus be assumed failed.

MCC-2B (SPS-I-11)

MCC-2B is used to distribute the power from either emergency diesel generator to the 480 volt bus network, or it distributes off-site power to the emergency loads connected directly to the bus. Loads connected directly to MCC-2B include the electric fire pump and both of the redundant core spray valves (MO-7070 and MO-7071). In addition, through manual closure of tie breakers 1A-2B and 2A-2B, the operators can power loads which are essential to plant shutdown during a loss of station power. As previously mentioned, all 480 volt and smaller loads (except for the condensate and condenser circulating pumps) may be powered by the operating diesel. The Birnbaum value for this event is 5.7E-4, which is a significant increase in core damage if this bus is assumed failed.

MCC-1A (SPS-1-9)

MCC-1A is a 480 vac motor control center that is powered by Bus 1, from station power transformer number 11. Some of the important loads that receive power from MCC-1A are: MO-7073 and MO-7074 (fire water to the hotwell), air compressors numbers 1 and 2, core spray pump number 1, and BUS-1C loads, service water pump number 1, screen drive number 1 and the screen wash pump.

Of these loads, the most significant are the screen wash pump and the fire water to hotwell motor operated valves. The screen wash pump is significant in that failure to clean plugged traveling screens results in a loss of both fire pumps, both service water pumps and both condenser circulating water pumps. This event has a Birnbaum value of 1E-4, which is a significant increase in CDF if the MCC is assumed failed.

125 vdc Panel D12 (Alternate Shutdown Panel)

This panel distributes 125 vdc power to the main steam isolation valve (MO-7050), the emergency condenser outlet valves (MO-7053 and MO-7063) and the fire water makeup valve (SV-4947) to the emergency condenser shell. In the loss of instrument air event tree (about 9% of CDF) the main condenser is assumed failed; the emergency condenser becomes the primary heat sink with backup by the RDS\CS combination. Without power from panel D12, the emergency condenser fails. The Birnbaum for this event is 1E-4, which is a significant increase in CDF if this panel is assumed failed.

MCC-2A (SPS-1-28)

MCC-2A is a 480 vac motor control center that is powered by Bus 2, from station power transformer number 22. Some of the important loads that receive power from MCC-2A are: air compressor number 3, core spray pump number 2, fire water to post incident heat exchanger (MO-7066), and the normal power feed for MCC-2B. Failure of MCC-2B causes the emergency diesel generator to automatically start and load to MCC-2B. The most important of these loads is post incident pump number 2. Without MCC-2A the post incident system is reduced to the operation of one pump and one valve. The Birnbaum for this event is 9.4E-5, which is nearly double the baseline CDF.

Panel 2P (SPS-I-23)

Panel 2P powers the 480 volt equipment inside containment, which includes: the two shutdown cooling pumps, four shutdown cooling motor operated valves, reactor cooling water pumps, the emergency condenser inlet valves, and the poison tank heaters. This event only appears in sequences derived from the ATWS event trees. The poison tank heaters were conservatively assumed to be required for proper system actuation. In reality, the heaters are required prior to the event in order to maintain the poison in solution. The time frame in which the poison system is required is sufficiently small that solution cooling (causing precipitation) would occur. The Birnbaum value of 1.4E-5 for this event indicates that it is not a significant event, even if assumed failed.

13.9.5 Sensitivity to Temperature In Rooms 418 and 400

A sensitivity was performed to asses the change in CDF if it was assumed that the temperature in rooms 418 (spent fuel pit heat exchanger) and 400 (steam drum enclosure) had a detrimental effect on equipment performance. The LOCA and steam line break event trees were re-analyzed assuming increased failure probabilities of equipment located within the two rooms. The equipment of interest in these rooms are the core spray level and pressure instrumentation used for automatic valve operation (room 418), reactor water level transmitters for the low reactor water level permissive signal for RDS (room 418), the primary core spray motor operated valves, MO-7051 and MO-7061 (room 400) and the emergency condenser outlet valves, MO-7053 and MO-7063 (room 400). Two sensitivities were performed:

- An assumed moderate degradation of equipment performance that effectively doubles the base failure rate.
- An assumed severe degradation of performance by an order of magnitude greater than the base failure rate.

The moderate degradation of equipment performance is presented below:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{2x} = 5.62E-5$$

$$\Delta CDF_{2x} = 2.9E-6$$

This small change in CDF indicates that some degradation in equipment performance can be tolerated without a significant impact on core damage. If a larger decrease in performance is assumed, the CDF is:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{10x} = 8.22E-5$$

$$\Delta CDF_{10x} = 3.8E-5$$

This change in CDF is still less than a factor of two from the case CDF, which indicates that, even if performance were to degrade by an order of magnitude, the change core damage frequency is not significantly affected.

13.9.6 Liquid Poison Squib Valve Sensitivity

Sensitivity calculations on the base CDF were performed on the performance of the squib valves which are part of the liquid poison system. The seven squib valves are explosive actuated valves that have replaceable primer and trigger assemblies. The primer is subject to thermal degradation while in service. Periodic testing of a primer and trigger assembly assures: 1) proper operation of the remaining assemblies, and 2) a fixed periodic replacement of the in-service assemblies. Two sensitivities were performed on the squib valves. The first sensitivity assumes the squib valves fail to function when exposed to a high temperature steam environment, as would be expected during safety relief valve operation during an ATWS event. The second sensitivity assumes that the poison valves will automatically actuate when exposed to prolonged (greater than 10 minutes) high temperatures. The results are discussed below.

Case 1 Squib Valve Failure

This sensitivity assumes that for all high pressure ATWS sequences (successful SRV opening), the squib valves fail to detonate. The change in core damage frequency is shown below:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{EV=1} = 5.6E-5$$

$$\Delta CDF_{EV=1} = 2.0E-6$$

Case 2 Auto-Actuation of Squib Valves

This sensitivity assumes that, for all high pressure ATWS sequences, the squib valves auto-detonate due to the elevated temperatures inside containment. This effectively removes the operator action from injecting poison during a high pressure sequence. The change in core damage frequency is shown below:

$$CDF_{BASE} = 5.4E-5 \qquad CDF_{AUTO} = 5.34E-5$$

$$\Delta CDF_{AUTO} = -6.0E-7$$

The small changes in core damage frequencies (reported above) for the two different cases result from the ATWS contribution to core damage being completely dominated by the low pressure turbine trip sequences. Total ATWS contribution to core damage is about 7% and the contribution to core damage from the turbine trip ATWS event tree is 6.4%. Since the squib valve failure/auto-detonation was assumed to occur only for sequences in which the SRVs are open, the dominant sequences are unaffected.

REFERENCES

13-1 J. U. Valente and J. W. Yang, "MAAP3.0B Code Evaluation Final Report," Brookhaven National Laboratory, FIN L-1499, October 1992.

13-2 M. A. Kenton and J. R. Gabor, "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP3.0B," EPRI TR-100107 to be published.

TABLE 13.1-1 FUSSELL VESELY IMPORTANCE MEASURES		
EVENT NAME	PROB	F-V
PH-IN-HEATR-FAIL	1.00E+00	4.20E-01
S-1	1.70E-03	2.54E-01
S-2	1.70E-03	1.96E-01
S-3	1.30E-03	1.19E-01
EC-MV-VSSLB-POIC	1.00E+00	1.08E-01
PI-PM-P2-CCFTS	8.79E-04	1.07E-01
RD-OO-PCNTL-POIC	1.50E-01	1.01E-01
T-IA	4.50E-02	9.07E-02
FP-PM-P6-FTR	8.52E-03	8.74E-02
S-5	8.80E-04	8.68E-02
CDS-BREAKS	6.60E-01	7.75E-02
CS-MV-BD-CCFTO	5.50E-04	7.68E-02
CS-MV-BC-CCFTO	5.50E-04	7.68E-02
CS-MV-AD-CCFTO	5.50E-04	7.68E-02
CS-MV-AC-CCFTO	5.50E-04	7.68E-02
FP-PM-P7-FTR	1.15E-02	7.27E-02
CD-HS-P9TRP-POIC	8.00E-02	7.23E-02
IR	1.00E+00	6.81E-02
RC-COND	1.00E+00	6.75E-02
LI-OO-INJ2-POIC	3.00E-01	6.72E-02
A-TT	1.10E-05	6.20E-02
CS-MV-7071-FTO	2.56E-02	5.38E-02
RP-RX-VSSLB-POIC	6.00E-04	4.80E-02
S-6	3.30E-04	4.68E-02
PI-CV-PI300-FTO	2.70E-04	3.28E-02
PC-RV-SDSRV-FTC	4.88E-01	3.07E-02
CS-MV-7061-FTO	6.61E-03	2.97E-02
CS-MV-7051-FTO	6.61E-03	2.97E-02
RDS-I-12	3.81E-02	2.91E-02
RDS-I-10	3.76E-02	2.84E-02
S-4	3.00E-04	2.74E-02
CDS-COND	1.00E+00	2.72E-02
S-8	1.70E-03	2.51E-02
PI-XV-PI004-PROC	2.00E-04	2.43E-02
FP-PM-P6-OOS	2.60E-03	2.29E-02

TABLE 13.1-1 FUSSELL VESELY IMPORTANCE MEASURES		
EVENT NAME	PROB	F-V
EM-KV-4947-FTE	5.73E-03	2.19E-02
S-9	1.70E-03	1.99E-02
RDS-I-16	1.93E-02	1.87E-02
RDS-I-14	1.77E-02	1.66E-02
FP-PM-P7-FTS	2.39E-03	1.47E-02
FP-OO-MAKUP-POIC	1.60E-02	1.45E-02
S-10	1.30E-03	1.44E-02
T-MS	5.60E+00	1.41E-02
CD-HS-7039-PROC	3.00E-02	1.40E-02
EM-XV-EC112-PROC	3.00E-03	1.34E-02
LR	1.00E-01	1.30E-02
138	2.90E-01	1.30E-02
SPS-I-7	2.04E-01	1.28E-02
RDS-I-15	1.33E-02	1.27E-02
FP-PM-P7-OOS	2.49E-03	1.25E-02
SPS-I-29	1.30E-01	1.25E-02
SP-OO-LOSP-FAIL	2.00E-01	1.23E-02
CS-MV-7070-FTO	5.80E-03	1.21E-02
CAS-I-1	1.24E-03	1.21E-02
FP-RV-5062-FTC	1.99E-03	1.18E-02
FP-PM-P6-FTS	1.15E-03	1.11E-02
PI-PM-P2-CCFTR	8.98E-05	1.09E-02
PI-OO-TSD07-PROC	8.50E-05	1.03E-02
S-7	7.70E-05	1.01E-02
RDS-I-9	1.40E-02	1.01E-02
PI-OO-PISYS-POIC	8.30E-05	1.01E-02

TABLE 13.1-2 BIRNBAUM IMPORTANCE MEASURES		
EVENT NAME	PROB	BIRNBAUM
A-FW	5.00E-07	3.02E-01
A-TT	1.10E-05	3.02E-01
A-BYP	4.90E-07	3.01E-01
T-ILCS	1.20E-07	6.48E-02
A-LOSP	6.10E-07	3.04E-02
FP-CV-VLV-CCFTO	2.70E-06	2.10E-02
FP-XV-FP013-PLUG	7.20E-08	2.07E-02
A-IA	4.50E-07	1.52E-02
A-MC	4.50E-07	1.34E-02
A-MSIV	1.80E-07	1.34E-02
SW-PP-001-LEAK	2.06E-08	9.72E-03
S-1	1.70E-03	7.98E-03
S-6	3.30E-04	7.57E-03
SP-BS-PNL1Y-SCLG	1.20E-05	7.48E-03
SP-BS-PNL3Y-SCLG	1.20E-05	7.48E-03
CS-MV-AC-CCFTO	5.50E-04	7.47E-03
CS-MV-AD-CCFTO	5.50E-04	7.47E-03
CS-MV-BC-CCFTO	5.50E-04	7.47E-03
CS-MV-BD-CCFTO	5.50E-04	7.47E-03
CS-MV-ABCD-CCFTO	4.95E-05	7.40E-03
CS-CV-3034-CCFTO	2.70E-06	7.32E-03
SPS-I-6	4.39E-05	7.05E-03
S-7	7.70E-05	6.98E-03
PI-XV-PI004-PROC	2.00E-04	6.51E-03
PI-OO-TSD07-PROC	8.50E-05	6.51E-03
PI-PM-P2-CCFTS	8.79E-04	6.51E-03
PI-CV-PI300-FTO	2.70E-04	6.49E-03
PI-PM-P2-CCFTR	8.98E-05	6.48E-03
PI-OO-PISYS-POIC	8.30E-05	6.48E-03
PI-CV-3067-CCFTO	2.70E-06	6.42E-03
PI-HE-S006-PLUG	3.10E-06	6.42E-03
CS-CV-3012-CCFTO	2.70E-06	6.18E-03
S-2	1.70E-03	6.16E-03
FP-PP-004-LEAK	2.06E-08	5.82E-03
PI-HE-006-PLUG	2.06E-07	5.78E-03

TABLE 13.1-2 BIRNBAUM IMPORTANCE MEASURES		
EVENT NAME	PROB	BIRNBAUM
S-5	8.80E-04	5.27E-03
S-3	1.30E-03	4.89E-03
S-4	3.00E-04	4.89E-03
RP-LE-IC-CCCAL	1.04E-05	4.87E-03
RD-TL-456-CCFAIL	9.26E-06	4.87E-03
RD-TL-457-CCFAIL	9.26E-06	4.87E-03
RD-TL-467-CCFAIL	9.26E-06	4.87E-03
RD-TL-567-CCFAIL	9.26E-06	4.87E-03
RD-TL-4567-CFAIL	7.10E-06	4.87E-03
RD-TL-IC55-CCCAL	3.74E-06	4.84E-03
RP-RX-VSSLB-POIC	6.00E-04	4.28E-03
CS-PS-IC-CCCAL	3.74E-06	3.33E-03
CS-LS-IC-CCCAL	2.74E-06	3.32E-03
RD-RV-4567-CCFTO	4.46E-05	3.06E-03
RD-KV-4567-CCFTE	2.78E-05	3.05E-03
RD-KV-467-CCFTE	4.10E-05	2.64E-03
RD-KV-567-CCFTE	4.10E-05	2.64E-03
RD-RV-467-CCFTO	3.82E-05	2.64E-03
RD-RV-567-CCFTO	3.82E-05	2.64E-03
RD-KV-457-CCFTE	4.10E-05	2.64E-03
RD-RV-457-CCFTO	3.82E-05	2.63E-03
RD-AV-013-CCFTO	2.81E-05	2.63E-03
RD-AV-023-CCFTO	2.81E-05	2.63E-03
RD-AV-123-CCFTO	2.81E-05	2.63E-03
RD-AV-0123-CCFTO	1.87E-05	2.62E-03
RD-TL-012-CCFAIL	1.10E-05	2.61E-03
RD-TL-013-CCFAIL	1.10E-05	2.61E-03
RD-TL-023-CCFAIL	1.10E-05	2.61E-03
RD-TL-123-CCFAIL	1.10E-05	2.61E-03
RD-TL-0123-CFAIL	8.47E-06	2.61E-03
RD-KV-013-CCFTE	6.10E-06	2.61E-03
RD-KV-023-CCFTE	6.10E-06	2.61E-03
RD-KV-123-CCFTE	6.10E-06	2.61E-03
RD-KV-0123-CCFTE	4.14E-06	2.61E-03
RD-TL-IC54-CCCAL	3.74E-06	2.61E-03

