H. L. Sumner, Jr.
Vice President
Hatch Project

Southern Nuclear Operating Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279

SOUTHERN COMPANY
Energy to Serve Your World℠

September 26, 2005

Docket Nos.: 50-321
50-366

NL-05-1515

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Response to Request for Additional Information on DC Sources
Technical Specifications Submittal

Ladies and Gentlemen:

Attached you will find the SNC response to an NRC Request for Additional Information (RFAI) on the DC Sources Technical Specifications amendment request (TSTF-360) submitted on July 20,2004.

The RFAI was received by facsimile transmission on April 27, 2005.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

H. L. Sumner, Jr.

HLS/OCV/daj

Enclosures: Enclosure 1 - Questions and Responses
Enclosure 2 - Peer Review F&O Comments and Observations

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

Enclosure 1

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

Enclosure 1

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

NRC Question:

Enclosure 6 discusses the industry peer review performed at Plant Hatch. Provide additional information as shown below:

(a) Please provide the date peer review was performed and the certification date for the applicable Plant Hatch probabilistic risk assessment (PRA) revision.

In addition, additional information on PRA quality is requested for the proposed amendment in the following areas:

(b) The plant specific PRA reflects the as-built as-operated plant. Enclosure 4 states that the Plant Hatch PRA revision is Revision 1. Confirm the revision number and date. Also, confirm that the PRA has been maintained and represents the current plant configuration, plant operating history, and component failure data, and is complete with respect to evaluating the proposed battery completion time (CT) extension.

(c) Discuss PRA updates including any individual plant examination (IPE) individual plant examination of external events (IPEEE) findings/improvements cable rerouting modifications credited in the fire analysis but not implemented.

(d) Specifically identify the A and B facts and observations (F&Os) identified during the peer review and their final disposition. Also, indicate any F&Os related to the proposed station service battery extended CT amendment request and their resolution.

(e) Enclosure 6 states that the comments are preliminary.

(f) Provide a summary of PRA quality assurance programs and applicable procedures including appropriate references.

SNC Response:

(a) The Plant Hatch Probabilistic Risk Assessment (PRA), Revision 1 peer review was performed during the week of December 4, 2000. The date of the certification of Plant Hatch PRA, Revision 1 was April 11, 2001.

(b) The application for the station service battery Allowed Outage Time (AOT) extension was based on the analysis using the Plant Hatch PRA, Revision 1a, dated May 25, 2001. Periodic updates are performed for the Plant Hatch PRA. These updates include model changes to reflect modifications to the plant configuration, operating history, and component failure data. The Revision 1 model has incorporated the then current plant configuration and operating history in terms of the initiating event frequencies. The plant-specific component failure data were updated in Revision 2 of the Plant Hatch PRA. Although the Revision 1a PRA model did not encompass the most current component failure data, it used, however, the Hatch-specific component failure data. As such, the Revision 1a model is complete with respect to evaluating the proposed station service battery completion time extension. The PRA has been reviewed for plant changes made up to and including the Unit 2 2005 Refueling Outage. Design changes and modifications made have had negligible affect on the PRA. There have been no changes to the 125/250 VDC Battery Systems at Plant Hatch that have necessitated any change to the calculations performed for this requested AOT extension.

(c) There are no updates or plant modifications credited in the PRA which have not been implemented. The IPEEE cable reroutes referenced in this RAI were addressing the resolution of the ThermoLag issue. This has been resolved and is considered complete.

(d) The A and B Peer Certification F&Os are included with disposition in ATTACHMENT A to this document. These issues are added to the Revision 2 PSA model. The physical availability and reliability of the battery systems are not affected. The unattended (no chargers) battery time for the A Station Service Battery bank was extended to 5 hours. This reduced conservatism in the Revision 1a model with regard to Station Blackout.

(e) The comments in Enclosure 6 are not in themselves preliminary. This is meant to imply that at the time of the writing the commentary was yet to be applied to a new PRA model revision.

(f) The Plant Hatch-specific Level 1 and Level 2 Probabilistic Risk Assessment (PRA) Model, Revision 1a, was used to evaluate the impacts on plant risk of the extension of the allowed outage time for the Station Service Battery. This model, when used in conjunction with deterministic evaluations, is of sufficient quality to support regulatory applications such as that submittal, as described below. The associated PRA calculations performed as part of the development of this battery submittal were originated, verified, approved and documented in accordance with SNC procedures for the preparation and control of calculations.

As an integral part of its initial development pursuant to NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," the PRA was repeatedly reviewed by an Independent Review Group which included experts in plant design, plant operation, and probabilistic risk assessment. Further, each subsequent revision to the model has been internally reviewed and approved in accordance with applicable SNC procedures. In addition, an evaluation based upon Appendix B of the EPRI PSA Applications Guide was performed to confirm that the PRA conforms to the industry state-of-the-art practices with respect to the scope of potential plant scenarios.

As noted in the response to the response to inquiry 1a) above, the Revision 1a of the Hatch PRA was extensively reviewed by an experienced Peer Review Team coordinated by the BWR Owners Group in a manner described in the Nuclear Energy Institute's document NEI 00-02, "Industry Peer Review Process." The peer review evaluated the eleven elements of the PRA and concluded that all elements were either a "Grade 3" or a "Contingency Grade 3." A "Grade 3" is defined in the Peer Review Process as:

> "This grade extends the requirements *[of previously defined Grades 1 and 2]* to assure that the risk significance determinations made by the PRA are adequate to support regulatory applications, when combined with deterministic insights. Therefore, a PRA with elements determined to be at Grade 3 can support physical plant changes when it is used in conjunction with other deterministic approaches that ensure that defense-in-depth is preserved. Grade 3 is acceptable for Grade 1 and 2 applications, and also for assessing safety significance of equipment and operator actions. This assessment can be used in licensing submittals to the NRC to support positions regarding absolute levels of safety significance if supported by deterministic evaluations."

Three PRA elements were judged by the peer review to have findings that resulted in their being considered "Contingency Grade 3." A "Contingency Grade 3" reverts to a "Grade 3" when items noted in the evaluation of the element are resolved. Such pending items are classified as one of four degrees of significance. None of the pending items noted in the Plant Hatch PRA evaluation were judged to be of a level of significance to require prompt resolution to ensure the technical adequacy of the PRA (i.e., significance level "A"). Issues with Facts and Observations classified as significance level "B" [Important and necessary to address, but may be deferred until the next PRA update (Contingent Grading Item.)] are addressed in the response to inquiry 1 d) above.

NRC Question:

The IPE and the revision 1a of the Hatch PRA indicate that the loss of station battery A is a significant contributor to plant core damage frequency (CDF). Discuss the risk contributors, including any asymmetry, that makes station service battery A a significant contributor to plant CDF.

SNC Response:

The main difference between Station Service Battery Subsystem A and Station Service Battery Subsystem B is that the loss of Station Service Battery Subsystem A, defined as a loss of Direct Current (DC) voltage on bus S016 or DC cabinet S001 or S003, leads to an immediate turbine trip. The loss of S016, S001, or S003 leads to a turbine trip without an automatic opening of the generator output breakers, and the operators are required to open these output breakers using remote manual means within 30 seconds. Loss of the A Battery Subsystem also prevents auto transfer of station service 4160VAC buses to their alternate power supply after the turbine trip, which loses the main condenser and the condensate/feedwater system. The loss of station battery subsystem B does not result in a plant trip. As such, two initiating events are included in the Plant Hatch PRA to model the loss of Division I DC power: LODC and DCPAN. LODC represents the loss of station battery subsystem A DC power initiating event and DCPAN represents the loss of 125V DC panel R25S001 initiating event.

Of course, there are also some minor asymmetry in terms of loads supplied by the Station Service Battery Subsystems A and B. For example, power for Reactor Core Isolation Cooling (RCIC) is provided by Station Service Battery Subsystem A, while control power for High Pressure Coolant Injection (HPCI) and is fed by Station Service Battery Subsystem B.

Early loss of the main condenser as a heat sink and loss of RCIC make the A Station Service Battery worth more. The main condenser is not in itself as significant as RCIC is however, because of the ability to use RCIC during the SBO case.