TABLE 13.1-2 BIRNBAUM IMPORTANCE MEASURES		
EVENT NAME	PROB	BIRNBAUM
T-ILBL	3.67E-06	1.43E-03
F-INT	1.40E-06	8.04E-04
S-8	1.70E-03	7.90E-04
S-11	3.00E-04	6.38E-04
DC-BS-MCC01-OPCI	1.20E-05	6.30E-04
DC-BS-MCC01-SCLG	1.20E-05	6.30E-04
S-9	1.70E-03	6.28E-04
DC-CK-7211-FTRC	6.48E-06	6.24E-04
S-10	1.30E-03	5.94E-04
SPS-i-11	2.40E-05	5.72E-04
FP-PM-P6-FTR	8.52E-03	5.49E-04
CAS-I-1	1.24E-03	5.22E-04
FP-PM-P6-FTS	1.15E-03	5.18E-04
DC-CK-S7211-FTRC	1.08E-06	4.99E-04
FP-CB-2B26-FTRC	1.76E-04	4.93E-04
FP-CV-8406-FTC	2.70E-04	4.90E-04
FP-CV-FP304-FTO	2.67E-04	4.89E-04
FP-CV-8405-FTRC	1.30E-05	4.75E-04
FP-PM-P6-OOS	2.60E-03	4.71E-04
FP-CB-PC17-FTRC	1.99E-05	4.70E-04
FP-CB-2B26-OOS	8.60E-05	4.39E-04

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
Circulating Water (CW)	
Fussell-Vesely = 1.9E-6	**Birnbaum = 3.5E-6**
CW-MV-7054-OOS	MO-7054 Out of Service
CW-MV-7055-OOS	MO-7055 Out of Service
Intake Structure (INTAKE)	
Fussell-Vesely = 1.7E-4	**Birnbaum = 1.1**
CW-PS-IC-CCCAL	Traveling Screen PS-601&611 Common Calibration Faults
CW-TS-001-PLUG	Traveling Screen #1 Plugged
CW-TS-002-PLUG	Traveling Screen #2 Plugged
Primary Coolant SRVs fails to open (PC)	
Fussell-Vesely = 1.3E-11	**Birnbaum = 4.5E-9**
PC-RV-Q6-CCFTO	Common Cause Failure of All 6 Steam Drum Safety Relief Valves To Open
Post Incident (PI)	
Fussell-Vesely = 1.7E-1	**Birnbaum = 1.4E-1**
PI-CB-S1A44-FTRC	Breaker 1A-44 (P-2A) Stand-By Failure
PI-CB-S2A44-FTRC	Breaker 2A-44 (P-2B) Stand-By Failure
PI-CK-S1Y23-FTRC	Breaker 1Y-23 (Scheme 6503) Stand-by Failure
PI-CV-3067-CCFTO	P-2A&B Check Valves VPI-306 & 7 Common Cause Failure To Open
PI-CV-PI300-FTO	Check Valve VPI-300 Fails To Open
PI-CV-PI306-FTO	P-2A Check Valve VPI-306 Fails To Open
PI-CV-PI307-FTO	P-2B Check Valve VPI-307 Fails To Open
PI-HE-S006-PLUG	Core Spray Heat Exchanger Stand-By Failure

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
PI-HO-SLOCA-LEAK	LOCA Hose Stand-By Failure
PI-MV-6680-CCFTO	PI System Valves MO-7066&80 Common Cause Failure To Open
PI-PM-P2-CCFTR	CS Pumps P2A & B Common Cause Failure To Run
PI-PM-P2-CCFTS	CS Pumps P2A & B Common Cause Failure To Start
PI-PM-P2A-FTR	P-2A Fails To Run
PI-PM-P2A-FTS	P-2A Fails To Start
PI-PM-P2B-FTR	Pump P-2B Fails To Run
PI-PM-P2B-FTS	Pump P-2B Fails To Start
PI-RE-5415-FTE	P-2A Mot Cont Relay Scheme 5415 Fails To Energize
PI-RE-5416-FTE	P-2B Mot Cont Relay Sch 5416 Fails To Energize
PI-TL-3175-OOS	Level Transmitter LT-3175 Out of Service
PI-TL-S3171-FAIL	Level Transmitter LT-3171 Stand-By Failure
PI-TL-S3175-FAIL	Level Transmitter LT-3175 Stand-By Failure
Instrument Air (CA)	
Fussell-Vesely = 1.0E-2	**Birnbaum = 5.3E-4**
CA-CM-AB-CCFTR	Compressors M-49A and M-49B Common Cause Failure To Run
CA-CM-AB-CCFTS	Compressors M-49A and M-49B Common Cause Failure To Start
CA-CM-ABC-CCFTR	Compressors M-49A & M-49B & M-49C Common Cause Fail To Run
CA-CM-ABC-CCFTS	Compressors M-49A & M-49B & M-49C Common Cause Fail To Start
CA-CM-AC-CCFTS	Compressors M-49A and M-49C Common Cause Failure To Run
CA-CM-BC-CCFTS	Compressors M-49B and M-49C Common Cause Failure To Start

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
CA-CV-12-CCFTO	Compr 1 & 2 Disch Check Valve Common Cause Failure To Open
CA-CV-123-CCFTO	Compr 1 2 & 3 Disch Check Valve Common Cause Failure To Open
CA-CV-13-CCFTO	Compr 1 & 3 Disch Check Valve Common Cause Failure To Open
CA-CV-23-CCFTO	Compr 2 & 3 Disch Check Valve Common Cause Failure To Open
CA-CV-VA301-FTC	Air Compressor 1 Disch Check Valve VA-301 Fails To Close
CA-CV-VA302-FTC	Air Compressor 2 Disch Check Valve VA-302 Fails To Close
CA-CV-VA302-FTRC	VA-302 F-25B Disch Check Valve Fails To Remain Closed
CA-CV-VA303-FTC	Air Compressor 3 Disch Check Valve VA-303 Fails To Close
CA-CV-VA303-FTRC	VA-303 F-25C Disch Check Valve Fails To Remain Closed
CA-CV-VA304-FTO	Check Valve VA-304 Fails To Open
CA KV-34-CCFTE	Compr SV Valves SV-4913, 4 Common Cause Fails to Energize
CA-KV-345-CCFTE	Compr SV Valves SV-4913,4,5 Common Cause Failure To Energize
CA-KV-35-CCFTE	Compr SV Valves SV-4913,5 Common Cause Failure To Energize
CA-KV-45-CCFTE	Compr SV Valves SV-4914,5 Common Cause Failure To Energize
CA-PS-12-CCFTC	Compr 1 & 2 Press Switch Common Cause Failure To Close
CA-PS-123-CCFTC	Compr 1 2 & 3 Press Switch Common Cause Failure To Close
CA-PS-13-CCFTC	Compr 1 & 3 Press Switch Common Cause Failure To Close
CA-PS-23-CCFTC	Compr 2 & 3 Press Switch Common Cause Failure To Close
CA-PS-STRT-CCFTC	Compr 2 & 3 Start Press Switch Common Cause Failure To Close
CA-RE-3406C-FTE	M-49C Relay 42/HC Sch 3406 Fails To Energize

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
CAS-I-1	Common Air Compressor Failures Such As Heat Exchanger Failures and Relief Valve Failures
CAS-I-10	Circuit Breaker 1A-34 (HSAC#1) Failures
CAS-I-12	Circuit Breaker 1A-35 (HSAC#2) Failures
CAS-I-13	Pressure Switch PS-602 Faults
CAS-I-14	Pressure Switch PS-610 Faults
CAS-I-15	Pressure Switch PS-614 Faults
CAS-I-2	Train A Faults;Check Valve VA-302
CAS-I-3	Train B Faults;SV-4885 and Check Valve VA-302
CAS-I-4	Train C Faults; SV-4886 and Check Valve VA-303
CAS-I-5	HSAC#1 And SV-4913 Faults
CAS-I-6	Operator Fails To Place Handswitch In Standby and Failure To Start HSAC#2
CAS-I-7	HSAC#3 OOS And Operator Fails To Start HSAC
Condensate (CD)	
Fussell-Vesely = 3.8E-3	**Birnbaum = 3.8E-5**
CD-AV-4009-FTO	CV-4009 Fails To Open Hot Well Fill Line
CD-CB-5212-FTC	Breaker 52-12 (P-9A) Fails To Close
CD-CB-5222-FTC	Breaker 52-22 (P-9B) Fails To Close
CD-CB-PMP-CCFAIL	Breakers 52-12 & 22 Common Cause Failure (P9A And P9B)
CD-CK-1Y7-FTRC	Breaker 1Y-7 CDS Valve Controls Fails To Remain Closed
CD-CV-VC300-FTO	P-9A Discharge Check Valve Fails To Open
CD-CV-VC301-FTC	VC-301 Fails To Close P-9B Disch Check Valve
CD-CV-VC301-FTO	P-9B Discharge Check Valve Fails To Open

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
CD-CV-VLV-CCFTO	P-9A & B VC-300 & 301 Common Cause Failure To Open
CD-KV-4855-FTE	SV-4855 Fails To Energize For CV-4009
CD-PM-P9-CCFTS	Condensate P-9A and P-9B Common Cause Failure To Start
CD-PM-P9A-FTS	Condensate Pump P-9A Fails To Start
CD-PM-P9B-FTS	Condensate Pump P-9B Fails To Start
Control Rod Drive (CRD)	
Fussell-Vesely = 3.7E-3	**Birnbaum = 2.9E-4**
CR-AV-4090-FTO	CV-4090 Fails To Open
CR-AV-4090-FTRO	CV-4090 Fails To Remain Open
CR-AV-NC18-FTRO	CV-NC18 Fails To Remain Open
CR-CB-2A58-FTRC	Breaker 2A-58 (P-4B) Fails To Remain Closed
CR-CV-RD303-FTO	Check Valve VRD-303 Fails To Open
CR-CV-RD313-FTO	Check Valve VRD-313 Fails To Open
CR-FL-5827-PLUG	CRD Strainer YS-5827 Plugged
CR-FL-NC07-PLUG	CRD Strainer YS-NC07 Plugged
CR-FL-S23A-PLUG	Control Rod Drive Filter S23A Plugged
CR-FL-S5826-PLUG	YS-5826 Stand-By Failure
CR-FU-6412-OPCI	Fuse Sch 6412 Open Circuit
CR-KV-4894-FTD	SV-4894 Fails To De-Energize For CV-4090
CR-OY-6412-FAIL	Circuit Transformer Sch 6412 Fails
CR-PM-P4-CCFTR	CRD P-4A and P-4B Common Cause Failure To Run

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
CR-PM-P4A-FTR	CRD Pump P4A Fails To Run
CR-PM-P4A-FTS	CRD Pump P4A Fails To Start
CR-PM-P4A-OOS	CRD Pump P4A Out of Service
CR-PM-P4B-FTR	CRD Pump P4B Fails To Run
CR-PS-6262-FTO	PS-626-2 Fails To Open
CR-PV-NC18-FTRO	Pressure Controller PCV-NC18 Fails To Operate
CR-RV-5050-FTRC	RV-5050 Fails To Remain Closed
CR-RV-5051-FTRC	RV-5051 Fails To Remain Closed
Core Spray (CS)	
Fussell-Vesely = 4.0E-1	**Birnbaum = 1.7E-1**
CS-CV-3012-CCFTO	CS RPV Inject Vlvs VPI-301& 302 Common Cause Failure To Open
CS-CV-3034-CCFTO	CS RPV Inject Vlvs VPI-303 & 304 Common Cause Failure To Open
CS-CV-PI303-FTO	Back-Up CS Check Valve VPI303 Fails To Open
CS-CV-PI304-FTO	Primary CS Check Valve VPI304 Fails To Open
CS-FE-S2815-PLUG	Flow Element FE-2815 Stand-By Failure
CS-LS-IC-CCCAL	Level Switch LS-RE09 Common Calibration Errors
CS-MV-7051-FTO	MO-7051 Fails To Open
CS-MV-7051-OOS	MO-7051 Out of Service
CS-MV-7061-FTO	MO-7061 Fails To Open
CS-MV-7061-OOS	MO-7061 Out of Service
CS-MV-7070-FTO	MO-7070 Fails To Open

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
CS-MV-7071-FTO	MO-7071 Fails To Open
CS-MV-ABCD-CCFTO	CS RPV Inj MO-7051,61,70,71 CC Failure To Open
CS-MV-AC-CCFTO	CS RPV Inj MO-7051 & 70 CC Failure To Open
CS-MV-AD-CCFTO	CS RPV Inj MO-7051 & 71 CC Failure To Open
CS-MV-BC-CCFTO	CS RPV INJ MO-7061 & 70 CC Failure To Open
CS-MV-BD-CCFTO	CS RPV INJ MO-7061 & 71 CC Failure To Open
CS-PS-IC-CCCAL	Core Spray PS-IG11A-H Common Calibration Faults
CS-RE-163O-FTE	Relay 42/O Sch 163 Fails To Energize
CS-RE-5601O-FTE	Open Relay 42/O Sch 5601 Fails To Energize
CS-RE-5602O-FTE	Open Relay 42/O Sch 5602 Fails To Energize
CS-RE-B152O-FTE	Relay 42/O Sch B152 Fails To Energize
DC Power (DC)	
Fussell-Vesely = 1.1E-4	**Birnbaum = 9.4E-5**
DC-BS-BSD02-OPCI	Bus D02 Open Circuit
DC-BS-BSD02-SCLG	Bus D02 Short To Ground
DC-BS-MCC01-OPCI	125Vdc MCC-01 Open Circuit
DC-BS-MCC01-SCLG	125 Vdc MCC-01 Short To Ground
DC-BY-STBAT-FTE	Station Batteries Fail To Operate On Demand
DC-CK-1D40-FTRC	Breaker 1D-40 (480V Load Center) Fails To Remain Closed
DC-CK-7211-FTRC	Breaker 72-11 (125vdc MCC D01 Feed) Fails To Remain Closed
DC-CK-7212-FTRC	Breaker 72-12 (125vdc Panel D02 Feed) Fails To Remain Closed

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
Emergency Condenser Valve (EV)	
Fussell-Vesely = 2.9E-3	**Birnbaum = 4.8E-4**
EC-HE-ECSHL-PLUG	Emergency Condenser Shell Leaks/Ruptures
EC-MV-563-CCFTO	EC Valves MV-7053 & 63 Common Cause Failure To Open
EC-MV-7053-FTO	MO-7053 Fails To Open
EC-MV-7053-OOS	Emergency Condenser Loop 2 Out of Service
EC-MV-7063-FTO	MO-7063 Fails To Open
EC-MV-7063-OOS	Emergency Condenser Loop 1 Out of Service
EC-PS-7AC-CCFTO	PS-RE07A & C Common Cause Failure To Open
EC-PS-7BD-CCFTO	PS-RE07B & D Common Cause Failure To Open
EC-RE-6602B-FTE	Relay 42-O Sch 6602 MO-7053 Fails To Energize
EC-RE-6603B-FTE	Relay 42-O Sch 6603 MO-7063 Fails To Energize
Demin Water Makeup (DM)	
Fussell-Vesely = 4.7E-5	**Birnbaum = 2.2E-7**
EM-AV-4028-FTO	Control Valve CV-4028 Fails To Open
EM-AV-4028-FTRO	Control Valve CV-4028 Fails To Remain Open
EM-AV-4041-FTRC	CV-4041 Demin Water To CST Fails To Remain Closed
EM-AV-4105-FTRO	Air Operated (Diaphram) Vlv CV-4105 Fails To Remain Open
EM-AV-S4041-FTRC	CV-4041 Demin Water To CST Stand-By Failure
EM-CB-1E22-FTRC	Breaker 1E-22 (P-37) Fails To Remain Closed
EM-CB-S1E22-FTRC	Breaker 1E-22 (P-37) Stand-By Failure
EM-CK-2Y7-FTRC	Breaker 2Y-7 (Scheme 6505) Fails To Remain Closed