NRC question:

External events are discussed in the submittal with the exception of high winds and tornados. Provide a discussion on these risk impacts with respect to the proposed extended station service battery allowed outage time (AOT). See Enclosure E4-14.

SNC Response:

High winds and tornados were previously evaluated as part of the response to the NRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." These analyses were performed using the progressive screening approach recommended by NUREG-1407. Plant Hatch can be affected by high winds (including tornados) if and when either the wind forces exceed the load capacity of those plant structures housing accident initiation/mitigation components or the missiles generated penetrate these structures and damage critical components or other contents inside the facilities. The frequency of wind loading exceeding the design capacity based on the most critical condition (i.e., a design basis tornado) was calculated to be 5.89E-8 per year, several orders of magnitude lower than the screening criterion of 1E-6 per year. The calculation of this wind loading exceedance frequency is based on the annual frequency of a tornado striking Plant Hatch with a windspeed greater than that of a design basis tornado. This frequency calculation is not affected in any way by the proposed extension of the station service battery AOT.

The tornado missile impact and damage frequencies presented in the Plant Hatch IPEEE were developed by scaling the results of a generic two-unit plant analyzed in an Electric Power Research Institute (EPRI) study. This analysis involved the evaluation of tornado occurrence frequency, tornado wind field, missile spectrum, missile generation and transport trajectory, plant layout, missile impact velocities, and potential damage to plant structures. The results of the analysis indicated that the tornado-missiles' contribution to damage to critical Plant Hatch components should be significantly less than the IPEEE screening criterion and is also unaffected by the proposed extension of the station service battery AOT.

Based on the preceding considerations, it is therefore concluded that the impacts on the risks associated with high winds and tornados with respect to the proposed extension of the station service battery should be very small and insignificant.

NRC Question:

Provide a discussion on cumulative risk as per the guidance of RG 1.173, Section 3.3.2, "Cumulative Risk". Include, for example, diesel generator amendment requests to extend diesel generator CT times to 14 days, extended power uprates and extended surveillance instrumentation CTs or surveillances. Additional CT or surveillance interval extensions should be discussed with respect to the proposed extended station service battery CT. Although the battery charger extended CT was evaluated deterministically, provide the results for ΔCDF, Δ large early release fraction (LERF), incremental conditional core damage probability (ICCDP), and incremental conditional large early release probability (ICLERP), and include the extended battery charger CT in the cumulative risk evaluation.

SNC Response:

The risk-informed amendments that Plant Hatch has submitted in the past include extended power uprate from 2558 CMWt to 2763 CMWt, extension of the completion time for inoperable emergency diesel generators to 14 days, and Technical Specification revision to extend the surveillance intervals (for the 24 month fuel cycle) for the instrumentation channel functional tests and channel calibrations from 92 days to 92 days on a staggered test basis. A licensed power increase was accomplished to take licensed thermal power to 2804 CMWt based on installation of the ADVANCED MEASUREMENT ANALYSIS GROUP, INC./WESTINGHOUSE CROSSFLOW ultrasonic flow measurement instruments on each plant unit. This was approved by the NRC in Amendments 238 and 110 to the Hatch Operating Licenses. Although not risk informed, the effects of this uprate are evaluated on the present PRA. The difference between 2763 CMWt and 2804 CMWt does not provide any appreciable change to the Hatch PRA. This battery AOT submittal is not affected by the increase in power between 2763 CMWt and 2804 CMWt. The current submittal includes the proposed, extended station service battery and charger AOTs. The following provides an account of these changes.

Extended Power Uprate

The extended power uprate increased the core thermal power from 2558 CMWt to 2763 CMWt. The most likely impacts of the extended power uprate on the PRA model include success criteria and operator action or recovery event timing. Based on a detailed review performed, the Level 1 success criteria from before the extended power uprate remain valid. For the grid recovery events, a reevaluation showed that the small time changes being considered did not alter the non-recovery probability values to a degree, which would cause a change in the Core Damage Frequency (CDF). The only operator action or event that is significantly affected by the extended power uprate is operator action failure event "DE4" (i.e., failure to depressurize with inadequate high pressure injection in non-ATWS sequences). Due to the shortened time window available for this operator action, the Human Error Probability (HEP) for this operator action increases from 5.16E-02 to 8.05E-02. The most significant CDF increases result from the medium Loss of Coolant Accident (LOCA), isolation of Plant Service Water discharge, loss of startup transformer D, and loss of main control room ventilation initiating events. With the exception of loss of main control room ventilation and medium LOCA, the increase in each sequence's CDF for other initiating events tends to be small. As a result, the change in CDF attributable to the extended power uprate is solely due to the increase in the value of the operator error probability for the DE4 depressurization action. The increase in CDF presented in the submittal (6.6% for Unit 1 and 4.1% for Unit 2) is based on a conservative and bounding HEP value for DE4; i.e., 1.032E-01 (twice the original IPE value of 5.16E-02). The change in the fire risk for Unit 1 is negligible. The change in Large Early Release Frequency (LERF) is considered insignificant (an approximate 1% change in CDF was noted for those sequences which were part of the LERF for each unit evaluated at the extended power uprate).

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

Extension of EDG Completion Times to 14 days

Application and Plant Elements Affected

This Technical Specification amendment extended the Completion Times for the Required Action associated with restoration of an inoperable Unit 1 or Unit 2 emergency diesel generator (EDG) to a maximum of 14 days (from 72 hours for the 1A, 1C, 2A, and 2C DGs and from 7 days for the 1B swing DG).

In addition, the extension of the Completion Time to 14 days for an inoperable DG results in a corresponding extension of the time period associated with discovery of failure to meet Limiting Conditions for Operation (LCO) 3.8.1 from 10 days to 17 days.

PRA Model Refinements

To support the evaluation of risk impact, Revision 1 of the Plant Hatch PRA model was revised to remove/lessen some of the conservatisms so that the results of the analysis would not be unduly pessimistic. These changes to the Revision 1 PRA model include:

- The following event combinations were added to the mutually exclusive event file to eliminate invalid cutsets:

XXOG_DEMAND	OPHEEPANOLINK
XXOG_DEMAND	OPHEEPA
XXOG_24HOURS	OPHEEPANOLINK
XXOG_24HOURS	OPHEEPA
MNUN1R43S001B	UOL1
MNUN1R43S001B	UOL3
MNUN1R43S001B	UOL24

  Operator actions OPHEEPA and OPHEEPANOLINK are associated with non (LOSP) events; operator actions OPHEEPB and OPHEEPBNOLINK are associated with the LOSP case. Diesel generator B cannot be aligned to the opposite unit to supply emergency power if it is in maintenance (i.e. UOL1, 3, or 24).

- A non-recovery factor of 0.4 was added to account for the possible recovery of diesel start failures. This involves the following basic events:

  CC-DGS-22, CC-DGS-23, CC-DGS-24, CC-DGS-25, CC-DGS-26, CC-DGS-27, CC-DGS-28, CC-DGS-36, CC-DGS-37, CC-DGS-38, CC-DGS-39, CC-DGS-40, CC-DGS-41, CC-DGS-42

Questions and Responses

This is to re-instate an IPE recovery factor that was removed for convenience in Revision 1a of the Hatch CAFTA PRA model.

- Grid non-recovery basic events GRA3 (0.21) and GRB3 (0.27) were used initially in place of GRA2&3 (0.27) and GRB2&3 (0.36), respectively, for the calculations of LERF.

  This was to restore the grid non-recovery factors used previously in IPE and conservatively simplified for convenience in the Revision 1a model.

- Based on engineering calculations, unattended (i.e., with no battery charger) station service batteries can support 5 hours of RCIC operation during a station blackout event. As such, if RCIC is available, it can operate on unattended battery power in an LOSP event for 5 hours. Basic Event DUR3 was therefore redefined to be 5 hours. With 5 hours of RCIC operation, the core damage and vessel failure times were also extended. Based on an analysis using data taken from NUREG/CR-5496, Basic Events DUR3 (LOSP events lasting between 30 minutes and 5 hours) and DUR24 (LOSP events lasting longer than 5 hours) were later re-calculated to be 0.3855 and 0.0964, respectively. In addition, the grid non-recovery factors GRA2&3 and GRB2&3 were re-evaluated to be 0.3538 and 0.4130, respectively. With the increased RCIC operating time because of the increased battery life, more time is available for the operator to connect either the 600VAC emergency Bus C or D to the 4.16kV F bus (powered by the B diesel generator) during a loss of power event (i.e., operator action OPHEEPB) when the other diesels fail. This led to a reduction in the HEP for OPHEEPB from 1.62E-2 to 5.9E-3 (to be of the same value as OPHEEPA).

  It must be noted that the above changes were made in the post-submittal evaluation and did not make it to the submittal or responses to RAIs.

- In recent years, with the installation of additional runback features to rapidly reduce reactor power via the recirculation system control, scrams on low vessel water level are greatly reduced when feedwater problems (such as loss of reactor feed pump suction pressure and loss of condensate booster pump suction pressure) occur. The Unit 1 600VAC emergency Bus C will allow the minimum flow valves for condensate and condensate booster pumps to open upon its loss. Since the installation of these runback features, a loss of the 600VAC Bus C on Unit 2 on March 8, 2001 did not result in a scram. In addition, the loss of 600VAC Bus C events (a loss of Unit 2 600VAC Bus C and scram on June 25, 1992 prior to the installation of additional runback features, a loss of Unit 2 600VAC Bus C and no scram on March 8, 2001, and a loss of Unit 1 600VAC Bus D on April 14, 1996) were all recovered in a few minutes. As such, a recovery factor was added (i.e., Basic Event %FL-BUSC was changed from 1.0 to 0.4).

- For the analysis of plant configurations with a diesel generator in maintenance, the following basic events associated with an independent failure of the diesel generator in maintenance to run or start or of its corresponding output circuit breaker to close were set to "False" in the Flag file.

  A diesel in maintenance: CC-DGS-2, CC-DGS-9, CC-DGS-15, CC-DGS-22, CC-DGS-36

  B diesel in maintenance: CC-DGS-3, CC-DGS-10, CC-DGS-16, CC-DGS-23, CC-DGS-37

  C diesel in maintenance: CC-DGS-1, CC-DGS-8, CC-DGS-17, CC-DGS-24, CC-DGS-38

  The above changes were only related to the calculations associated with Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP).

Model Changes due to The Application

The preceding changes allow a more reasonable calculation of the risk associated with the extended DG AOT. In addition to these, changes made to the model to reflect the extended AOT for the emergency DGs were to increase the maintenance unavailabilities associated with DGs A, B, and C (Basic Events MNUN1R43S001A, MNUN1R43S001B, and MNUN1R43S001C) from 5.51E-3, 7.205E-3, 5.51E-3 to 2.0E-2, 1.545E-2, and 2.0E-2, respectively. This was based on a conservative assumption that the existing maintenance unavailabilities were all due to corrective maintenance.

Compensatory Measures

To compensate for the risk increase resulting from the extended DG AOT when a DG is removed from service for planned maintenance, the following compensatory measures were considered in the evaluation of the risk impact (applies only when a DG is removed from service for maintenance):

- Dedication of the B DG to the unit with a DG in planned maintenance that exceeds 72 hours. This ensures the presence of two DGs per unit in the event of an accident situation. On undervoltage, the ability for automatic alignment of the swing diesel to the unit with a LOCA signal does not depend on the position of the diesel select mode switch. The B diesel generator is affixed to a single unit with A or C diesel generator removed for planned maintenance. The purpose of this preferential alignment is to avoid the situation in which the unit with a DG removed from service is left with only one diesel generator in response to selected initiating events.

Basic events UOL1, UOL3, UOL24, and MIUNDGS_DGB were set to "False" in the Flag file to model this selection for configurations involving removal of A or C diesel generator from service for planned maintenance.

- The Reactor Protection System (RPS) "Throwover" switch, which controls the location of the alternate source of AC power for the RPS system, is selected to 1R25S037 when diesel generator A is in maintenance and selected to 1R25S036 when diesel generator C is in maintenance. The purpose of this electrical alignment is to ensure that the alternate source of AC power for the RPS system is supplied from a bus that is not affected by the removal of a diesel generator for maintenance.

  This alignment is modeled by the logic state set for the relevant flag events (FL-RPSBUS-S036 and FL-RPSBUS-S037) in the Flag file. Note that for average risk calculations, these flags were set to 0.5.

- Only one DG of the five DGs for both units will be removed for planned maintenance at a time.

- Planned DG maintenance will not coincide with planned work in the High Voltage Switchyard.

- When a diesel generator is removed from service for maintenance with the unit online, no additional risk-significant maintenance or other activities will be performed, except for battery charger swapping and ATTS surveillance including functional tests and calibrations (i.e., Basic Events with their first 4 characters designated as TTUN). A list is established to include components which will not be removed for planned maintenance during diesel maintenance which exceeds 3 days for A and C diesels or 7 days for B diesel.

  This is modeled by setting all maintenance terms (except battery charger swapping and ATTS surveillance) with a value greater than 5E-10 to 0.0, or setting them to 0.0 for all maintenance terms down to a value until the average CDF no longer changes.

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

Calculated Change in Risk

The results of the Tier 1 risk impact analysis for DG AOT extension are as follows:

Increase in average risk from internal events:

ΔCDF = 3.0E-07 event/year
ΔLERF = 1.79E-07 event/year

Although the increase in LERF exceeds the regulatory guidance of 1.0E-7 event/year, the new baseline LERF value with the extended DG Completion Times is still very low; i.e., 1.602E-6 event/year. The significant increase in the LERF average risk is primarily due to the fact that the loss of offsite power (LOSP) initiating event is a dominant contributor to the LERF value [which is not uncommon among the Boiling Water Reactor (BWR) 4 Mark I units] and it places a significant worth on the diesel generator availability. As a result, the LERF values for certain Hatch diesels exceed the regulatory guidance.

Conservative increase in average risk from fire events:

ΔCDF = 6.0E-07 event/year
ΔLERF = 3.9E-07 event/year

Extension of Instrumentation Surveillance Intervals

Application and Plant Elements Affected

This Technical Specification amendment changed the surveillance intervals for instrumentation channel function tests and channel calibrations from 92 days to 92 days on a staggered test basis.

The instruments affected include:

1/2B21N015A,B,C,D
1/2B21N056A,B,C,D
1/2B21N620A,B,C,D 1/2B21N621A,B,C,D 1/2B21N622A,B,C,D
1/2B21N623A,B,C,D 1/2B21N624A,B,C,D 1/2B21N625A,B,C,D 1/2B21N626A,B,C,D
1/2B21N641B,C 1/2B21N642A,B 1/2B21N643A,B
1/2B21N678A,B,C,D
1/2B21N680A,B,C,D 1/2B21N681A,B,C,D 1/2B21N682A,B,C,D 1/2B21N685A,B
1/2B21N686A,B,C,D 1/2B21N687A,B,C,D 1/2B21N688A,B,C,D 1/2B21N689A,B,C,D
1/2B21N690A,B,C,D,E,F 1/2B21N691A,B,C,D 1/2B21N692A,B,C,D

1/2B21N693A,B,C,D 1/2B21N694A,B,C,D 1/2B21N695A,B
1/2B31N679A,D
1/2C71N650A,B,C,D
1/2E11N655A,B,C,D 1/2E11N656A,B,C,D
1/2E11N682A,B
1/2E11N694A,B,C,D
1/2E21N651A,B 1/2E21N652A,B 1/2E21N655A,B
1/2E41N651 1/2E41N655A,B,C,D 1/2E41N657A,B 1/2E41N658A,B,C,D
1/2E41N662B,D
1/2E41N670A,B 1/2E41N671A,B
1/2E51M602A,B 1/2E51M603A,B
1/2E51N657A,B 1/2E51N658A,B,C,D
1/2E51N661A,B 1/2E51N665A,B,C,D 1/2E51N666A,B,C,D
1/2E51N685A,B,C,D
1/2G31N662A,D,E,H,J,M 1/2G31N663A,D,E,H,J,M

Conservatisms in PRA Sensitivity Analysis

A PRA sensitivity analysis was performed to evaluate the maximum risk impact of the extension of the surveillance intervals for the Reactor Protection/Emergency Core Cooling System (RPS/ECCS) instrumentation. All of the instrumentation failures modeled in the Plant Hatch PRA were conservatively assumed to be latent. As such, the instrumentation failure probabilities were conservatively multiplied by a factor of 2 to model the extension of the surveillance intervals by the same factor.