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
EM-CK-2Y8-FTRC	Breaker 2Y-8 (SV-4874 For CV-4028) Fails To Remain Closed
EM-CK-S2Y8-FTRC	Breaker 2Y-8 (SV-4874 For CV-4028) Stand-By Failure
EM-CV-EC300-FTO	Check Valve VEC-300 Fails To Open
EM-CV-MU300-FTO	VMU-300 Demin Water To Sphere Check Valve Fails To Open
EM-FU-5412-OPCI	P-37 Fuse Sch 5412 Open Circuit
EM-FU-S5412-OPCI	P-37 Fuse Sch 5412 Open Circuit Stand-By Failure
EM-HS-5412-FTRE	Lvl Bypass Switch HS-7045 Sch 5412 Fails To Remain Energized
EM-HS-5412A-FTRE	Hand Switch 42/CS Sch 5412 Fails To Remain In Run
EM-HS-7025-FTRE	Hand Switch HS-7025 Fails To Remain Energized
EM-HS-7902-FTE	HS-7902 Fails To Energize
EM-HS-S412A-FTRE	Hand Switch 42/CS SCH 5412 Stand-By Failure
EM-HS-S5412-FTRE	Level Bypass Switch HS-7045 Sch 5412 Stand-By Failure
UV-RELAY-1F-1E	Undervoltage Relays For Breaker 1F-1E Failure
EM-HS-S7025-FTRE	Hand Switch HS-7025 Stand-By Failure
EM-KV-4874-FTE	SV-4874 Fails To Energize For CV-4028
EM-KV-4874-FTRE	SV-4874 Fails To Remain Energized For CV-4028
EM-KV-4897-FTE	SV-4897 Fails To Energize For CV-4105
EM-KV-4897-FTRE	SV-4897 Fails To Remain Energized For CV-4105
EM-OO-DEMIN-OOS	Demin Water System Unavailable To Provide Water
EM-OO-T31-EMTY	Demin Water Tank T-31 Empty

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
EM-PM-P37-FTR	Demin Water Pump P-37 Fails To Run
EM-PM-P37-OOS	Demin Water Pump P-37 Out of Service
EM-RE-5412-FTE	Relay 42/HC Sch 5412 Fails To Energize
EM-RE-5412-FTRE	Relay 42/HC Sch 5412 Fails To Remain Energized
EM-TK-T31-PLUG	Demin Water Tank T-31 Failure
Emergency Condesner Makeup-Fire Protection (FM)	
Fussell-Vesely = 2.6E-2	**Birnbaum = 1.3E-3**
EM-KV-4947-FTE	SV-4947 Fails To Energize
EM-RE-TDR-FTRE	TDR Relay- Sch 6618 Fails To Remain Energized
EM-RE-6618K-FTE	K Relay Sch 6618 Fails To Energize
EM-RE-6618K-FTRE	K Relay Sch 6618 Fails To Remain Energized
EM-RE-TDR-FTE	SV-4947 TDR Relay-Sch 6618 Fails To Energize
EM-PM-P98-FTR	Portable Pump P98 Fails To Run
EM-PM-P98-FTS	Portable Pump P98 Fails To Start
EM-PM-SP98-FTR	Portable Pump P98 Stand-By Failure
EM-HS-S618C-FTRE	RS-6618-1 Stand-By Failure T30-56
EM-KV-4947-FTRE	SV-4947 Fails To Remain Energized
EM-HS-6618C-FTRE	TRS-6618-1 Fails To Remain Energized or in Contr.Rm Position
EM-CK-S2D14-FTRC	Breaker 2D-14 (ASD Panel C-31) Stand-By Failure
EM-CK-2D14-FTRC	Breaker 2D-14 (ASD Panel C-31 Feed) Fails To Remain Closed

TABLE 13.1-3
SYSTEM GROUPING FOR
IMPORTANCE DETERMINATION

Emergency Power (EP)	
Fussell-Vesely = 1.5E-2	**Birnbaum = 4.7E-5**
EP-GE-DG-CCFTR	SDG and EDG Common Cause Failure To Run
EP-GE-DG-CCFTS	SDG and EDG Common Cause Failure To Start
SPS-I-29	SDG Independent Failures
SPS-I-7	EDG Independent Failures
Enclosure Spray (ES)	
Fussell-Vesely = 4.4E-3	**Birnbaum = 8.0E-5**
ES-CV-PI310-FTO	VPI-310 Fails To Open (Drum Enclosure Sprays)
ES-CV-PI312-FTO	VPI-312 Fails To Open (Drum Enclosure Sprays)
ES-HS-S5514-FTRE	MO-7064 RMC-5514 Stand-By Failure (2-2C Cont Pair)
ES-MV-48-CCFTO	ES Valves MV-7064 & 68 Common Cause Failure To Open
ES-MV-7064-FTO	MO-7064 Fails To Open
ES-MV-7068-FTO	MO-7068 Fails To Open
ES-RE-D01O-FTE	MO-7064 Relay 42/O Fails To Energize
Fire Protection (FP)	
Fussell-Vesely = 1.2E-1	**Birnbaum = 1.7**
FP-BY-BAT-CCFTE	DFP Battery A And B Common Cause Failure
FP-CB-2B26-FTRC	Breaker 2B-26 (P-6) Fails To Remain Closed
FP-CB-2B26-OOS	Breaker 2B-26 (P-6) Out of Service
FP-CB-PC17-FTRC	Breaker EFP Local (Panel C-17) Fails To Remain Closed
FP-CB-S2B26-FTRC	Breaker 2B-26 (P-6) Stand-By Failure
FP-CB-SPC17-FTRC	Breaker EFP Local (Panel C-17) Stand-By Failure

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
FP-CS-1CSKK-FTRE	Breaker 1CB (Panel C-09) Fails To Remain Closed
FP-CS-ECS-FTRE	Engine Control Switch Fails To Remain Closed
FP-CS-S1CSK-FTRE	Control Switch 1CS On KK Panel Stand-By Failure
FP-CS-SECS-FTRE	Engine Control Switch Stand-By Failure
FP-CV-8405-FTC	DFP Air Release Valve VAR-8405 Fails To Close
FP-CV-8405-FTRC	DFP Air Release Valve VAR-8405 Fails To Remain Closed
FP-CV-8406-FTC	EFP Air Release Valve VAR-8406 Fails To Close
FP-CV-8406-FTRC	EFP Air Release Valve VAR-8406 Fails To Remain Closed
FP-CV-FP304-FTC	Check Valve VFP-304 Fails To Close
FP-CV-FP304-FTO	Electric Fire Pump Check Valve VFP-304 Fails To Open
FP-CV-FP309-FTC	Check Valve VFP-309 Fails To Close
FP-CV-FP309-FTO	Diesel Fire Pump Check Valve VFP-309 Fails To Open
FP-CV-VLV-CCFTO	Fire Pump Check Valves Common Cause Failure To Open
FP-KV-FSOS-FTE	DFP Fuel Sol Fails To Energize
FP-KV-FSOS-FTRE	DFP Fuel Sol Fails To Remain Energized
FP-KV-PNS1-FTE	DFP Pinion Solenoid PNS1 Fails To Energize
FP-PM-P6-FTR	Electric Fire Pump P-06 Fails To Run
FP-PM-P6-FTS	Electric Fire Pump Fails To Start
FP-PM-P6-OOS	EFP Out of Service
FP-PM-P7-FTR	Diesel Fire Pump P-07 Fails To Run

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
FP-PM-P7-FTS	Diesel Fire Pump Fails To Start
FP-PM-P7-OOS	DFP Out of Service
FP-PS-PMP-CCFTC	FPS PS-612 & 615 Common Cause Failure To Close
FP-PS-PS612-FTC	DFP Pressure Switch PS-612 Fails To Close
FP-PV-4515-FTRO	DFP Cooling Water PCV-4515 Fails To Remain Open
FP-PV-4515-PLUG	DFP Cooling Water PCV-4515 Plugged
FP-RE-1CR-FTE	Relay 1CR Fails To Energize
FP-RE-2CR-FTE	Relay 2CR Fails To Energize
FP-RE-4CR-FTE	Relay 4CR Fails To Energize
FP-RE-A1-FTE	EFP Start Relay A1 Fails To Energize
FP-RE-CR-FTE	Electric Fire Pump Start Relay CR Fails To Energize
FP-RE-SR1-FTE	DFP Fuel Relay SR1 Fails To Energize
FP-RV-5062-FTC	RV-5062 DFP 4 Inch Relief Valve Fails To Close
FP-RV-5062-FTRC	RV-5062 DFP 4 Inch Relief Valve Fails To Remain Closed
FP-TK-T53-FAIL	DFP Fuel Tank T-53 Empty
FP-XV-FP013-PLUG	VFP-013 L.O. Plugged Screenhouse Out
FP-XV-FP033-FTO	VFP-33 Fails To Open FPS Hotwell Fill Line
Fire Water Makeup to Hotwell (FH)	
Fussell-Vesely = 9.6E-3	**Birnbaum = 8.0E-5**
FP-MV-7073-FTO	MO-7073 Fails To Open FPS Hotwell Fill Line
FP-MV-7074-FTO	MO-7074 Fails To Open FPS Hotwell Fill Line

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
Feedwater (FW)	
Fussell-Vesely = 4.8E-3	**Birnbaum = 2.0E-4**
FW-BV-VFW9-FTO	Stop Check Valve VFW-9 Fails To Open
FW-CB-52103-FTC	Breaker 52-103 (P-8A) Fails To Energize
FW-CB-52105-FTC	Breaker 52-105 (P-8B) Fails To Energize
FW-CB-PMP-CCFTC	CBS 52-103 & 52-105 For P8A&B Fails To Close On Pump Restart
FW-CB-S1A43-FTRC	Breaker 1A-43 (P-26A) Stand-By Failure
FW-CB-S2A43-FTRC	Breaker 2A-43 (P-26B) Stand-By Failure
FW-CV-FW300-FTC	VFW-300 Feed Pump P-8A Disch Check Valve Fails To Close
FW-CV-FW300-FTO	VFW-300 Feed Pump P-8A Disch Check Valve Fails To Open
FW-CV-FW301-FTC	VFW-301 Feed Pump P-8B Disch Check Valve Fails To Close
FW-CV-FW301-FTO	VFW-301 Feed Pump P-8B Disch Check Valve Fails To Open
FW-CV-FW304-FTO	Check Valve VFW-304 Fails To Open
FW-CV-FW305-FTO	Check Valve VFW-305 Fails To Close
FW-CV-VLV-CCFTO	P-8A & B VFW-300 and 301 Common Cause Failure To Open
FW-FU-S2401-OPCI	Fuse Sch 2401 Stand-By Failure
FW-FU-S2402-OPCI	Fuse Sch 2402 Stand-By Failure
FW-FV-4000-FTO	Feed Water Reg Valve CV-4000 Fails To Open
FW-FV-4000-FTRO	CV-4000 Feed Water Reg Valve Fails To Remain Open
FW-HS-S2401-FTRE	P-8A Aux Oil Pump Hand Switch Sch 2401 Stand-By Failure
FW-HS-S2402-FTRE	P-8B Aux Oil Pump Hand Switch Sch 2402 Stand-By Failure

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
FW-OX-S2401-FAIL	P-26A Circuit Transformer Stand-By Failure
FW-OX-S2402-FAIL	P-26B Circuit Transformer Stand-By Failure
FW-PM-P26-CCFTR	Aux Oil P-26A and P-26B Common Cause Failure To Run
FW-PM-P26-CCFTS	Aux Oil P-26A and P-26B Common Cause Failure To Start
FW-PM-P26A-FTR	P-8A Aux Oil Pump P-26A Fails To Run
FW-PM-P26A-FTS	P-8A Aux Oil Pump P-26A Fails To Start
FW-PM-P26A-OOS	P-8A Aux Oil Pump P-26A Out For Maintenance
FW-PM-P26B-FTR	P-8B Aux Oil Pump P-26B Fails To Run
FW-PM-P26B-FTS	P-8B Aux Oil Pump P-26B Fails To Start
FW-PM-P26B-OOS	P-8B Aux Oil Pump P-26B Out For Maintenance
FW-PM-P8-CCFTS	Feedwater P-8A and P-8B Common Cause Failure To Run
FW-PM-P8A-FTR	Feed Pump P-8A Fails To Run
FW-PM-P8A-FTS	Feedwater Pump P-8A Fails To Start
FW-PM-P8B-FTR	Feed Pump P-8B Fails To Run
FW-PM-P8B-FTS	Feedwater Pump P-8B Fails To Start
FW-PS-608A-FTO	Feed Pump Suction Pressure Switch PS-608A Fails To Open
FW-PS-608B-FTO	Feed Pump Suction Pressure Switch PS-608B Fails To Open
FW-PS-6291-FTC	P-8A Lube Oil Pressure Switch PS-629-1 Fails To Close
FW-PS-630-FTC	P-8A Lube Oil Pressure Switch PS-630 Fails To Close
FW-PS-6321-FTC	P-8B Lube Oil Pressure Switch PS-632-1 Fails To Close

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
FW-PS-633-FTC	P-8B Lube Oil Pressure Switch PS-633 Fails To Close
FW-PS-AB-CCFTO	Pressure Switches PS-608A&B Common Cause Failure To Open
FW-PS-PLUB-CCFTC	P-8A&B Lube Oil PS-629-1&2 Common Cause Standby Failure
FW-RE-1621-FTD	Feed Pump P-8A Auxiliary Relay 162-1 Fails To De-Energize
FW-RE-1622-FTD	Feed Pump P-8B Auxiliary Relay 162-2 Fails To De-Energize
FW-RE-2401-FTE	Relay 42/HC Sch 2401 Fails To Energize
FW-RE-2402-FTE	Relay 42/HC Sch 2402 Fails To Energize
Main Condenser (MC)	
Fussell-Vesely = 7.4E-4	**Birnbaum = 1.1E-5**
MC-KV-4868-FTRE	SV-4868 Fails To Remain Energized For CV-4030
MC-RG-SSREG-FAIL	Steam Seal Regulator Loss of Function
MC-RV-5007-FTRC	RV-5007 Fails To Remain Closed Steam Supply To Turbine Seal
MTBMTSSEAL	Turbine Steam Seals Degraded During Operation
Main Steam Isolation (MS)	
Fussell-Vesely = 2.9E-3	**Birnbaum = 1.8E-5**
MS-AV-4106-FTO	CV-4106 Bypass Vlve Disc Pipe Warming Line Fails To Open
MS-AV-4106-OOS	CV-4106 Bypass Valve Warming Line Out of Service
MS-AV-4106-POOC	CV-4106 Bypass Valve Disc Pipe Warming Line Not Opened
MS-HV-4014-FTO	Turbine Bypass Valve CV-4014 Fails To Open
MS-KV-4899-FTRE	SV-4899 Fails To Remain Energized For CV-4104
MS-KV-4916-FTE	SV-4916 Fails To Energize For CV-4106
MS-KV-4932-FTE	SV-4932 Fails To Energize For CV-4014