The specific changes in the basic event values for the sensitivity analysis are listed in the following:

RPS Instrumentation

Basic Event	Description	Revision 1 Failure Probability	Modification	Sensitivity Failure Probability
Reactor Water Level				
RZFD1C71K14_ CCF	Common cause failure of RPS actuation relays.	2.28E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-06

CCFR2	Common cause failure for all relays in RPS individual logic channels for each failure input.	2.28E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-06
CCSH1C71CHANA1	Common cause failure of RPS logic channel A1	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANA2	Common cause failure of RPS logic channel A2	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANB1	Common cause failure of RPS logic channel B1	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANB2	Common cause failure of RPS logic channel B2	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
RLFD1C71K14A	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14B	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C71K14C	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14D	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14E	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14F	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14G	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14H	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
LZFD1B21N080_CC	Common cause failure of water level transmitters for all N080 channels	1.35E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.7E-05
RLFD1C71K6A	RPS relay for N680A channel	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K6B	RPS relay for N680B channel	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C71K6C	RPS relay for N680C channel	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K6D	RPS relay for N680D channel	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
LXOR1B21N080 A	Water level transmitter	2.74E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.48E-03
LXOR1B21N080 B	Water level transmitter	2.74E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.48E-03
LXOR1B21N080 C	Water level transmitter	2.74E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.48E-03
LXOR1B21N080 D	Water level transmitter	2.74E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.48E-03
BIFD1B21N680A	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
BIFD1B21N680B	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
BIFD1B21N680C	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04

BIFD1B21N680D	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
RLFD1C71K15A	RPS manual scram relay (4 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K15B	RPS manual scram relay (4 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K15C	RPS manual scram relay (4 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K15D	RPS manual scram relay (4 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
SWFD1C71S3A	Manual scram pushbutton switch A.	1.69E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.38E-03
SWFD1C71S3B	Manual scram pushbutton switch A.	1.69E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.38E-03
SWFD1C71S3_CC	Common cause failure of pushbutton switches A and B	8.45E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.69E-04

Questions and Responses

OPHEMANSCRAM	Operator action to manually scram	1.06E-03	This value stays the same but the mechanical success of the activity is actually failed in another part of the model. As modeled here it follows procedure and is a low probability of failure to do so because of training and simplicity of performance. The physical result will be a failure of the RPS system to respond to manual input if auto input fails.	1.06E-03
MSIV Closure Scram (APRM Input)				
RZFD1C71K14_CCF	Common cause failure of RPS actuation relays.	2.28E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-06
CCFR2	Common cause failure for all relays in RPS individual logic channels for each failure input.	2.28E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-06
CCSH1C71CHANA1	Common cause failure of RPS logic channel A1	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANA2	Common cause failure of RPS logic channel A2	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANB1	Common cause failure of RPS logic channel B1	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04

Questions and Responses

CCSH1C71CHA NB2	Common cause failure of RPS logic channel B2	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
RLFD1C71K14A	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14B	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14C	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14D	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14E	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14F	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14G	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14H	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

CCFS2	Common cause failure of APRMs	3.08E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.16E-04
RLFD1C71K12A	RPS relay for APRMA High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12B	RPS relay for APRMB High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12C	RPS relay for APRMC High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12D	RPS relay for APRMD High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12E	RPS relay for APRMA High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12F	RPS relay for APRMB High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12G	RPS relay for APRMC High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12H	RPS relay for APRMD High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YA	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C51K1YB	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YC	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YD	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XE	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XF	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XG	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XH	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
LMFD1C51K617 A	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.7E-04
LMFD1C51K617 B	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.7E-04

LMFD1C51K617 C	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.7E-04
LMFD1C51K617 D	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.7E-04
APRMA	APRMA sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
APRMB	APRMB sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
APRMC	APRMC sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
APRMD	APRMD sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
TUFD1C51K615 A	APRMA electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
TUFD1C51K615 B	APRMB electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
TUFD1C51K615 C	APRMC electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
TUFD1C51K615 D	APRMD electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04

Enclosure 1

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

MSIV Closure Scram (MSIV Closure Input)				
RLFD1C71K3A	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03
RLFD1C71K3B			This relay is mistakenly left out of model, RLFD1C71K3C is used twice.	
RLFD1C71K3C	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03
RLFD1C71K3D	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03
RLFD1C71K3E	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03
RLFD1C71K3F	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03
RLFD1C71K3G	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03
RLFD1C71K3H	RPS relay for MSIV closure (8 total)	2.87E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.747E-03

Questions and Responses

CCFS1A	Common cause failure of MSIV position limit switches	1.35E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.7E-05
SWFO1C71S1	Reactor Mode Switch contacts fail in such a manner as to maintain MSIV closure in a Bypassed State	1.69E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.38E-03
RLFD1C71K11A	RPS MSIV Closure Bypass relay	1.34E-04	Not modified for sensitivity study because the closure bypass function is not a significant contributor to ATWS	2.68E-04
RLFD1C71K11B			This relay is mistakenly left out of the model. RLFD1C71K11C is used twice	
RLFD1C71K11C	RPS MSIV Closure Bypass relay	1.34E-04	Not modified for sensitivity study because the closure bypass function is not a significant contributor to ATWS	2.68E-04
RLFD1C71K11D	RPS MSIV Closure Bypass relay	1.34E-04	Not modified for sensitivity study because the closure bypass function is not a significant contributor to ATWS	2.68E-04
LSFD1B21F028A	MSIV F028A limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04

LSFD1B21F028B	MSIV F028B limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
LSFD1B21F028C	MSIV F028C limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
LSFD1B21F028D	MSIV F028D limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
LSFD1B21F022A	MSIV F022A limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
LSFD1B21F022B	MSIV F022B limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
LSFD1B21F022C	MSIV F022C limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
LSFD1B21F022D	MSIV F022D limit switch	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
MSIV Closure Scram (High Pressure Scram Signal)				
CCFS3A	Common cause failure of the transmitter and ATTS trip units	2.28E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-05
RLFD1C71K5A	RPS High Pressure Scram relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K5B	RPS High Pressure Scram relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C71K5C	RPS High Pressure Scram relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K5D	RPS High Pressure Scram relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
PXOR1B21N078 A	Reactor Pressure transmitter	2.77E-03	All failure assumed to be due to latent conditions. Original value multiplied times 2	5.534E-03
PXOR1B21N078 B	Reactor Pressure transmitter	2.77E-03	All failure assumed to be due to latent conditions. Original value multiplied times 2	5.534E-03
PXOR1B21N078 C	Reactor Pressure transmitter	2.77E-03	All failure assumed to be due to latent conditions. Original value multiplied times 2	5.534E-03
PXOR1B21N078 D	Reactor Pressure transmitter	2.77E-03	All failure assumed to be due to latent conditions. Original value multiplied times 2	5.534E-03
BIFD1B21N678A	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
BIFD1B21N678B	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
BIFD1B21N678C	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
BIFD1B21N678D	ATTS trip card and ATTS relay	1.85E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	3.7E-04
Turbine Trip Scram (APRM Input)				
RZFD1C71K14_ CCF	Common cause failure of RPS actuation relays.	2.28E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-06