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
MS-MV-7050-FTC	MO-7050 Main Steam Isolation Valve Fails To Close
MS-RE-66051-FTE	Relay 42/1C Sch 6605 Fails To Energize
MS-RE-66052-FTE	Relay 42/1C Sch 6605 Fails To Remain Energized
Primary Coolant SRV fails to reclose (PC-RV)	
Fussell-Vesely = 4.4E-3	**Birnbaum = 4.5E-7**
PC-RV-SDSRV-FTC	Steam Drum Safety Relief Valves Fail To Close
Reactor Depressurization System (RD)	
Fussell-Vesely = 1.4E-1	**Birnbaum = 2.2E-1**
RD-AV-01-CCFTO	CV-4180 and CV-4181 Common Cause Failure To Open
RD-AV-012-CCFTO	CV-4180 & CV-4181 & CV-4182 Common Cause Failure To Open
RD-AV-0123-CCFTO	RDS Isolation Vlvs Lethal Shock Common Cause Failure To Open
RD-AV-013-CCFTO	CV-4180 & CV-4181 & CV-4183 Common Cause Failure To Open
RD-AV-02-CCFTO	CV-4180 and CV-4182 Common Cause Failure To Open
RD-AV-023-CCFTO	CV-4180 & CV-4182 & CV-4183 Common Cause Failure To Open
RD-AV-03-CCFTO	CV-4180 and CV-4183 Common Cause Failure To Open
RD-AV-12-CCFTO	CV-4181 and CV-4182 Common Cause Failure To Open
RD-AV-123-CCFTO	CV-4181 & CV-4182 & CV-4183 Common Cause Failure To Open
RD-AV-13-CCFTO	CV-4181 and CV-4183 Common Cause Failure To Open
RD-AV-23-CCFTO	CV-4182 and CV-4183 Common Cause Failure To Open
RD-CK-1Y26-FTRC	Breaker 1Y-26 (Panel C-40) Fails To Remain Closed
RD-CK-SUP7A-FTRC	Breaker CB-7 (Ups A) Stand-By Failure
RD-FU-UP9A-OPCI	Fuse UPS CB-9A Stand-By Failure

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
RD-FU-UP9C-OPCI	Fuse UPS CB-9C Stand-By Failure
RD-KV-01-CCFTE	SV-4180 and SV-4181 Common Cause Failure To Energize
RD-KV-012-CCFTE	SV-4180 & SV-4181 & SV-4182 Common Cause Failure To Energize
RD-KV-0123-CCFTE	RDS Iso Valves Lethal Shock Common Cause Failure To Energize
RD-KV-013-CCFTE	SV-4180 & SV-4181 & SV-4183 Common Cause Failure To Energize
RD-KV-02-CCFTE	SV-4180 and SV-4182 Common Cause Failure To Energize
RD-KV-023-CCFTE	SV-4180 & SV-4182 & SV-4183 Common Cause Failure To Energize
RD-KV-03-CCFTE	SV-4180 and SV-4183 Common Cause Failure To Energize
RD-KV-12-CCFTE	SV-4181 and SV-4182 Common Cause Failure To Energize
RD-KV-123-CCFTE	SV-4181 & SV-4182 & SV-4183 Common Cause Failure To Energize
RD-KV-13-CCFTE	SV-4181 and SV-4183 Common Cause Failure To Energize
RD-KV-23-CCFTE	SV-4182 and SV-4183 Common Cause Failure To Energize
RD-KV-45-CCFTE	SV-4984 and SV-4985 COmmon Cause Failure To Energize
RD-KV-456-CCFTE	SV-4984 & SV-4985 & SV-4986 Common Cause Failure To Energize
RD-KV-4567-CCFTE	RDS Iso Valves Lethal Shock Common Cause Failure To Energize
RD-KV-457-CCFTE	SV-4984 & SV-4985 & SV-4987 Common Cause Failure To Energize
RD-KV-46-CCFTE	SV-4984 and SV-4986 Common Cause Failure To Energize
RD-KV-467-CCFTE	SV-4984 & SV-4986 & SV-4987 Common Cause Failure To Energize
RD-KV-47-CCFTE	SV-4984 and SV-4987 Common Cause Failure To Energize
RD-KV-56-CCFTE	SV-4985 and SV-4986 Common Cause Failure To Energize

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
RD-KV-567-CCFTE	SV-4985 & SV-4986 & SV-4987 Common Cause Failure To Energize
RD-KV-57-CCFTE	SV-4985 and SV-4987 Common Cause Failure To Energize
RD-KV-67-CCFTE	SV-4986 and SV-4987 Common Cause Failure To Energize
RD-RE-AC3K7-FTE	Fire Pump Start Circuit AC3 Relay K7 Fails To Energize
RD-RE-RF-FTD	RDS Fire Pump Start Relay RF Fails To De-Energize
RD-RV-45-CCFTO	SV-4984 and SV-4985 Common Cause Failure To Open
RD-RV-456-CCFTO	SV-4984 & SV-4985 & SV-4986 Common Cause Failure To Open
RD-RV-4567-CCFTO	RDS Dpres Valves Lethal Shock Common Cause Failure To Open
RD-RV-457-CCFTO	SV-4984 & SV-4985 & SV-4987 Common Cause Failure To Open
RD-RV-46-CCFTO	SV-4984 and SV-4986 Common Cause Failure To Open
RD-RV-467-CCFTO	SV-4984 & SV-4986 & SV-4987 Common Cause Failure To Open
RD-RV-47-CCFTO	SV-4984 and SV-4987 Common Cause Failure To Open
RD-RV-56-CCFTO	SV-4985 and SV-4986 Common Cause Failure To Open
RD-RV-567-CCFTO	SV-4985 & SV-4986 & SV-4987 Common Cause Failure To Open
RD-RV-57-CCFTO	SV-4985 and SV-4987 Common Cause Failure To Open
RD-RV-67-CCFTO	SV-4986 and SV-4987 Common Cause Failure To Open
RD-SC-C40-FAIL	General Fault - No Signal Generated From The C-40 Panel
RD-SC-EC42-FAIL	No Signal From Actuation Cabinet AC-1
RD-SC-EC46-FAIL	No Signal From Actuation Cabinet AC-3
RD-SC-EC46-OOS	RDS Actuation Cabinet AC-3 Out of Service

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
RD-TL-012-CCFAIL	TL-3180 & TL-3181 & TL-3182 Common Cause Failure To Function
RD-TL-0123-CCFAIL	RDS RPV Vessel Low Level Transmitter Lethal Shock Common Cause Failure
RD-TL-013-CCFAIL	TL-3180 & TL-3181 & TL-3183 Common Cause Failure To Function
RD-TL-023-CCFAIL	TL-3180 & TL-3182 & TL-3183 Common Cause Failure To Function
RD-TL-123-CCFAIL	TL-3181 & TL-3182 & TL-3183 Common Cause Failure To Function
RD-TL-456-CCFAIL	TL-3184 & TL-3185 & TL-3186 Common Cause Failure To Function
RD-TL-4567-CCFAIL	RDS Steam Drum Level Transmitter Lethal Shock Common Cause Failure
RD-TL-457-CCFAIL	TL-3184 & TL-3185 & TL-3187 Common Cause Failure To Function
RD-TL-467-CCFAIL	TL-3184 & TL-3186 & TL-3187 Common Cause Failure To Energize
RD-TL-567-CCFAIL	TL-3185 & TL-3186 & TL-3187 Common Cause Failure To Function
RD-TL-IC54-CCCAL	Level Transmitter LT-3180 To 83 Common Calibration Faults
RD-TL-IC55-CCCAL	Level Transmitter LT-3184 To 87 Common Calibration Faults
RDS-I-1	UPSA Failures
RDS-I-10	SV-4984 Failures
RDS-I-11	CV-4181 Failures
RDS-I-12	SV-4985 Failures
RDS-I-13	CV-4182 Failures
RDS-I-14	SV-4986 Failures
RDS-I-15	CV-4183 Failures
RDS-I-16	SV-4987 Failures
RDS-I-21	CV-4184 Failures
RDS-I-22	Nitrogen Supply Failures
RDS-I-24	Level Transmitter LT-3180 Failures
RDS-I-25	Level Transmitter LT-3181 Failures
RDS-I-26	Level Transmitter LT-3182 Failures

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
RDS-I-27	Level Transmitter LT-3183 Failures
RDS-I-9	CV-4180 Failures
Reactor Protection System Level Elements (RPL)	
Fussell-Vesely = 9.2E-4	**Birnbaum = 4.4E-3**
RP-LE-IC-CCCAL	RPS LE-RE08A&B and RE09A-D Common Calibration Faults
RP-PS-IC-CCCAL	RPS Pressure Switches PS-RE09A-D Common Calibration Faults
RP-LE-RE08A-FAIL	Level Element LE-RE08A Loss of Function
RP-LE-RE08B-FAIL	Level Element LE-RE08B Loss of Function
Station Power (SP)	
Fussell-Vesely = 2.2E-2	**Birnbaum = 2.6E-2**
SP-BS-PNL1Y-OPCI	Panel 1Y Open Circuit
SP-BS-PNL1Y-SCLG	Panel 1Y Short To Ground
SP-BS-PNL2Y-OOS	Panel 2Y Open Circuit
SP-BS-PNL2Y-SCLG	Panel 2Y Short To Ground
SP-BS-PNL3Y-SCLG	Panel 3Y Short To Ground
SP-CB-1126-FTO	Breaker OCB-1126 (Sta Pwr Trans No 1 Output) Fails To Open
SP-CB-2A2B-FTRC	Breaker 2A-2B (MCC-2A To MCC-2B) Fails To Remain Closed
SP-CB-2A2B-OOS	Breaker 2A-2B (MCC-2A To MCC-2B) Out of Service
SP-CK-1D43-FTRC	Breaker 1D-43 (2400V Switch-Gear) Fails To Remain Closed
SP-OO-138KV-FAIL	Conditional Failure of The 138 kv Transmission Line (24 Hrs)
SP-OO-LOSP-FAIL	Conditional Failure Of 46kv Line For LOSP
SPS-I-11	Motor Control Center MCC-2B Faults
SPS-I-13	Breaker 1F-1E Faults
SPS-I-16	OCB-1126 and OCB-199 Faults

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
SPS-I-19	Panel 2Y Faults
SPS-I-21	Panel C04 Faults
SPS-I-22	Breaker 2B-23 Faults
SPS-I-23	Power To Panel 2P Failures
SPS-I-24	Fuse 1307A,C and D Failures
SPS-I-25	Fuse 1307B,E and F Failures
SPS-I-27	Bus 2 Failures
SPS-I-28	Motor Control Center MCC-2A Faults
SPS-I-3	Bus 1F Failures
SPS-I-30	OCB-1199 Failures
SPS-I-4	Bus 1 Failures
SPS-I-5	OCB-2299 Failures
SPS-I-6	OCB-1136 Faults
SPS-I-8	OCB-1126/7726 Handswitch Failures
SPS-I-9	Motor Control Center MCC-1 Failures
ASD Power Supply (ASD)	
Fussell-Vesely = 2.3E-4	**Birnbaum = 1.2E-4**
SPS-I-1	Bus D12 Failures
SPS-I-2	Circuit 1610 OOS
ASD-BAT-DISCNCT	No Power From ASD Battery Disconnect
Service Water (SW)	
Fussell-Vesely = 4.3E-4	**Birnbaum = 1.5E-4**
SW-FL-5751-PLUG	Service Water Basket Strainer BS-5751 Plugged
SW-PM-P12A-FTR	CC Failures of SW Pumps P12-A&B To Run
SW-PM-P12-CCFTR	CC Failures of SW Pumps P12-A&B To Run
Liquid Poison (LP)	
Fussell-Vesely = 1.4E-4	**Birnbaum = 5.5E-5**
LI-AN-A1433-FAIL	LPS Nitrogen System Low Pressure Annunciator
LI-CB-2P8-FTRC	Circuit Breaker 2P-8 Fails To Remain Closed
LI-CV-VP300-FTO	Check Valve VP-300 Fails To Open
LI-OY-6401-FAIL	Transformer Scheme 6401 Fails

TABLE 13.1-3 SYSTEM GROUPING FOR IMPORTANCE DETERMINATION	
LI-PS-PS607-FTC	Pressure Switch PS-607 Fails To Close
Reactor Protection System (RPS)	
Fussell-Vesely = 3.8E-1	**Birnbaum = 7.0E-2**
These are the importance measures for the entire ATWS contribution to the core damage frequency	

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

System Component	
Circulating Water (CW)	
Fussell-Vesely =1.9E-6	**Birnbaum =3.5E-6**
MO-7054	MO-7054 Out of Service
MO-7055	MO-7055 Out of Service
Intake Structure (INTAKE)	
Fussell-Vesely =1.7E-4	**Birnbaum = 1.1**
PS-601 and PS-602	Traveling Screen PS-601&611 Common Calibration Faults
TS-001	Traveling Screen #1 Plugged
TS-002	Traveling Screen #2 Plugged
Primary Coolant SRV's fail to open (PC)	
Fussell-Vesely = 1.3E-11	**Birnbaum = 4.5E-9**
Steam Drum Safety Relief Valves	Common Cause Failure of All 6 Steam Drum Safety Relief Valves To Open
Post Incident (PI)	
Fussell-Vesely = 1.7E-1	**Birnbaum =1.4E-1**
Breaker 1A-44	Breaker 1A-44 (P-2A) Stand-By Failure
Breaker 2A-44	Breaker 2A-44 (P-2B) Stand-By Failure
Breaker 1Y-23	Breaker 1Y-23 (Scheme 6503) Stand-By Failure
VPI-306 and VPI-307	P-2A&B Check Valves VPI-306 & 7 Common Cause Failure To Open
VPI-300	Check Valve VPI-300 Fails To Open
VPI-306	P-2A Check Valve VPI-306 Fails To Open
VPI-307	P-2B Check Valve VPI-307 Fails To Open
HX-06	Core Spray Heat Exchanger Stand-By Failure
LOCA Hose	LOCA Hose Stand-By Failure

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

MO-7066 and MO-7080	PI System Valves MO-7066&80 Common Cause Failure To Open
P-2A and P-2B	CS Pumps P2A & B Common Cause Failure To Run
P-2A and P-2B	CS Pumps P2A & B Common Cause Failure To Start
P-2A	P-2A Fails To Run
P-2A	P-2A Fails To Start
P-2B	Pump P-2B Fails To Run
P-2B	Pump P-2B Fails To Start
Scheme 5415 Motor Control Relay	P-2A Mot Cont Relay Scheme 5415 Fails To Energize
Scheme 5416 Motor Control Relay	P-2B Mot Cont Relay Sch 5416 Fails To Energize
LT-3175	Level Transmitter LT-3175 Out of Service
LT-3171	Level Transmitter LT-3171 Stand-By Failure
LT-3175	Level Transmitter LT-3175 Stand-By Failure
Instrument Air (CA)	
Fussell-Vesely = 1.0E-2	**Birnbaum = 5.3E-4**
M-49A and M-49B	Compressors M-49A and M-49B Common Cause Failure To Run
M-49A and M-49B	Compressors M-49A and M-49B Common Cause Failure To Start
M-49A, M-49B and M-49C	Compressors M-49A & M-49B & M-49C Common Cause Fail To Run
M-49A, M-49B and M-49C	Compressors M-49A & M-49B & M-49C Common Cause Fail To Start
M-49A and M-49C	Compressors M-49A and M-49C Common Cause Failure To Run
M-49B and M-49C	Compressors M-49B and M-49C Common Cause Failure To Start
VA-301 and VA-302	Compr 1 & 2 Disch Check Valve Common Cause Failure To Open