CCFR2	Common cause failure for all relays in RPS individual logic channels for each failure input.	2.28E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.56E-06
CCSH1C71CHANA1	Common cause failure of RPS logic channel A1	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANA2	Common cause failure of RPS logic channel A2	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANB1	Common cause failure of RPS logic channel B1	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
CCSH1C71CHANB2	Common cause failure of RPS logic channel B2	1.21E-04	Original value was a failure rate (Y1BMR) multiplied by 730hours÷2. This value was multiplied by 6 (i.e. [6 × 730]÷ 2	7.25E-04
RLFD1C71K14A	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14B	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C71K14C	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14D	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14E	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14F	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K14G	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
Turbine Trip Scram (APRM Input)				
RLFD1C71K14H	RPS actuation relay (8 in total)	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
CCFS2	Common cause failure of APRMs	3.08E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.16E-04
RLFD1C71K12A	RPS relay for APRMA High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12B	RPS relay for APRMB High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12C	RPS relay for APRMC High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C71K12D	RPS relay for APRMD High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12E	RPS relay for APRMA High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12F	RPS relay for APRMB High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12G	RPS relay for APRMC High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K12H	RPS relay for APRMD High Flux trip	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YA	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YB	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YC	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1YD	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C51K1XE	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XF	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XG	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C51K1XH	APRM High Flux relay	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
LMFD1C51K617 A	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.704E-04
LMFD1C51K617 B	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.704E-04
LMFD1C51K617 C	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.704E-04
LMFD1C51K617 D	APRM Voting Module	8.52E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.704E-04
APRMA	APRMA sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02

APRMB	APRMB sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
APRMC	APRMC sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
APRMD	APRMD sensor and cabling	7.77E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.55E-02
TUFD1C51K615 A	APRMA electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
TUFD1C51K615 B	APRMB electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
TUFD1C51K615 C	APRMC electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
TUFD1C51K615 D	APRMD electronics module	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.018E-04
Turbine Trip Scram (Turbine Stop Valve input)				
LSFDSTOPVALVE_1	Turbine Stop Valve Closure limit switch input	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.02E-03
LSFDSTOPVALVE_2	Turbine Stop Valve Closure limit switch input	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.02E-03
LSFDSTOPVALVE_3	Turbine Stop Valve Closure limit switch input	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.02E-03

LSFDSTOPVALVE_4	Turbine Stop Valve Closure limit switch input	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.02E-03
LSFDSTOPVALVE_CC	Common cause failure of stop valve limit switches	1.34E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-05
PZFD1C71N003_CC	Common cause failure of turbine first stage pressure switches.	2.65E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.3E-05
RLFD1C71K10A	RPS relay for Turbine Stop Valve closure-TSV1	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K10B	RPS relay for Turbine Stop Valve closure-TSV1	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K10C	RPS relay for Turbine Stop Valve closure-TSV3	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K10D	RPS relay for Turbine Stop Valve closure-TSV2	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K10E	RPS relay for Turbine Stop Valve closure-TSV2	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K10F	RPS relay for Turbine Stop Valve closure-TSV3	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

RLFD1C71K10G	RPS relay for Turbine Stop Valve closure-TSV4	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K10H	RPS relay for Turbine Stop Valve closure-TSV4	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
PSFD1C71N003A	Turbine 30% power bypass switch. Failure of this mechanism to un-bypass could cause an ATWS	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
PSFD1C71N003B	Turbine 30% power bypass switch. Failure of this mechanism to un-bypass could cause an ATWS	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
PSFD1C71N003C	Turbine 30% power bypass switch. Failure of this mechanism to un-bypass could cause an ATWS	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04
PSFD1C71N003D	Turbine 30% power bypass switch. Failure of this mechanism to un-bypass could cause an ATWS	2.69E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.38E-04

RLFD1C71K9A	RPS relay for Turbine Stop Valve/Turbine Control Valve bypass	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K9B	RPS relay for Turbine Stop Valve/Turbine Control Valve bypass	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K9C	RPS relay for Turbine Stop Valve/Turbine Control Valve bypass	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K9D	RPS relay for Turbine Stop Valve/Turbine Control Valve bypass	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
Turbine Trip Scram (Turbine Control Valve input)				
PZFD1C71N005_CC	Common cause failure of Turbine Control Valve fast close pressure switches	2.65E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.3E-05
PSFD1C71N005A	Turbine Control Valve fast closure pressure switch (1 for each TCV)	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.0E-03
PSFD1C71N005B	Turbine Control Valve fast closure pressure switch (1 for each TCV)	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.0E-03

PSFD1C71N005C	Turbine Control Valve fast closure pressure switch (1 for each TCV)	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.0E-03
PSFD1C71N005D	Turbine Control Valve fast closure pressure switch (1 for each TCV)	3.01E-03	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.0E-03
RLFD1C71K8A	RPS relay for Turbine Control Valve fast closure	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K8B	RPS relay for Turbine Control Valve fast closure	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K8C	RPS relay for Turbine Control Valve fast closure	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1C71K8D	RPS relay for Turbine Control Valve fast closure	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04

ECCS Instrumentation

Basic Event	**Description**	**Present Failure Probability**	**Modification**	**Final Failure Probability**
HPCI (High Reactor Water Level Trip)				
LXOR1B21N093 B	Reactor Water Level transmitter	3.31E-05	The same value as used for LXOR1B21N080A-D was used for consistency.	5.48E-03
LXOR1B21N095 B	Reactor Water Level transmitter	3.31E-05	The same value as used for LXOR1B21N080A-D was used for consistency.	5.48E-03

TUFD1B21N693 B	ATTS trip card-unit associated with LXOR1B21N093 B	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.02E-04
TUFD1B21N695 B	ATTS trip card-unit (Master) associated with LXOR1B21N095 B	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.02E-04
TUFD1B21N693 D	ATTS trip card-unit (Slave) associated with LXOR1B21N095 B	5.09E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.02E-04
RLFD1A70K363 B	ATTS relay associated with trip card-unit TUFD1B21N693 B	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
RLFD1A70K366 B	ATTS relay associated with trip card-unit TUFD1B21N693 D	1.34E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.68E-04
HPCI/RCIC (Auto Start)				
Common cause failure affiliated with relay 1E41K41-HPCI				
CC-HP-8		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-15		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-21		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

Questions and Responses

CC-HP-26		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-30		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-33		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-35		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-36		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K42-HPCI				
CC-HP-1		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-9		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-10		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-11		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-12		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-13		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-14		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-15		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K52-HPCI				
CC-HP-7		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-14		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-20		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-25		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-29		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-32		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-34		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-36		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K53-HPCI				
CC-HP-2		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-9		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-16		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-17		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-18		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-19		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-20		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-21		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K79A-RCIC				
CC-HP-4		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-11		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-17		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-22		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-27		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-28		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-29		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-30		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K79B-RCIC				
CC-HP-5		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-12		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-18		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-23		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-27		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-31		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-32		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-33		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K80A-RCIC				
CC-HP-3		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-10		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-16		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-22		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-23		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-24		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-25		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-26		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure affiliated with relay 1E41K80B-RCIC				
CC-HP-6		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-HP-13		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-19		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07

CC-HP-24		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-28		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-31		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-34		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-35		4.79E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.58E-07
CC-HP-37		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with level transmitter LXOR1B21N091A				
CC-HP-38		7.85E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.57E-05
CC-HP-42		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-43		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07

CC-HP-44		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-48		2.07E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.137E-07
Common cause failure associated with level transmitter LXOR1B21N091B				
CC-HP-41		7.85E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.57E-05
CC-HP-44		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-46		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-47		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-48		2.07E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.137E-07
Common cause failure associated with level transmitter LXOR1B21N091C				
CC-HP-39		7.85E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.57E-05
CC-HP-42		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07

CC-HP-45		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-46		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-48		2.07E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.137E-07
Common cause failure associated with level transmitter LXOR1B21N091D				
CC-HP-40		7.85E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.57E-05
CC-HP-43		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-45		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-47		6.88E-08	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.37E-07
CC-HP-48		2.07E-07	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.137E-07
Common cause failure associated with ATTS master trip card TUFD1B21N691A				
CC-HP-55		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05

CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS slave trip card TUFD1B21N692A				
CC-HP-52		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS master trip card TUFD1B21N691B				
CC-HP-53		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS slave trip card TUFD1B21N692B				
CC-HP-56		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS master trip card TUFD1B21N691C				
CC-HP-54		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06

Questions and Responses

Common cause failure associated with ATTS slave trip card TUFD1B21N692C				
CC-HP-50		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS master trip card TUFD1B21N691D				
CC-HP-49		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS slave trip card TUFD1B21N692D				
CC-HP-51		4.84E-05	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	9.68E-05
CC-HP-57		2.55E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	5.1E-06
Common cause failure associated with ATTS relay RLFD1A70K362A (associated with N692A)				
CC-HP-58		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-HP-62		6.71E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.34E-05