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

VA-301, VA-302 and VA-303	Compr 1 2 & 3 Disch Check Valve Common Cause Failure To Open
VA-301 and VA-303	Compr 1 & 3 Disch Check Valve Common Cause Failure To Open
VA-302 and VA-303	Compr 2 & 3 Disch Check Valve Common Cause Failure To Open
VA-301	Air Compressor 1 Disch Check Valve VA-301 Fails To Close
VA-302	Air Compressor 2 Disch Check Valve VA-302 Fails To Close
VA-302	VA-302 F-25B Disch Check Valve Fails To Remain Closed
VA-303	Air Compressor 3 Disch Check Valve VA-303 Fails To Close
VA-303	VA-303 F-25C Disch Check Valve Fails To Remain Closed
VA-304	Check Valve VA-304 Fails To Open
SV-4913 and SV-4914	Compr SV Valves SV-4913, 4 Common Cause Fails to Energize
SV-4913, SV-4914 and SV-4915	Compr SV Valves SV-4913, 4, 5 Common Cause Failure To Energize
SV-4913 and SV-4915	Compr SV Valves SV-4913, 5 Common Cause Failure To Energize
SV-4914 and SV-4915	Compr SV Valves SV-4914, 5 Common Cause Failure To Energize
PS-602 and PS-610	Compr 1 & 2 Press Switch Common Cause Failure To Close
PS-602, PS-610 and PS-614	Compr 1 2 & 3 Press Switch Common Cause Failure To Close
PS-602 and PS-614	Compr 1 & 3 Press Switch Common Cause Failure To Close
PS-610 and PS-614	Compr 2 & 3 Press Switch Common Cause Failure To Close
PS-687 and PS-688	Compr 2 & 3 Start Press Switch Common Cause Failure To Close
Scheme 3406 Relay 42/CS	M-49C Relay 42/HC Sch 3406 Fails To Energize

TABLE 13.1-4
SYSTEM GROUPING FOR IMPORTANCE MEASURES

F-26B, RV-5033, RV-5011 and RV-5010	Common Air Compressor Failures Such As Heat Exchanger Failures and Relief Valve Failures
Breaker 1A-34	Circuit Breaker 1A-34 (HSAC#1) Failures
Breaker 1A-35	Circuit Breaker 1A-35 (HSAC#2) Failures
PS-602	Pressure Switch PS-602 Faults
PS-610	Pressure Switch PS-610 Faults
PS-614	Pressure Switch PS-614 Faults
VA-302	Train A Faults; Check Valve VA-302
SV-4885 and VA-302	Train B Faults; SV-4885 and Check Valve VA-302
SV-4886 and VA-303	Train C Faults; SV-4886 And Check Valve VA-303
M-49A and SV-4913	HSAC#1 and SV-4913 Faults
M-49B and SV-4914	Operator Fails To Place Handswitch In Standby and Failure To Start HSAC#2
M-49C	HSAC#3 OOS and Operator Fails To Start HSAC
Condensate (CD)	
Fussell-Vesely = 3.8E-3	**Birnbaum = 3.8E-5**
CV-4009	CV-4009 Fails To Open Hot Well Fill Line
Breaker 52-12	Breaker 52-12 (P-9A) Fails To Close
Breaker 52-22	Breaker 52-22 (P-9B) Fails To Close
Breakers 52-12 and 52-22	Breakers 52-12 & 22 Common Cause Failure (P9A And P9B)
Breaker 1Y-7	Breaker 1Y-7 CDS Valve Controls Fails To Remain Closed
VC-007	Condensate Supply Check Valve Fails To Open
VC-301	P-9B Disch Check Valve Fails To Close
VC-301	P-9B Discharge Check Valve Fails To Open

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

VC-300 and VC-301	P-9A & B VC-300 & 301 Common Cause Failure To Open
SV-4855	SV-4855 Fails To Energize For CV-4009 (Hotwell Fill Valve)
P-9A and P-9B	Condensate P-9A and P-9B Common Cause Failure To Start
P-9A	Condensate Pump P-9A Fails To Start
P-9B	Condensate Pump P-9B Fails To Start
Control Rod Drive (CRD)	
Fussell-Vesely = 3.7E-3	**Birnbaum = 2.9E-4**
CV-4090	CV-4090 Fails To Open
CV-4090	CV-4090 Fails To Remain Open
VCRD-NC-18	CV-NC18 Fails To Remain Open
Breaker 2A-58	Breaker 2A-58 (P-4B) Fails To Remain Closed
VRD-303	Check Valve VRD-303 Fails To Open
VRD-313	Check Valve VRD-313 Fails To Open
YS-5827	CRD Strainer YS-5827 Plugged
YS-NC-07	CRD Strainer YS-NC07 Plugged
S-23A	Control Rod Drive Filter S23A Plugged
YS-5826	YS-5826 Stand-By Failure
Scheme 6412 Fuse	Fuse Sch 6412 Open Circuit
SV-4894	Sv-4894 Fails To De-Energize For CV-4090
Scheme 6412 Transformer	Circuit Transformer Sch 6412 Fails
P-4A and P-4B	CRD P-4A And P-4B Common Cause Failure To Run

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

P-4A	CRD Pump P4A Fails To Run
P-4A	CRD Pump P4A Fails To Start
P-4A	CRD Pump P4A Out of Service
P-4B	CRD Pump P4B Fails To Run
PS-626-2	PS-626-2 Fails To Open
PC-NC18	Pressure Controller PCV-NC18 Fails To Operate
RV-5050	RV-5050 Fails To Remain Closed
RV-5051	RV-5051 Fails To Remain Closed
Core Spray (CS)	
Fussell-Vesely = 4.0E-1	**Birnbaum = 1.7E-1**
VPI-301 and VPI-302	CS RPV Inject Vlvs VPI-301&302 Common Cause Failure To Open
VPI-303 and VPI-304	CS RPV Inject Vlvs VPI-303&304 Common Cause Failure To Open
VPI-303	Back-Up CS Check Valve VPI303 Fails To Open
VPI-304	Primary CS Check Valve VPI304 Fails To Open
FE-2815	Flow Element FE-2815 Stand-By Failure
LS-RE09	Level Switch LS-RE09 Common Calibration Error
MO-7051	MO-7051 Fails To Open
MO-7051	MO-7051 Out of Service
MO-7061	MO-7061 Fails To Open
MO-7061	MO-7061 Out of Service
MO-7070	MO-7070 Fails To Open

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

MO-7071	MO-7071 Fails To Open
MO-7051, MO-7061,MO-7070 and MO-7071	CS RPV Inj MO-7051, 61, 70, 71 CC Failure To Open
MO-7051 and MO-7070	CS RPV Inj MO-7051 & 70 CC Failure To Open
MO-7051 and MO-7071	CS RPV Inj MO-7051 & 71 CC Failure To Open
MO-7061 and MO-7070	CS RPV Inj MO-7061 & 70 CC Failure To Open
MO-7061 and MO-7071	CS RPV Inj MO-7061 & 71 CC Failure To Open
PS-IG11A-H	Core Spray PS-IG11A-H Common Calibration Faults
Scheme 163 Relay 42/O	Relay 42/O Sch 163 Fails To Energize
Scheme 5601 Relay 42/O	Open Relay 42/O Sch 5601 Fails To Energize
Scheme 5602 Relay 42/O	Open Relay 42/O Sch 5602 Fails To Energize
Scheme B152 Relay 42/O	Relay 42/O Sch B152 Fails To Energize
DC Power (DC)	
Fussell-Vesely = 1.1E-4	**Birnbaum =9.4E-5**
Bus D02	Bus D02 Open Circuit
Bus D02	Bus D02 Short To Ground
MCC-01	125vdc MCC-01 Open Circuit
MCC-01	125 vdc MCC-01 Short To Ground
Station Batteries	Station Batteries Fail To Operate On Demand
Breaker 1D-40	Breaker 1D-40 (480V Load Center) Fails To Remain Closed
Breaker 72-11	Breaker 72-11 (125vdc MCC D01 Feed) Fails To Remain Closed

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Breaker 72-12	Breaker 72-12 (125Vdc Panel D02 Feed) Fails To Remain Closed
Emergency Condenser Valve (EV)	
Fussell-Vesely = 2.9E-3	**Birnbaum = 4.8E-4**
Emergency Condenser Shell	Emergency Condenser Shell Leaks/Ruptures
MO-7053 and MO-7063	EC Valves MV-7053 & 63 Common Cause Failure To Open
MO-7053	MO-7053 Fails To Open
MO-7053	Emergency Condenser Loop 2 Out of Service
MO-7063	MO-7063 Fails To Open
MO-7063	Emergency Condenser Loop 1 Out of Service
PS-RE07A and C	PS-RE07A & C Common Cause Failure To Open
PS-RE07B and D	PS-RE07B & D Common Cause Failure To Open
Scheme 6602 Relay 42/O	Relay 42-O Sch 6602 MO-7053 Fails To Energize
Scheme 6603 Relay 42/O	Relay 42-O Sch 6603 MO-7063 Fails To Energize
Demin Water Makeup (DM)	
Fussell-Vesely =4.7E-5	**Birnbaum =2.2E-7**
CV-4028	Control Valve CV-4028 Fails To Open
CV-4028	Control Valve CV-4028 Fails To Remain Open
CV-4041	CV-4041 Demin Water To CST Fails To Remain Closed
CV-4105	Air Operated (Diaphram) Vlv CV-4105 Fails To Remain Open
CV-4041	CV-4041 Demin Water To CST Stand-By Failure
Breaker 1E-22	Breaker 1E-22 (P-37) Fails To Remain Closed
Breaker 1E-22	Breaker 1E-22 (P-37) Stand-By Failure

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Breaker 2Y-7	Breaker 2Y-7 (Scheme 6505) Fails To Remain Closed
Breaker 2Y-8	Breaker 2Y-8 (SV-4874 For CV-4028) Fails To Remain Closed
Breaker 2Y-8	Breaker 2Y-8 (SV-4874 For CV-4028) Stand-By Failure
VEC-300	Check Valve VEC-300 Fails To Open
VMU-300	VMU-300 Demin Water To Sphere Check Valve Fails To Open
Scheme 5412 Fuse	P-37 Fuse Sch 5412 Open Circuit
Scheme 5412 Fuse	P-37 Fuse Sch 5412 Open Circuit Stand-By Failure
HS-7045	Lvl Bypass Switch HS-7045 Sch 5412 Fails To Remain Energized
Scheme 5412 Handswitch 42/CS	Hand Switch 42/CS Sch 5412 Fails To Remain In Run
HS-7025	Hand Switch HS-7025 Fails To Remain Energized
HS-7902	HS-7902 Fails To Energize
Scheme 5412 Handswitch 42/CS	Hand Switch 42/CS SCH 5412 Stand-By Failure
HS-7045	Level Bypass Switch HS-7045 Sch 5412 Stand-By Failure
HS-7025	Hand Switch HS-7025 Stand-By Failure
SV-4874	SV-4874 Fails To Energize For Cv-4028
SV-4874	SV-4874 Fails To Remain Energized For CV-4028
SV-4897	SV-4897 Fails To Energize For CV-4105
SV-4897	SV-4897 Fails To Remain Energized For CV-4105
Breaker 1F-1E	Undervoltage Relays For Breaker 1F-1E Failure
Demin Water System	Demin Water System Unavailable To Provide Water

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

T-31	Demin Water Tank T-31 Empty
P-37	Demin Water Pump P-37 Fails To Run
P-37	Demin Water Pump P-37 Out of Service
Scheme 5412 42/HC	Relay 42/HC Sch 5412 Fails To Energize
Scheme 5412 42/HC	Relay 42/HC Sch 5412 Fails To Remain Energized
T-31	Demin Water Tank T-31 Failure
Emergency Condenser Makeup-Fire Protection (FM)	
Fussell-Vesely = 2.6E-2	**Birnbaum =1.3E-3**
SV-4947	SV-4947 Fails To Energize
SV-4947	SV-4947 Fails To Remain Energized
Scheme 6618 Relay K	K Relay Sch 6618 Fails To Energize
Scheme 6618 Relay K	K Relay Sch 6618 Fails To Remain Energized
Scheme 6618 Relay TDR	SV-4947 TDR Relay Sch 6618 Fails To Energize
Scheme 6618 Relay TDR	TDR Relay Sch 6618 Fails To Remain Energized
P-98	Portable Pump P98 Fails To Run
P-98	Portable Pump P98 Fails To Start
P-98	Portable Pump P98 Stand-By Failure
TRS-6618-1	TRS-6618-1 Fails To Remain Energized or In Contr.Rm Position
TRS-6618-1	TRS-6618-1 Stand-By Failure T30-56
Breaker 2D-14	Breaker 2D-14 (ASD Panel C-31) Stand-By Failure
Breaker 2D-14	Breaker 2D-14 (ASD Panel C-31 Feed) Fails To Remain Closed

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Emergency Power (EP)	
Fussell-Vesely = 1.5E-2	**Birnbaum = 4.7E-5**
SDG and EDG	SDG and EDG Common Cause Failure To Run
SDG and EDG	SDG and EDG Common Cause Failure To Start
SDG	SDG Independent Failures
EDG	EDG Independent Failures
Enclosure Spray (ES)	
Fussell-Vesely = 4.4E-3	**Birnbaum =8.0E-5**
VPI-310	VPI-310 Fails To Open (Drum Enclosure Sprays)
VPI-312	VPI-312 Fails
MO-7064	MO-7064 Fails To Open
MO-7068	MO-7068 Fails To Open
MO-7064 Relay 42/O	MO-7064 Relay 42/O Fails To Energize
Fire Protection (FP)	
Fussell-Vesely = 1.2E-1	**Birnbaum = 1.7**
DFP Battery	DFP BATTERY A and B Common Cause Failure
Breaker 2B-26	Breaker 2B-26 (P-6) Fails To Remain Closed
Breaker 2B-26	Breaker 2B-26 (P-6) Out of Service
EFP Breaker	Breaker EFP Local (Panel C-17) Fails To Remain Closed
Breaker 2B-26	Breaker 2B-26 (P-6) Stand-By Failure
EFP Breaker	Breaker EFP Local (Panel C-17) Stand-By Failure
Breaker 1CB	Breaker 1CB (Panel C-09) Fails To Remain Closed

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Engine Control Switch	Engine Control Switch Fails To Remain Closed
Control Switch 1CS	Control Switch 1CS On KK Panel Stand-By Failure
Engine Control Switch	Engine Control Switch Stand-By Failure
VAR-8405	DFP Air Release Valve VAR-8405 Fails To Close
VAR-8405	DFP Air Release Valve VAR-8405 Fails To Remain Closed
VAR-8406	EFP Air Release Valve VAR-8406 Fails To Close
VAR-8406	EFP Air Release Valve VAR-8406 Fails To Remain Closed
VFP-304	Check Valve VFP-304 Fails To Close
VFP-304	Electric Fire Pump Check Valve VFP-304 Fails To Open
VFP-309	Check Valve VFP-309 Fails To Close
VFP-309	Diesel Fire Pump Check Valve VFP-309 Fails To Open
VFP-304 And VFP-309	Fire Pump Check Valves Common Cause Failure To Open
Relay FSOS	DFP Fuel Sol Fails To Energize
Relay FSOS	DFP Fuel Sol Fails To Remain Energized
Relay PNS1	DFP Pinion Solenoid PNS1 Fails To Energize
P-6	Electric Fire Pump P-06 Fails To Run
P-6	Electric Fire Pump Fails To Start
P-6	EFP Out of Service
P-7	Diesel Fire Pump P-07 Fails To Run
P-7	Diesel Fire Pump Fails To Start