Enclosure 1

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

Common cause failure associated with ATTS relay RLFD1A70K362B (associated with N692B)				
CC-HP-59		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-HP-62		6.71E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.34E-05
Common cause failure associated with ATTS relay RLFD1A70K365A (associated with N692C)			Relay K365A has been replaced by 1E21K371C. Amodel revision will be necessary to change the description. The quantified results will not change.	
CC-HP-60		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-HP-62		6.71E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.34E-05
Common cause failure associated with ATTS relay RLFD1A70K365B (associated with N692B)			Relay K365B has been replaced by 1B21K311D. Amodel revision will be necessary to change the description. The quantified results will not change.	
CC-HP-61		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-HP-62		6.71E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.34E-05

Core Spray/RHR Low Reactor Water level start instruments				
Common cause failure associated with ATTS Master trip card (TUFD1B21N691A)			Note: This is modeled twice, once here and once under HPCI/RCIC start logic. The common cause treatment is different because this particular logic only deals with a Master trip card, as opposed to a Master/Slave arrangement.	
CC-LC-25		1.17E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.34E-04
CC-LC-28		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-30		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-32		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-33		3.09E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.18E-06
Common cause failure associated with ATTS Master trip card (TUFD1B21N691B)			Note: This is modeled twice, once here and once under HPCI/RCIC start logic. The common cause treatment is different because this particular logic only deals with a Master trip card, as opposed to a Master/Slave arrangement.	
CC-LC-24		1.17E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.34E-04

CC-LC-27		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-30		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-31		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-33		3.09E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.18E-06
Common cause failure associated with ATTS Master trip card (TUFD1B21N691C)			Note: This is modeled twice, once here and once under HPCI/RCIC start logic. The common cause treatment is different because this particular logic only deals with a Master trip card, as opposed to a Master/Slave arrangement.	
CC-LC-26		1.17E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.34E-04
CC-LC-29		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-31		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-32		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06

CC-LC-33		3.09E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.18E-06
Common cause failure associated with ATTS Master trip card (TUFD1B21N691D)			Note: This is modeled twice, once here and once under HPCI/RCIC start logic. The common cause treatment is different because this particular logic only deals with a Master trip card, as opposed to a Master/Slave arrangement.	
CC-LC-23		1.17E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.34E-04
CC-LC-27		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-28		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-29		1.03E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.06E-06
CC-LC-33		3.09E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.18E-06
Common cause failure associated with ATTS relay RLFD1B21K361A (associated with N691A)				
CC-LC-14		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04

CC-LC-17		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-19		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-21		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-22		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K7A (associated with N691A)				
CC-LC-3		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-LC-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06

Questions and Responses

Common cause failure associated with ATTS relay RLFD1B21K361B (associated with N691B)				
CC-LC-13		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-LC-16		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-19		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-20		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-22		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K7B (associated with N691B)				
CC-LC-2		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-LC-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

CC-LC-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ATTS relay RLFD1B21K370C(associated with N691C)				
CC-LC-12		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-LC-16		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-17		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-18		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-22		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K8A (associated with N691C)				
CC-LC-1		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-LC-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

Questions and Responses

CC-LC-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ATTS relay RLFD1B21K310D(associated with N691D)				
CC-LC-15		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.55E-04
CC-LC-18		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-20		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-21		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.235E-06
CC-LC-22		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K8B (associated with N691D)				
CC-LC-4		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-LC-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

CC-LC-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-LC-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Core Spray/RHR Low Pressure Permissive Instrumentation				
PXOR1B21N090 A	Reactor Pressure Instrument	1.1E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.2E-04
PXOR1B21N090 B	Reactor Pressure Instrument	1.1E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.2E-04
PXOR1B21N090 C	Reactor Pressure Instrument	1.1E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.2E-04
PXOR1B21N090 D	Reactor Pressure Instrument	1.1E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.2E-04
Common cause failure associated with ATTS Master trip card (TUFD1B21N690A)				
CC-NS-12		2.45E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.9E-04
CC-NS-16		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06

CC-NS-17		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-18		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-22		6.45E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.29E-05
Common cause failure associated with ATTS Master trip card (TUFD1B21N690B)				
CC-NS-13		2.45E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.9E-04
CC-NS-16		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-19		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-20		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-22		6.45E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.29E-05
Common cause failure associated with ATTS Master trip card (TUFD1B21N690C)				
CC-NS-14		2.45E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.9E-04

CC-NS-17		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-19		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-21		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-22		6.45E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.29E-05
Common cause failure associated with ATTS Master trip card (TUFD1B21N690D)				
CC-NS-15		2.45E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.9E-04
CC-NS-18		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-20		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-21		2.15E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	4.29E-06
CC-NS-22		6.45E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.29E-05

Questions and Responses

Common cause failure associated with ATTS relay RLFD1B21K307C(associated with N690A)				
CC-NS-23		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-NS-27		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-28		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-29		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-33		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K9A (associated with N690A & CS)				
CC-NS-3		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NS-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

CC-NS-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ATTS relay RLFD1B21K307D(associated with N690B)				
CC-NS-24		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-NS-27		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-30		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-31		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-33		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K9B (associated with N690B &CS)				
CC-NS-4		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NS-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

CC-NS-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ATTS relay RLFD1B21K309C(associated with N690C)				
CC-NS-25		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-NS-28		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-30		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-32		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-33		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K19A (associated with N690C&CS)				
CC-NS-1		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NS-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

CC-NS-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ATTS relay RLFD1B21K309D(associated with N690D)				
CC-NS-26		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.549E-04
CC-NS-29		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-31		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-32		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-33		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K19B (associated with N690D&CS)				
CC-NS-2		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04

CC-NS-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NS-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K34A (associated with N690A&RHR)				
CC-NSRHRA-3		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRA-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06

Questions and Responses

Common cause failure associated with ESF relay K34B (associated with N690B&RHR)				
CC-NSRHRA-4		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRA-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K35A (associated with N690C&RHR)				
CC-NSRHRA-1		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRA-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

Questions and Responses

CC-NSRHRA-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K35B (associated with N690D&RHR)				
CC-NSRHRA-2		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRA-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRA-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
CC-JS-17		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-JS-18		6.71E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.34E-05
Common cause failure associated with ESF relay K36A (associated with N690A&RHR)				
CC-NSRHRB-3		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04

CC-NSRHRB-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K36B (associated with N690B&RHR)				
CC-NSRHRB-4		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRB-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-10		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06

Questions and Responses

Common cause failure associated with ESF relay K37A (associated with N690C&RHR)				
CC-NSRHRB-1		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRB-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-6		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-7		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
Common cause failure associated with ESF relay K37B (associated with N690D&RHR)				
CC-NSRHRB-2		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-NSRHRB-5		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-8		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06
CC-NSRHRB-9		1.12E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.24E-06

Questions and Responses

CC-NSRHRB-11		3.36E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	6.72E-06
CC-JS-16		1.27E-04	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	2.54E-04
CC-JS-18		6.71E-06	Demand failure assumed to be due all to latent failure. Original value multiplied times 2	1.34E-05

Calculated Change in Risk

Based on the preceding conservative treatment, the calculated change in risk due to internal events shown as follows reflects a very small change in CDF and essentially no change in LERF:

ΔCDF = 5E-7 event/year
ΔLERF $\approx$ 0 event/year

Extension of Station Service Battery AOT

Application and Plant Elements Affected

This proposed Technical Specification amendment is intended to increase the completion time for an inoperable station service battery from 2 hours to 12 hours.

Model Change due to The Application

Due to a proposed increase in the station service battery AOT, the maintenance unavailabilities (Basic Events MNUNSA_BATT and MNUNSB_BATT) would increase from 2.0E-4 to 1.2E-3. This is based on a very conservative treatment of both the frequency of maintenance (6 events in 5 years) and maintenance duration (8 hours per event).

Calculated Change in Risk

For internal events:

ΔCDF = 3.47E-7 event/year
ΔLERF = 3.72E-8 event/year

For internal fire events (based on conservative treatment):

ΔCDF = 1.8E-8 event/year
ΔLERF = 3.5E-9 event/year

In addition to the low risk significance demonstrated by ΔCDF and ΔLERF, the guidance for small quantitative impact on plant risk is met for ICCDP and ICLERP.

Compensatory Measure

To avoid risk-significant plant configurations, it is conservatively required that no planned maintenance will take place on the Maintenance Rule systems during the time when the proposed, extended AOT for the station service battery is invoked. This is based on the consideration that the station service batteries provide control power to a relatively diverse set of equipment. In addition, the proposed, extended station service battery AOT will only be used for emergent work.

Extension of Station Service Battery Charger AOT

Application and Plant Elements Affected

This proposed Technical Specification amendment is intended to increase the completion time for an inoperable station service battery charger from 2 hours to 7 days.

Model Change due to The Application

Due to a proposed increase in the station service battery charger AOT, the maintenance unavailabilities of the standby battery chargers (Basic Events MNUN1R42S028 and MNUN1R42S031) were conservatively increased to 8.35E-3 (it is assumed that the battery charger in maintenance will be switched to be the standby battery charger and an operable battery charger will be placed in service). This is based on a very conservative treatment of the data collected from 1995 through 2001 for both Unit 1 and Unit 2. Most of these maintenance events (including both preventive and corrective maintenance) did not actually involve entry into the station service battery charger AOT. It is, however, conservatively assumed that all of these events would render the associated battery chargers unavailable and entry into the corresponding AOT.

Calculated Change in Risk

For internal events:

ΔCDF = 2.0E-9 event/year
ΔLERF = 1.0E-10 event/year

Compensatory Measure

Only one battery charger in each division will be placed in planned maintenance at a time. Please see response to RAI Question 7 for the calculated ICCDP and ICLERP associated with the proposed, extended station service battery charger AOT.

Cumulative Risk from Combined Change Request

The preceding description provides a summary of the changes in risk resulting from the past and currently proposed applications. Although all of these applications (except the power uprate from 2558 CMWt to 2763 CMWt) were evaluated using the same base model (i.e., Revision 1a of the Plant Hatch CAFTA PRA model), various model enhancements and conservative assumptions were used for each of these applications. The 2558 CMWt to 2763 CMWt upgrade was evaluated with the original RISKMAN IPE model. The Rev 1 model which was the conversion to CAFTA was in progress during the time of the evaluation. As a result, the ΔCDF and ΔLERF calculated for each application are, in principle, not additive. However, the combined total ΔCDF and ΔLERF from all of these applications can serve as an upper bound reference value.

To provide a more accurate picture of change in risk due to the combined effects of all of these applications, it is appropriate to incorporate all of the model enhancements adopted in these applications (in fact, only the DG AOT extension evaluation involves model enhancements) into the Revision 1a model to use as the baseline model for the evaluation of cumulative risk. Then, one can implement the corresponding model changes due to the applications and evaluate the resulting change in risk. It must be noted that the model changes thus incorporated are still encompassing the conservative assumptions used in each of the applications for the evaluation of change in risk. The following provides the change in risk from internal events due to the combined effects of all of these changes including past and current applications:

ΔCDF = 1.06E-6 event/year
ΔLERF = 1.85E-7 event/year

As can be seen from above, the cumulative risk due to internal events from the combined effects of all of the past and current applications are still very small in terms of ΔCDF and ΔLERF. Although these values are slightly higher than the risk acceptance limit specified by Regulatory Guide (RG) 1.174 for ΔCDF and ΔLERF (1.0E-6 and 1.0E-7, respectively), it must be recognized that this is due primarily to the many conservative assumptions used in the calculations of ΔCDF and ΔLERF.

NRC Question:

The proposed extensions to station service batteries is stated to be used only for emergent conditions (e.g., corrective maintenance – component failed). Based on this, provide a discussion on how common cause was considered per the guidance of RG 1.177 (Appendix A, A.1.3.2.1), "An Approach for plant specific, Risk Informed Decisionmaking: Technical Specifications," for the proposed station service battery CT extension.

SNC Response:

Revision 1a of the Plant Hatch PRA model included only those common cause failure (CCF) events that occur with significant rate. These typically involve active failure modes of electro-mechanical, electrical, and mechanical equipment; such as pump failures to start/run, compressor failures to start/run, fan failures to start/run, chiller failures to start/run, diesel generator failures to start/run, breaker failures to open/close, relay failures to operate, switch/trip unit failures to operate on demand, motor- or air-operated valve failures to open/close, check valve failures to open/reseat, etc.

The major equipment in the DC power supply system includes batteries, battery chargers, DC buses, and DC breakers. In considering the CCF modeling in the Station Service DC power system, the industry CCF database (at the time this submittal for the proposed amendment was being prepared) was reviewed. For battery banks, there were 12 events in the Idaho National Engineering and Environmental Laboratory (INEEL) CCF database (corresponding to approximately 15 years of data over 100 nuclear plants). However, all of these failure events were excluded from consideration because they were generally not within the PRA definition of failure. Most of the events included in the database involve improper electrolyte level (i.e., high or low), inadequate specific gravity for full electrolyte level, and failures due to aging or electrical cycling in a few cells. These out-of-specification batteries typically are still above the design minimum capacity and are capable of performing their intended function. It was, therefore, the view of the reviewing analyst that the common cause failures contribution for batteries was very small and did not warrant complicating the model with CCF terms.

The events in the INEEL CCF Database for battery chargers (at the time the evaluation was performed) were also reviewed in the same manner as the battery banks. Most of the failures were related to returning battery chargers to service following maintenance, placing standby battery chargers into service, repowering the battery chargers following a loss of AC supply, failures of different sizes of battery chargers, failures involving different parts, failures that occurred with sufficient time apart, failures that would be detected and corrected prior to unit startup, or failures to operate in a mode that is not consistent with response to initiating events. As such, with the exception of a couple of events such as one in which the battery charger failures occurred during the repowering phase of a power transient, most other events occurred in a condition that is not consistent with the response of a normally operating battery charger to initiating events. Therefore, there was very little evidence that common cause failures, as defined in PRA for normally operating battery chargers, occurred at a significant rate. As such, it was not considered necessary to model the common cause failures for the battery chargers either.

As to the DC buses, there were no common cause failure events in the INEEL database (at the time of the evaluation) to suggest that common cause failures between DC buses occur at a significant rate. It was therefore not considered necessary to model common cause failures between and among the DC buses.

Finally, the DC power breakers included in the Station Service DC Power System model for Hatch Revision 1a PRA do not involve any active failure modes such as failures to open and failures to close. As such, no CCFs were modeled for these breakers in the Station Service DC Power System.

Based on the above considerations, Revision 1a of the Plant Hatch PRA did not include CCF basic events in the Station Service DC Power System model. Therefore, the treatment of conditional probabilities for the common cause failure events (as per the guidance of RG 1.177) was not needed in the evaluation of the proposed extension to station service batteries for configurations involving removal of one station service battery due to emergent conditions (e.g., corrective maintenance).

NRC Question:

The Tier 2 evaluation states that no planned maintenance will be allowed on maintenance rule systems while the service battery extended AOT is employed. Provide a discussion of the Tier 2, "Avoidance of Risk-Significant Plant Configurations" evaluation performed and the methodology used to identify risk significant equipment outage configuration and any risk outliers. Discuss the adequacy/completeness of using maintenance rule systems as an all inclusive compensatory measure with a station battery out of service and the applicability to the guidance presented in section 2.3.6 of RG 1.177. Areas of discussion should include that the identified compensatory measures were incorporated as part of the analysis, the compensatory measures identified do not compensate for inherent weakness in plant design, the compensatory measures are not already credited, and the identified compensatory measures will become part of the licensing basis. In addition will these compensatory measures be imposed independently of Tier 3 and the Hatch 10 CFR 50.65 based configuration risk management program (CRMP) results? Enclosure E4-15 states that CT will be used for emergent work only.

Since the extended CT is intended for emergent work only, provide a discussion on how the extensive Tier 2 compensatory measures can be implemented without prior planning before entering the limiting condition for operation (LCO) for repair.