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

P-7	DFP Out of Service
PS-612 and PS-615	FPS PS-612 & 615 Common Cause Failure To Close
PS-612	DFP Pressure Switch PS-612 Fails To Close
PCV-4515	DFP Cooling Water PCV-4515 Fails To Remain Open
PCV-4515	DFP Cooling Water PCV-4515 Plugged
Relay 1CR	Relay 1CR Fails To Energize
Relay 2CR	Relay 2CR Fails To Energize
Relay 4CR	RELAY 4CR Fails To Energize
Relay A1	EFP Start Relay A1 Fails To Energize
Relay CR	Electric Fire Pump Start Relay CR Fails To Energize
Relay SR1	DFP Fuel Relay SR1 Fails To Energize
RV-5062	RV-5062 DFP 4 Inch Relief Valve Fails To Close
RV-5062	RV-5062 DFP 4 Inch Relief Valve Fails To Remain Closed
T-53	DFP Fuel Tank T-53 Empty
VFP-013	VFP-013 L.O. Plugged Screenhouse Out
VFP-033	VFP-33 Fails To Open FPS Hotwell Fill Line
Fire Water Makeup to Hotwell (FH)	
Fussell-Vesely = 9.6E-3	**Birnbaum = 8.0E-5**
MO-7073	MO-7073 Fails To Open FPS Hotwell Fill Line
MO-7074	MO-7074 Fails To Open FPS Hotwell Fill Line
Feedwater (FW)	
Fussell-Vesely = 4.8E-3	**Birnbaum = 2.0E-4**

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

VFW-9	Stop Check Valve VFW-9 Fails To Open
Breaker 52-103	Breaker 52-103 (P-8A) Fails To Energize
Breaker 52-105	Breaker 52-105 (P-8B) Fails To Energize
Breaker 52-103 and 52-105	CBS 52-103 & 52-105 For P8A&B Fails To Close On Pump Restart
Breaker 1A-43	Breaker 1A-43 (P-26A) Stand-By Failure
Breaker 2A-43	Breaker 2A-43 (P-26B) Stand-By Failure
VFW-300	VFW-300 Feed Pump P-8A Disch Check Valve Fails To Close
VFW-300	VFW-300 Feed Pump P-8A Disch Check Valve Fails To Open
VFW-301	VFW-301 Feed Pump P-8B Disch Check Valve Fails To Close
VFW-301	VFW-301 Feed Pump P-8B Disch Check Valve Fails To Open
VFW-304	Check Valve VFW-304 Fails To Open
VFW-305	Check Valve VFW-305 Fails To Close
VFW-300 and VFW-301	P-8A & B VFW-300 and 301 Common Cause Failure To Open
Scheme 2401 Fuse	Fuse Sch 2401 Stand-By Failure
Scheme 2402 Fuse	Fuse Sch 2402 Stand-By Failure
CV-4000	Feed Water Reg Valve CV-4000 Fails To Open
CV-4000	CV-4000 Feed Water Reg Valve Fails To Remain Open
Scheme 2401 Handswitch	P-8A Aux Oil Pump Hand Switch Sch 2401 Stand-By Failure
Scheme 2402 Handswitch	P-8B Aux Oil Pump Hand Switch Sch 2402 Stand-By Failure
Scheme 2401 Transformer	P-26A Circuit Transformer Stand-By Failure

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Scheme 2402 Transformer	P-26B Circuit Transformer Stand-By Failure
P-26A and B	Aux Oil P-26A and P-26B Common Cause Failure To Run
P-26A and B	Aux Oil P-26A and P-26B Common Cause Failure To Start
P-8A	P-8A Aux Oil Pump P-26A Fails To Run
P-8A	P-8A Aux Oil Pump P-26A Fails To Start
P-8A	P-8A Aux Oil Pump P-26A Out For Maintenance
P-8B	P-8B Aux Oil Pump P-26B Fails To Run
P-8B	P-8B Aux Oil Pump P-26B Fails To Start
P-8B	P-8B Aux Oil Pump P-26B Out For Maintenance
P-8A and B	Feedwater P-8A and P-8B Common Cause Failure To Run
P-8A	Feed Pump P-8A Fails To Run
P-8A	Feedwater Pump P-8A Fails To Start
P-8B	Feed Pump P-8B Fails To Run
P-8B	Feedwater Pump P-8B Fails To Start
PS-608A	Feed Pump Suction Pressure Switch PS-608A Fails To Open
PS-608B	Feed Pump Suction Pressure Switch PS-608B Fails To Open
PS-629-1	P-8A Lube Oil Pressure Switch PS-629-1 Fails To Close
PS-630	P-8A Lube Oil Pressure Switch PS-630 Fails To Close
PS-633-1	P-8B Lube Oil Pressure Switch PS-632-1 Fails To Close
PS-633	P-8B Lube Oil Pressure Switch PS-633 Fails To Close

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

PS-608A and B	Pressure Switches PS-608A&B Common Cause Failure To Open
PS-629-1 and -2	P-8A&B Lube Oil PS-629-1&2 Common Cause Standby Failure
P-8A Relay 162-1	Feed Pump P-8A Auxiliary Relay 162-1 Fails To De-Energize
P-8B Relay 162-2	Feed Pump P-8B Auxiliary Relay 162-2 Fails To De-Energize
Scheme 2401 Relay 42/HC	Relay 42/HC Sch 2401 Fails To Energize
Scheme 2402 Relay 42/HC	Relay 42/HC Sch 2402 Fails To Energize
Main Condenser (MC)	
Fussell-Vesely = 7.4E-4	**Birnbaum = 1.1E-5**
SV-4868	SV-4868 Fails To Remain Energized For CV-4030
Steam Seal Regulator	Steam Seal Regulator Loss of Function
RV-5007	RV-5007 Fails To Remain Closed Steam Supply To Turbine Seal
Turbine Seals	Turbine Steam Seals Degraded During Operation
Main Steam Isolation (MS)	
Fussell-Vesely = 2.9E-3	**Birnbaum = 1.8E-5**
CV-4106	CV-4106 Bypass Vlv Disc Pipe Warming Line Fails To Open
CV-4106	CV-4106 Bypass Valve Warming Line Out of Service
CV-4106	CV-4106 Bypass Valve Disc Pipe Warming Line Not Opened
CV-4014	Turbine Bypass Valve CV-4014 Fails To Open
S-4899	SV-4899 Fails To Remain Energized For CV-4104
SV-4916	SV-4916 Fails To Energize For CV-4106
SV-4932	SV-4932 Fails To Energize For CV-4014
MO-7050	MO-7050 Main Steam Isolation Valve Fails To Close

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Scheme 6605 Relay 42/1C	Relay 42/1C Sch 6605 Fails To Energize
Scheme 6605 Relay 42/1C	Relay 42/1C Sch 6605 Fails To Remain Energized
Primary Coolant SRV fails to reclose (PC-RV)	
Fussell-Vesely = 4.4E-3	**Birnbaum = 4.5E-7**
Steam Drum Relief Valves	Steam Drum Safety Relief Valves Fail To Close
Reactor Depressurization System (RD)	
Fussell-Vesely = 1.4E-1	**Birnbaum = 2.2E-1**
CV-4180 and CV-4181	CV-4180 and CV-4181 Common Cause Failure To Open
CV-4180, CV-4181 and CV-4182	CV-4180 & CV-4181 & CV-4182 Common Cause Failure To Open
RDS Isolation Valves	RDS Isolation Vlvs Lethal Shock Common Cause Failure To Open
CV-4180, CV-4181 and CV-4183	CV-4180 & CV-4181 & CV-4183 Common Cause Failure To Open
CV-4180 and CV-4182	CV-4180 and CV-4182 Common Cause Failure To Open
CV-4180, CV-4182 and CV-4183	CV-4180 & CV-4182 & CV-4183 Common Cause Failure To Open
CV-4180 and CV-4183	CV-4180 and CV-4183 Common Cause Failure To Open
CV-4181 and CV-4182	CV-4181 and CV-4182 Common Cause Failure To Open
CV-4181, CV-4182 and CV-4183	CV-4181 & CV-4182 & CV-4183 Common Cause Failure To Open
CV-4181 and CV-4183	CV-4181 and CV-4183 Common Cause Failure To Open
CV-4182 and CV-4183	CV-4182 and CV-4183 Common Cause Failure To Open
Breaker 1Y-26	Breaker 1Y-26 (Panel C-40) Fails To Remain Closed
Breaker CB-7	Breaker CB-7 (Ups A) Stand-By Failure
Fuse CB-9A	Fuse UPS CB-9A Stand-By Failure
Fuse CB-9C	Fuse UPS CB-9C Stand-By Failure

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

SV-4180 and SV-4181	SV-4180 and SV-4181 Common Cause Failure To Energize
SV-4180, SV-4181 and SV-4182	SV-4180 & SV-4181 & SV-4182 Common Cause Failure To Energize
RDS Isolation Valves	RDS Iso Valves Lethal Shock Common Cause Failure To Energize
SV-4180, SV-4181 and SV-4183	SV-4180 & SV-4181 & SV-4183 Common Cause Failure To Energize
SV-4180 and SV-4182	SV-4180 and SV-4182 Common Cause Failure To Energize
SV-4180, SV-4182 and SV-4183	SV-4180 & SV-4182 & SV-4183 Common Cause Failure To Energize
SV-4180 and SV-4183	SV-4180 and SV-4183 Common Cause Failure To Energize
SV-4181 and SV-4182	SV-4181 and SV-4182 Common Cause Failure To Energize
SV-4181, SV-4182 and SV-4183	SV-4181 & SV-4182 & SV-4183 Common Cause Failure To Energize
SV-4181 and SV-4183	SV-4181 and SV-4183 Common Cause Failure To Energize
SV-4182 and SV-4183	SV-4182 and SV-4183 Common Cause Failure To Energize
SV-4984 and SV-4985	SV-4984 and SV-4985 Common Cause Failure To Energize
SV-4984, SV-4985 and SV-4986	SV-4984 & SV-4985 & SV-4986 Common Cause Failure To Energize
RDS isolation Valves	RDS Iso Valves Lethal Shock Common Cause Failure To Energize
SV-4984, SV-4985 and SV-4987	SV-4984 & SV-4985 & SV-4987 Common Cause Failure To Energize
SV-4984 and SV-4986	SV-4984 and SV-4986 Common Cause Failure To Energize
SV-4984, SV-4986 and SV-4987	SV-4984 & SV-4986 & SV-4987 Common Cause Failure To Emergize
SV-4984 and SV-4987	SV-4984 and SV-4987 Common Cause Failure To Energize
SV-4985 and SV-4986	SV-4985 and SV-4986 Common Cause Failure To Energize
SV-4985, SV-4986 and SV-4987	SV-4985 & SV-4986 & SV-4987 Common Cause Failure To Energize

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

SV-4985 And SV-4987	SV-4985 and SV-4987 Common Cause Failure To Energize
SV-4986 and SV-4987	SV-4986 and SV-4987 Common Cause Failure To Energize
Relay K7	Fire Pump Start Circuit AC3 Relay K7 Fails To Energize
Relay RF	RDS Fire Pump Start Relay RF Fails To De-Energize
SV-4984 and SV-4985	SV-4984 and SV-4985 Common Cause Failure To Open
SV-4984, SV-4985 and SV-4986	SV-4984 & SV-4985 & SV-4986 Common Cause Failure To Open
RDS Depressurization Valves	RDS Dpres Valves Lethal Shock Common Cause Failure To Open
SV-4984, SV-4985 and SV-4987	SV-4984 & SV-4985 & SV-4987 Common Cause Failure To Open
SV-4984 and SV-4986	SV-4984 and SV-4986 Common Cause Failure To Open
SV-4984, SV-4986 and SV-4987	SV-4984 & SV-4986 & SV-4987 Common Cause Failure To Open
SV-4984 and SV-4987	SV-4984 and SV-4987 Common Cause Failure To Open
SV-4985 and SV-4986	SV-4985 and SV-4986 Common Cause Failure To Open
SV-4985, SV-4986 and SV-4987	SV-4985 & SV-4986 & SV-4987 Common Cause Failure To Open
SV-4985 and SV-4987	SV-4985 and SV-4987 Common Cause Failure To Open
SV-4986 and SV-4987	SV-4986 and SV-4987 Common Cause Failure To Open
Panel C-40	General Fault - No Signal Generated From The C-40 Panel
Actuation Cabinet AC-1	No Signal From Actuation Cabinet AC-1
Actuation Cabinet AC-3	No Signal From Actuation Cabinet AC-3
Actuation Cabinet AC-3	RDS Actuation Cabinet AC-3 Out of Service
TL-3180, TL-3181 and TL-3182	TL-3180 & TL-3181 & TL-3182 Common Cause Failure To Function

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

RPV Low Level Transmitter	RDS RPV Vessel Low Level Transmitter Lethal Shock Common Cause Failure
TL.3180, TL-3181 and TL-3183	TI-3180 & TI-3181 & TI-3183 Common Cause Failure To Function
TL-3180, TL-3182 and TL-3183	TI-3180 & TI-3182 & TI-3183 Common Cause Failure To Function
TL-3181,TL-3182 & TL-3183	TI-3181 & TI-3182 & TI-3183 Common Cause Failure To Function
TL-3184 & TL-3185 & TL-3186	TI-3184 & TI-3185 & TI-3186 Common Cause Failure To Function
Steam Drum Level Transmitter	RDS Steam Drum Level Transmitter Lethal Shock Common Cause Failure
TL-3184, TL-3185 and TL-3187	TL-3184 & TL-3185 & TI-3187 Common Cause Failure To Function
TL-3184, TL-3186 and TL-3187	TL-3184 & TL-3186 & TL-3187 Common Cause Failure To Energize
TL-3185, TL-3186 and TL-3187	TL-3185 & TL-3186 & TL-3187 Common Cause Failure To Function
LT-3180-4	Level Transmitter LT-3180 To 83 Common Calibration Faults
LT-3184-7	Level Transmitter LT-3184 To 87 Common Calibration Faults
UPSA	UPSA Failures
SV-4984	SV-4984 Failures
CV-4181	CV-4181 Failures
SV-4985	SV-4985 Failures
CV-4182	CV-4182 Failures
SV-4986	SV-4986 Failures
CV-4183	CV-4183 Failures
SV-4987	SV-4987 Failures
SV-4987	CV-4184 Failures
RDS Nitrogen	Nitrogen Supply Failures
LT-3180	Level Transmitter LT-3180 Failures
LT-3181	Level Transmitter LT-3181 Failures
LT-3182	Level Transmitter LT-3182 Failures
LT-3183	Level Transmitter LT-3183 Failures
CV-4180	CV-4180 Failures

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Reactor Protection SystemLevel Elements (RPL)	
Fussell-vesely = 9.2E-4	**Birnbaum = 4.4E-3**
LE-RE08A-D	RPS LE-RE08A&B and RE09A-D Common Calibratic , Faults
PS-RE09A-D	RPS Pressure Switches PS-RE09A-D Common Calibration Faults
LE-RE08A	Level Element LE-RE08A Loss of Function
LE-RE08B	Level Element LE-RE08B Loss of Function
Station Power (SP)	
Fussell-Vesely = 2.2E-2	**Birnbaum = 2.6E-2**
Panel 1Y	Panel 1Y Open Circuit
Panel 1Y	Panel 1Y Short To Ground
Panel 2Y	Panel 2Y Open Circuit
Panel 2Y	Panel 2Y Short To Ground
Panel 3Y	Panel 3Y Short To Ground
OCB-1126	Breaker OCB-1126 (Sta Pwr Trans No 1) Output Fails To Open
Breaker 2A-2B	Breaker 2A-2B (MCC-2A To MCC-2B) Fails To Remain Closed
Breaker 2A-2B	Breaker 2A-2B (MCC-2A To MCC-2B) Out of Service
Breaker 1D-43	Breaker 1D-43 (2400V Switchgear) Fails To Remain Closed
138 kv Line	Conditional Failure of The 138 kv Transmission Line (24 Hrs)
46 kv Line	Conditional Failure Of 46kv Line For LOSP
MCC-2B	Motor Control Center MCC-2B Faults
Breaker 1F-1E	Breaker 1F-1E Faults
OCB-1126 and OCB-199	OCB-1126 and OCB-199 Faults
Panel 2Y	Panel 2Y Faults
Panel C04	Panel C04 Faults

TABLE 13.1-4

SYSTEM GROUPING FOR IMPORTANCE MEASURES

Breaker 2B-23	Breaker 2B-23 Faults
Panel 2P	Power To Panel 2P Failures
Fuse 1307A, C and D	Fuse 1307A, C and D Failures
Fuse 1307 B, E and F	Fuse 1307B, E and F Failures
Bus 2	Bus 2 Failures
MCC-2A	Motor Control Center MCC-2A Faults
Bus 1F	Bus 1F Failures
OCB-1199	OCB-1199 Failures
Bus 1	Bus 1 Failures
OCB-2299	OCB-2299 Failures
OCB-1136	OCB-1136 Faults
OCB-1126/7726	OCB-1126/7726 Handswitch Failures
MCC-1	Motor Control Center MCC-1 Failures
ASD Power Supply (ASD)	
Fussell-Vesely = 2.3E-4	**Birnbaum = 1.2E-4**
Bus D12	Bus D12 Failures
Scheme 1610	Circuit 1610 OOS
ASD Battery Disconnect	No Power From ASD Battery Disconnect
Service Water (SW)	
Fussell-Vesely = 4.3E-4	**Birnbaum = 1.5E-4**
BS-5751	Service Water Basket Strainer BS-5751 Plugged
P-12A and B	CC Failures of SW Pumps P12-A&B To Run
P-12A and B	CC Failures of SW Pumps P12-A&B To Run
Liquid Poison (LP)	
Fussell-Vesely = 1.4E-4	**Birnbaum = 5.5E-5**
Annunciator 1.4-33	LPS Low Nitrogen System Pressure Annunciator Alarm
Breaker 2P-8	Circuit Breaker 2P-8 Fails to Remain Closed
VP-300	Check Valve VP-300 Fails to Open
Scheme 6401 Transformer	Transformer Scheme 6401 Fails
PS-607	Pressure Switch PS-607 Fails To Close

TABLE 13.4-1
KEY UNCERTAINTY ISSUES

Analysis Task	Characteristics	Potential Impact on Overall Results and Specific Sequence/Task	Description
Initiating Events			
	LOCA Frequency	High	The BWR baseline model currently uses the IPEM LOCA initiating frequencies (. E-2). The LOCA frequency is approximately a factor of 240 greater than the recently published BWROG upper bound mean values.
	Loss of Feedwater	Low	No events have occurred, one assumed.
	Loss of Main Condenser	Low	No events have occurred, one assumed.
	Spurious Bypass Valve Opening	Low	No events have occurred, one assumed.
	Loss of DC Power	Low	No events have occurred, one assumed.
System Models			
FPS	Fire pumps unavailability	Low	Unavailability values for EFP and DFP are over estimated due to technique for summing hours out of service.
RDS	RDS Target Rock valves demands	Low	One demand was assumed. Although this is accurate for the first demand, any subsequent demands are not accounted for. Only out of 4 valves is required for subsequent operation.
	Number of valves required for depressurization	High	Assume 3 out of 4 valves required for all transients except small above core and below core LOCA and bypass line interfacing system LOCA. In most transients 2 out of 4 or even out of 4 may be adequate for depressurization.
	No recovery to manually start fire pumps following low drum level signal	Low	A minimum of 2 minutes is available to start the fire pumps following the initiation of the RDS actuation logic. No credit is taken for this action, although the time frame is not long.

TABLE 13.4-1
KEY UNCERTA!NTY ISSUES

Analysis Task	Characteristics	Potential Impact on Overall Results and Specific Sequence/Task	Description
Initiating Events			
PIS	LOCA hose	Low	No credit is taken for the LOCA hose for cooling the core spray heat exchanger.
	PIS Recovery	High	No credit is taken for recovery actions for the PI system. Hours are available between the time of failure and when usage is required due to continued injection of water.
CS	Core Spray valve common cause failures	Low	The primary core spray valves and redundant core spray valves have common cause failure associated with one valve from each train. These valves have different operators, power supplies and locations. Refer to Table 3.4-2 for valve diversities.
	Core spray-manual actuation	Medium	During small and medium steam line breaks or medium above core LOCA, the temperature rise in the spent fuel pump and heat exchanger room, results in failure of the core spray pressure and level switches, due to the harsh environment should the enclosure sprays fail. It is also conservatively assumed that the reactor water level switches located in room 445 fail for the same reason. Therefore operator diagnosis and manual control of the core spray valves is necessary to ensure core cooling for these LOCA initiators. Re-evaluation of the uncertainty in the EEQ analysis could change this assumption.
	Emergency Condenser makeup	Medium	No credit is taken for makeup from the portable pump except in station blackout and internal flooding events. Greatest benefit would be seen in loss of instrument air transients.

TABLE 13.4-1
KEY UNCERTAINTY ISSUES

Analysis Task	Characteristics	Potential Impact on Overall Results and Specific Sequence/Task	Description
Initiating Events			
	Service water filter plugging	Low	.5E-4 currently used. The Birnbaum measure is 2.3E-4 indicates that the CDF would not change appreciably if the event is set equal to .0. Current practices should be continued.
	Load Rejection event tree heading	Low	In loss of the 38 kv line sequences, it is assumed that the plant responds to a load rejection successfully 90% of the time. The successful response to a load rejection is more important for plant availability than it is for core damage. Due to the availability of a second offsite power source and an emergency condenser that is ac power independent, the effects of a failure to respond to a load rejection are minimized.
	Safety Relief valves demands	Low	The number of demands on the valves was estimated at 70. The Birnbaum importance measure of 4.5E-9 indicates the potential impact of this event is extremely low.
	Fill the Ball	Medium	Because of the BRP large dry containment and large volume-to-core power ratio, the many passive heat sinks may preclude *any* core damage during this evolution. However, because of the uncertainties involved in this method of core cooling, it has been assumed that the fill-the-ball strategy is unable to provide adequate core cooling. It has been incorporated into the LOCA event trees as a potential mitigative action, although *failure is always* assumed to occur.

TABLE 13.4-1
KEY UNCERTAINTY ISSUES

Analysis Task	Characteristics	Potential Impact on Overall Results and Specific Sequence/Task	Description
Initiating Events			
	Instrument Air	Low	Now modeled as of 3 required based upon final approval of the modification to install a separate (4th) air compressor dedicated to service air requirements. No credit was taken for supplying instrument air with the new air compressor.
Human Actions			
RM/RDS-MAN	Conditional HEP to control pressure.	Low	If emergency condenser unavailability is due to mechanical failures, the HEP for pressure control using RDS is quantified as conditional on the HEP for actuating EC, rather than the HEP value for initiating manual RDS.
Data			
	Surveillance test frequency used to estimate equipment demands.	Low	Equipment demands most likely underestimated as any demands outside of documented surveillance testing was not counted.
	Unavailability estimates high.	Low	Time out of service from all data sources were summed to determine component unavailability. This summation included some double counting of events.
Containment Analysis			

TABLE 13.4-2

CORE SPRAY VALVE DIVERSITY

Valve #	MO-7051 & MO-7061	MO-7070 & MO-7071
Valve Size	4.0 inch	4.0 inch
Valve Type	Gate	Gate
Disk Type	Flex	Flex
Packing Material	Graphite	Graphite
Wedge/Seat Material	Stellite	Stellite
Actuator Manufacturer	Limitorque	Rotork
Actuator Model	SMB-OO	4NA
Power Supply	25 vdc	ac Bus 2B
Location	Recirc pump room	External steam drum enclosure wall

TABLE 13.7-1

NRC IDENTIFIED PARAMETERS FOR SENSITIVITY STUDY (NUREG-1335)

- Performance of containment heat removal systems during core meltdown accidents.

- In-vessel phenomena (primary system at high pressure)

  - H_2 production and combustion in containment
  - Induced failure of the reactor coolant system pressure boundary
  - Core relocation characteristics
  - Mode of reactor vessel melt-through

- In-vessel phenomena (primary system at low pressure)

  - H_2 production and combustion in containment
  - Core relocation characteristics
  - Fuel/coolant interactions
  - Mode of reactor vessel melt-through

- Ex-vessel phenomena (primary system at high pressure)

  - Direct containment heating concerns
  - Potential for early containment failure due to pressure load
  - Long-term disposition of core debris (coolable or not coolable)

- Ex-vessel phenomena (primary system at low pressure)

  - Potential for early containment failure due to direct contact by core debris
  - Long-term core concrete interactions:

    -- Water availability
    -- Coolable or not coolable

TABLE 13.7-2

LIST OF SENSITIVITY ITEMS

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined in BRP PRA Response	Proposed Cases for Accident Management Investigations
In-Vessel Core Melt Progression			
- Hydrogen Production	X	X	
- Temperature of Melt			
- Model for Control Rods			
- Model for Candling			
- RPV Breach Model and Assumptions	X	X	
- In-Vessel Steam Explosion			
- Induced Primary System LOCAs	X		
- In-Vessel Recovery			X
- In-Vessel Reactivity Excursion			X
Ex-Vessel Core Melt Progression			
- Debris Temperature	X	X	
- Amount of Debris Discharged From Vessel		X	
- Sump Coolability		X	
- Coolability with Water Present	X	X	
- Effective DW Floor Area		N/A	

X - Identifies sensitivity cases satisfying the column heading

N/A - Not applicable to BRP

TABLE 13.7-2

LIST OF SENSITIVITY ITEMS
(continued)

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined in BRP PRA Response	Proposed Cases for Accident Management Investigations
- Pool Bypass		N/A	
-- Vacuum Breaker		N/A	
-- Downcomers		N/A	
-- Other		N/A	
- Quenching Model in Pool (MKII)	X	N/A	
- DCH		X	
- Amount of Material		X	
-- Retained in Enclosure		X	
-- Retained in Cavity		X	
Containment Failure			
- Size	X	X	
- Location	X	X	
- Pressure (Ultimate Capability)		X	
- Temperature			
- ATWS Induced Dynamic Containment Failure Mode			
- Containment Venting			X

X - Identifies sensitivity cases satisfying the column heading

N/A - Not applicable to BRP

TABLE 13.7-2

LIST OF SENSITIVITY ITEMS
(continued)

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined in BRP PRA Response	Proposed Cases for Accident Management Investigations
- Pool Bypass		N/A	
- Aerosol Plugging			
- Direct Contact of Debris	X	N/A	
- Pressure Rise	X	X	
Reactor Building Effectiveness	X	N/A	
- Hydrogen Burn			
- Circulation Established			
- Direct Release			
Critical Safety Functions			
- Reactivity Control			
- Pressure Control			X
- High Pressure Makeup			X
- Depressurization			X
- Low Pressure Makeup			X
- Containment Heat Removal	X		X
- Containment Temperature Control			X
- Containment Pressure Control			X

X - Identifies sensitivity cases satisfying the column heading

N/A - Not applicable to BRP

TABLE 13.7-2

LIST OF SENSITIVITY ITEMS
(continued)

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined in BRP PRA Response	Proposed Cases for Accident Management Investigations
- Combustible Gas Control			X
- Containment Water Level Control			X
- Containment Flooding		X	X
- Containment Spray Use		X	X
Other Actions			
- Accident Management Actions			X
- Disregard DWSI Curve			X
- Containment Flood Always by Procedure		X	
- Containment Flood With No RPV Vent		N/A	X
- Containment Flood Only Late in Sequence		X	X
- Fill DW With Water (MKI)		N/A	
- Vent to 0 psig			X
- Vent to Control 40 to 60 psig			X

X - Identifies sensitivity cases satisfying the column heading

N/A - Not applicable to BRP

TABLE 13.7-3

DEBRIS RETAINED IN RPV

				Containment Conditions VF + 36 Hours				CsI Fraction at VF + 36 Hours	
MAAP Case	Assumed FMAXCP	Containment Sprays	Core Fraction Long Term in RPV	Failure Time (hr)	Sphere Temp (F)	Sphere Pressure (psia)	RPV Temp (F)	Remaining in RPV	To Env (Leakage)
1	0.1	No	0.52	N/A	240	39.4	672	0.4	.194E-3
1a	0.6	No	0	N/A	244	42.1	494	0.38	.154E-3
2	0.1	Yes	0.49	N/A	227	36.6	551	0.35	.757E-5
2a	0.6	Yes	0	N/A	231	38.4	433	0.32	.685E-5
4	0.1	No	0.52	4.07	158	14.7	797	0.32	.125E-0

TABLE 13.7-4

IN-VESSEL HYDROGEN GENERATION

			H_2 Production (pounds)		Containment Conditions VF + 36 Hours		CsI Release Fraction to Env
MAAP Case	Accident Description	FCRBLK (Blockage)	In-Vessel	CCI	Sphere Temperature (F)	Sphere Pressure (psia)	
15	Transient with no RDS No injection	0 (Local)	123.0	0.0	258	45.6	.302E-3
15c	Transient with no RDS No injection	-1 (None)	295.0	105.0	274	50.2	.223E-3

TABLE 13.7-5

RPV PRESSURE AT VESSEL FAILURE
(WITH AND WITHOUT CONTAINMENT SPRAYS)

MAAP Case	RDS	Containment Sprays	Containment Conditions VF + 36 Hours: Failure Time	Containment Conditions VF + 36 Hours: Pressure (psia)	Containment Conditions VF + 36 Hours: Temp (F)	RPV Failure: Time (hours)	RPV Failure: Vessel Pressure (psia)	RPV Failure: Sphere Pressure at VF	CsI Release Fraction at VF + 36 Hours: To Env
15d	No	Yes	N/A	38	230	8.94	1480 (High)	31.8	.254E-4
2	Yes	Yes	N/A	36.6	227	4.37	41.7 (Low)	34.5	.760E-5
15	No	No	N/A	45.6	258	8.76	1470 (High)	41.6	.302E-3
1	Yes	No	N/A	39.4	240	4.09	47.1 (Low)	41.2	.194E-3
15a	No	Yes [1]	N/A	34.8	230	8.75	1530 (High)	40.3	.307E-3

[1] Introduced at floor level (i.e. no fission product scrubbing)

TABLE 13.7-6

HIGH PRESSURE CASES WITH FORCED EARLY FAILURE

MAAP Case	Description	Peak Sphere Pressure (Time)	CsI Release Fraction at VF + 36 Hours	Containment Failure		
				Time (hours)	Pressure (psia)	Temp (F)
15	High pressure vessel failure followed by normal leakage - No sprays	45.0 psia (38.7 hrs)	.302E-3	N/A	N/A	N/A
14	High pressure vessel failure with forced containment failure at VF - No sprays	37.0 psia (8.76 hrs)	.172E-0	8.76	37	320
14a	Same as 14, but with containment sprays operational	26.6 psia (8.97 hrs)	.121E-1	8.96	16.9	131

TABLE 13.7-7

EX-VESSEL DEBRIS COOLABILITY -
CORE-MELT PROGRESSION WITH WATER AVAILABLE TO DEBRIS (EX-VESSEL)

	Base Case 2 (Recirc)	Base Case 3 (Fill the Ball)	Sensitivity Case 11 (Recirc)	Sensitivity Case 11a (Recirc)	Sensitivity Case 12 (Fill the Ball)
RDS	Yes	Yes	Yes	Yes	Yes
Switch to Recirc (hrs)	6.23 (before 587')	No	6.2 (before 587')	6.13 (before 587')	No
FCHF	0.09	0.09	0.01	0.01	0.01
H_2 Generation: In-Vessel (lbs)	42	42	42	46	42
H_2 Generation: CCI (lbs)	< .001	< .001	63	124	63
Core Fraction Retained In-Vessel	0.48	0.48	0.48	0	0.48
CRD Room Concrete Attack Depth (ft) VF + 36 Hours	3.95×10^{-4}	3.95×10^{-4}	1.5	1.6	1.49
CRD Room Temperature (F) VF + 36 Hours	302	227	300	271	209
Pressure in CRD Room (psia) VF + 36 Hours	36.6	15.4	37.6	39.5	15.9
CsI Release Fraction to Env	.757E-5	.632E-5	.798E-5	.749E-5	.655E-5

TABLE 13.7-8

DEBRIS COOLING IN THE SUMPS (NO SPRAYS)

MAAP Case	Description	CRD Concrete Attack (ft)	Junction 9 Flow	CRD Room Peak Corium Temp	CRD Room Corium Temp VF + 36 Hours
1	Transient with RDS	0	No	1600°F (16.0 hrs)	1320°F (40.1 hrs)
1a	Transient with RDS and 100% core on floor	0.17	Corium from CRD to Enclosure (13.8 hrs)	3600°F (13.8 hrs)	1800°F (40.1 hrs)
4	Transient with RDS and Early Forced Fail	1.55	Water from Enclosure to CRD (18.9 hrs)	3200°F (18.9 hrs)	2350°F (40.1 hrs)

TABLE 13.7-9

CONCRETE ATTACK DUE TO MOLTEN DEBRIS TRANSIENTS WITH RDS

	Case 1: Base Case	Case 1a: Core on the Floor	Case 3: Containment Sprays (Cooling)	Case 11a: Sprays, Core on the Floor (No Cooling)	Case 1c: Core on the Floor (Junction 9 Blocked)
Concrete Attack CRD Room (ft^2)	0	0.17	< .001	1.6	2.15
Concrete Attack Enclosure Room (ft^2)	0	0.29	0	1.2	0
Core Fraction In-Vessel	0.52	0	0.48	0	0
Core Fraction in CRD Room	0.48	0.67	0.52	0.54	1
Core Fraction in Enclosure Room [1]	0	0.33	0	0.46	0

[1] At Vessel Failure + 36 hours

TABLE 13.7-10

DEBRIS SPREADING IN HIGH PRESSURE VESSEL FAILURE CASES
TRANSIENT WITH NO RDS, NO CONTAINMENT SPRAYS

	Case 14: Early Failure	Case 15: Base Case	Case 14b: Early Failure No Entrainment	Case 15b: Base Case No Entrainment
Containment Failure Time (hours)	8.76	N/A	8.78	N/A
CRD Room Dryout (hours)	11	N/A	12.5	15
Core Fraction In-Vessel	0.26	0.26	0.26	0.26
Core Fraction in CRD Room	0.17	0.21	0.56	0.61
Core Fraction in Enclosure Room	0.57	0.53	0.18	0.13
Concrete Attack in CRD Room	< .001	< .001	1.37	1.05
Concrete Attack in Enclosure Room	1.15	0.1	< .001	0
CsI Release Fraction VF + 36 Hours	.172E-0	.302E-3	.190E-3	.312E-3

TABLE 13.7-11

CONTAINMENT FAILURE AREA

	Containment Failure			CsI Release	Release Characteristic	
MAAP Case	Size	Location	Time (hrs)	To Env	Magnitude	Timing
4	3.14 ft²	Sphere to Env	4.07	0.13	Medium	Early
4b	10 ft²	Sphere to Env	4.08	0.19	Medium	Early

TABLE 13.7-12

CONTAINMENT FAILURE TYPES

	Containment Failure			Sprays	CsI Release Fraction	Release
MAAP Case	Size	Type	Time	Available	To Env	Magnitude
2	0.5% per day	Normal Leakage	4.3	Yes	.757E-5	Negligible
1	0.5% per day	Normal Leakage	4.1	No	.194E-3	Low-Low
8	3.14 ft²	Isolation Failure	0	Yes	.353E-2	Low
6	3.14 ft²	Isolation Failure	0	No	.701E-1	Moderate
4a	3.14 ft²	Early Failure	4.4	Yes	.327E-1	Moderate
4	3.14 ft²	Early Failure	4	No	.125E-0	Moderate

TABLE 13.7-13

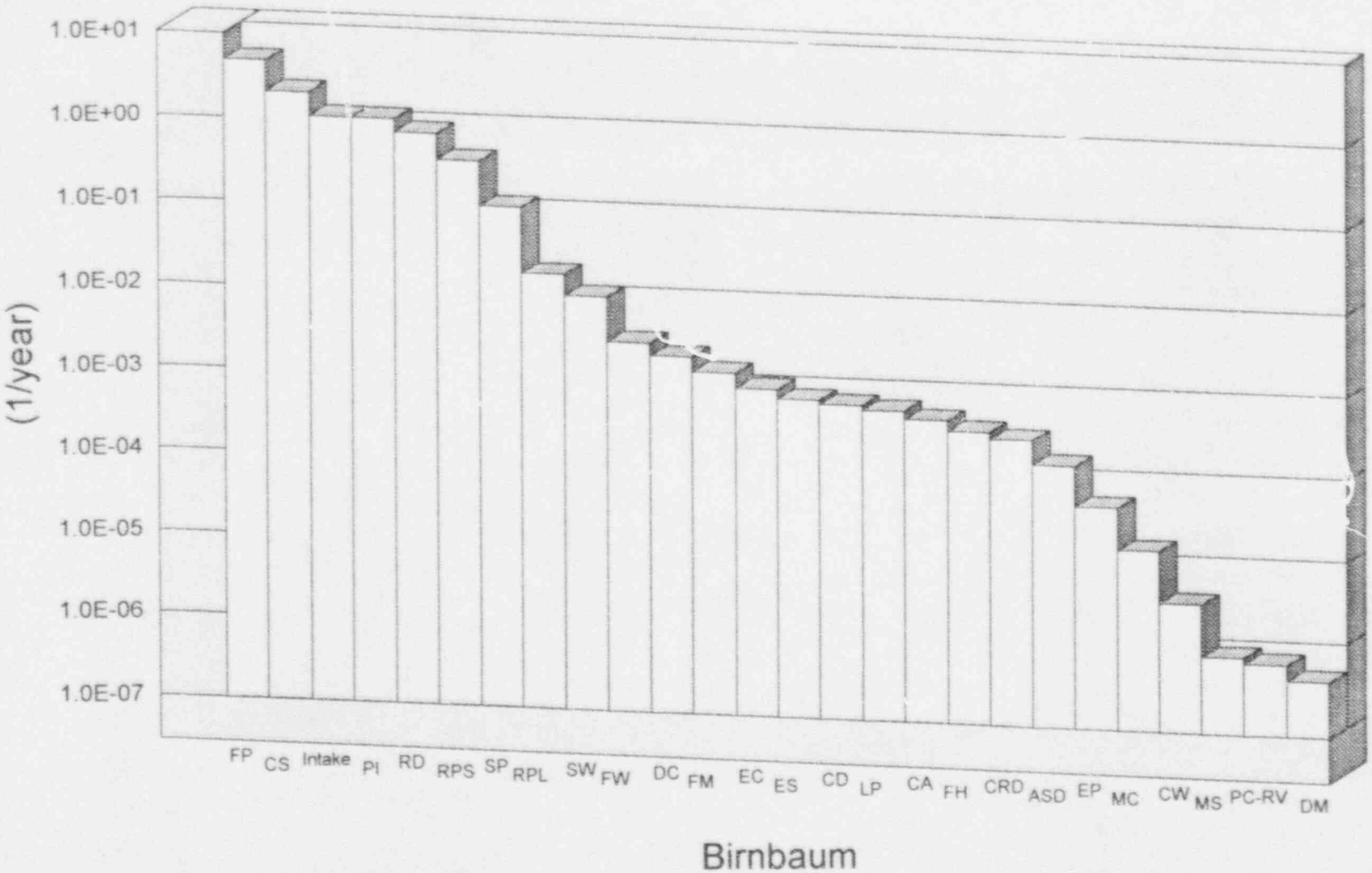
CONTAINMENT FLOODING SCENARIOS TRANSIENTS WITH RDS

			Containment Conditions VF + 36 Hours			CsI Release Fraction
MAAP Case	Recirc	VF	Pressure (psia)	Sphere Temp (F)	Water Level (ft)	
Case 3: Fill the Ball	No	4.37	15.4	100	589.3	.632E-5
Case 3b: Increased Flow	No	4.4	18	82	619.2	.621E-5
Case 2: Switch to Recirc	6.23 hours (before 587')	4.37	36.6	227	578.1	.757E-5

Key For Section 13.1 Figures
Plant Damage Descriptions

CW	Circulating Water
INTAK	Traveling Screens
PC	SRV Open
PI	Post Incident
CA	Incident Air
CD	Condensate
CRD	Control Rod Drive
CS	Core Spray
DC	DC Power
EV	Emergency Condenser Valves
DM	Demin Water Makeup
FM	Emergency Condenser Fire Pump Makeup
EP	Enclosure Power
ES	Enclosure Spray
FP	Fire Protection
FH	Fire Water Makeup to Hotwell
FW	Feedwater
MC	Main Condenser
MS	Main Steam Isolation
PC-RV	SRV Fail to Reclose
RD	Reactor Depressurization
RPL	Reactor Level Instrumentation
RPS	Reactor Protection
SP	Station Power
ASD	ASD Power Supply
SW	Service Water
LP	Liquid Poison

Figure 13.1-1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1.0E+00
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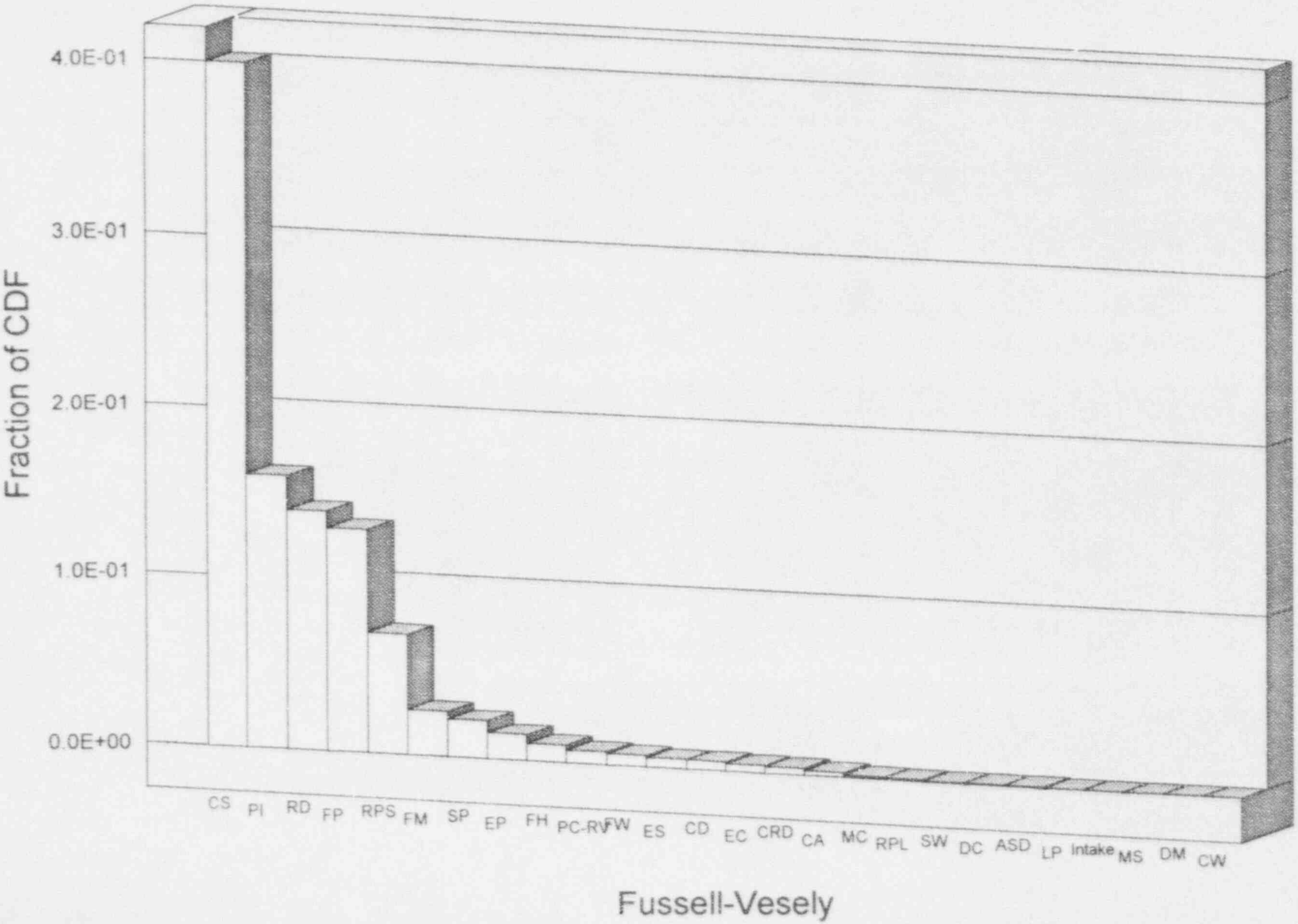
1.0E-02
1.0E-03
1.0E-04
1.0E-05
1.0E-06
1.0E-07
(1/year)
FP
CS
Intake
PI
RD
RPS
SP
RPL
SW
FW
DC
FM
EC
ES
CD
LP
CA
FH
CRD
ASD
EP
MC
CW
MS
PC-RV
DM
Birnbaum

Figure 13.1-2



Fraction of CDF
4.0E-01
3.0E-01
2.0E-01
1.0E-01
0.0E+00
CS
PI
RD
FP
RPS
FM
SP
EP
FH
PC-RV
FW
ES
CD
EC
CRD
CA
MC
RPL
SW
DC
ASD
LP
Intake
MS
DM
CW
Fussell-Vesely

Figure 13.1-3
Big Rock Point PRA
System Importance to Core Damage