SNC Response:

The purpose of the Tier 2 (Avoidance of Risk-Significant Plant Configurations) evaluation is to identify and avoid those high-risk plant configurations that involve a combination of a station service battery and other pieces of equipment being out of service at the same time. The approach used in the analysis in support of the submittal Tier 2 evaluation was to first identify those components that have the highest values in the Risk Achievement Worth (RAW) ranking for core damage frequency given a station service battery is out of service. The lists of minimum cutsets generated from the CDF quantification for configurations with a station service battery removed from service were used to develop the RAW ranking for basic events contained in the above cutsets. The highest RAW value on these lists implies the greatest increase in core damage likelihood if the associated equipment is also made unavailable during the time period when a station service battery is out of service.

Using this approach, a number of the most risk-significant equipment outage configurations were identified. ICCDP were then calculated for these configurations, each involving one of these components/events and a station service battery removed from service simultaneously. The results of the ICCDP values calculated for the configurations associated with this set of components/events were presented in the submittal under the section for "Tier 2: Avoidance of Risk-Significant Plant Configurations." A higher ICCDP value implies a greater increase in CDF when the corresponding configuration is entered. The results of the evaluation show that risks associated with these configurations are not minimal. When station service battery A is inoperative, the highest contributors are ECCS instrument channel 1B21N690B, the common breaker for the Division A battery chargers, one of three fuses for Division B station service battery, Division B station service battery charger swapping in progress, and HPCI injection valve. With station service battery B inoperative, the greatest contributions come from the common breaker for the Division B battery chargers, one of three fuses for Division A station service battery, Division A station service battery charger swapping in progress, Residual Heat Removal (RHR) pump A, and RCIC pump. In light of this, the compensatory measure selected is to disallow any planned maintenance on Maintenance Rule systems while one station service battery is inoperative.

The intent of the compensatory measure to disallow all planned maintenance on any other Maintenance Rule systems is to ensure that the increased core damage likelihood will not be voluntarily introduced to exceed the level induced by the removal of a station service battery alone. This actually is a very conservative approach. It covers not only those components modeled in the PRA (which implies that they may affect the quantitative risk contribution), but also the remaining systems included in the Maintenance Rule. By encompassing all of the Maintenance Rule systems, this compensatory measure is certainly complete in terms of avoiding voluntary entry into risk-significant configurations. Of course, emergent conditions may still occur during the time while one station service battery is inoperative. However, since planned maintenance including surveillance testing on the PRA and Maintenance Rule systems is disallowed, it is highly unlikely that an emergent failure involving those components identified with the risk-significant configurations would occur during this very short period of time (because many of the component failures were uncovered during surveillance testing); i.e., 12 hours. Therefore, it is considered adequate to have the compensatory action to disallow all planned maintenance on the Maintenance Rule systems during the time when a station service battery is inoperative.

As shown in the Tier 1 (PSA Capability and Insights) section of the submittal, the increase in risk is small even using very conservative assumptions in the calculations. Due to the capability of the battery chargers available at Plant Hatch and the conservative assumptions used, it is judged that no additional compensatory measure is needed to balance the calculated risk increase caused by the proposed, extended station service battery AOT. Since no compensatory measures are considered in the Tier 1 portion of the evaluation, a risk impact analysis of the compensatory measures is not necessary. Regarding the compensatory measure for the Tier 2 evaluation (i.e., disallowing planned maintenance on Maintenance Rule systems while a station service battery is inoperative), the risk impact analysis for ΔCDF/ΔLERF and ICCDP/ILERP is consistent with the measure of interest because only the removal of a station service battery is considered in the calculation of ICCDP and ICLERP. However, this compensatory measure should not be considered as part of the analysis of the change since RG 1.177 dictates that the evaluation of ΔCDF/ΔLERF and ICCDP/ILERP only consider the removal from service of the subject equipment involved in the Technical Specification change. Based on the results and the insights from the analysis of the change, there is no weakness in the plant design with respect to the Station Service Battery System. As such, no compensatory measure was identified for the Tier 1 evaluation. In addition, the compensatory measure identified for the Tier 2 evaluation is not a result of any weakness in plant design. It is simply a conservative approach to minimize the risk that the plant may be exposed to if it enters voluntarily into a configuration with simultaneous outages of more than one piece of safety system equipment and a station service battery. Therefore, the plant does not have to rely on this compensatory measure since it does not have any inherent weakness in its design. As implied previously, this compensatory measure is really not credited in the Tier 1 evaluation of risk impact, and as such, it should not be considered as part of the licensing basis. Nevertheless, this compensatory measure will be incorporated into the Plant Hatch administrative procedure for work control and scheduling. It will be part of the Tier 3 and the Hatch 10 CFR 50.65 based configuration risk management program.

Since the proposed, extended station service battery AOT will be used for emergent work only, it will be specified in the Hatch procedure for work control and scheduling that all planned maintenance previously scheduled will be delayed before entering the LCO for repair until the failed station service battery is returned to service. The maximum period of delay is the extended AOT; i.e., 12 hours.

NRC Question:

Page E1-1 of Enclosure 1, item 1 notes that in addition to the proposed extended CT for the station service batteries, a CT for an inoperable battery charger is also proposed. Risk insights are not provided in the submittal for the extended 7 day CT. Provide the results for ΔCDF, ΔLERF, ICCDP and ICLERP. Include an evaluation of the combined change request as outlined in RG 1.174.

Response:

Risk Insights were not provided because this Tech Spec change was performed under TSTF-360, as noted in our original submittal. The TSTF states that any licensees wishing to request a longer Completion time for the batteries should perform risk evaluations of that CT increase per the guidance of RG 1.177. The TSTF did not require such risk insights for the increase in the battery charger AOT, which was already justified, on a generic basis, in the TSTF itself. Nevertheless, these insights are provided below:

With the extension of AOT associated with the station service battery charger from 2 hours to 7 days, it is expected that the maintenance unavailability of these chargers will increase. To evaluate the risk impact of this proposed extension of the battery charger

AOT, a conservative estimate of the maintenance unavailability was performed. This estimate was based on the relevant corrective and preventive maintenance performed on the station service battery chargers during the period from 1995 through 2001 for both Unit 1 and Unit 2. The duration of these maintenance events ranges from more than an hour to over 74 hours with great majority of them last more than 5 hours. For this evaluation, however, it is conservatively assumed that all of these maintenance events rendered the affected battery charger unavailable and entry into the corresponding AOT is necessary. The average maintenance unavailability calculated for each charger using this data is 2.78E-3.

Since the battery charger in maintenance is assumed to be removed from service, the maintenance unavailability of the standby battery charger is assumed to be the total of all three battery chargers in the same subsystem; i.e., 8.35E-3. Assuming that battery chargers 1R42S028 and 1R42S031 are the standby chargers for subsystem A and subsystem B, respectively, the values of Basic Events MNUN1R42S028 and MNUN1R42S031 are increased to 8.35E-3 for the calculation of the new baseline CDF/LERF. For the calculation of CDF/LERF during the period when a station service battery charger is out of service, the corresponding maintenance unavailability basic event is set to TRUE. The results of the risk impact calculation for the case of extending the station service battery charger from 2 hours to 7 days are listed in the following:

Risk Values for Evaluation of Station Service Battery Charger AOT Extension	
CDF (Base)	1.2403E-05
CDF (New Base)	1.2405E-05
LERF (Base)	2.1859E-06
LERF (New Base)	2.1860E-06
CDF (New AOT-1 Div. A Charger OOS)	1.2679E-05
CDF (New AOT-1 Div. B Charger OOS)	1.2481E-05
LERF (New AOT-1 Div. A Charger OOS)	2.2026E-06
LERF (New AOT-1 Div. B Charger OOS)	2.1965E-06

Using the calculated risk values, a comparison of the results against the criteria set forth in RG 1.174 and RG 1.177 is listed in the table below:

Comparison with Regulatory Guide 1.174 Risk Criteria	
RG 1.174 Risk Criteria	**Plant Hatch AOT Extension for Station Service Battery Chargers**
ΔCDF = 1.0E-06	ΔCDF = 2.0E-09
ΔLERF = 1.0E-07	ΔLERF = 1.0E-10
Comparison with Regulatory Guide 1.177 Risk Criteria	
RG 1.177 Risk Criteria	**Plant Hatch AOT Extension for Station Service Battery Chargers**
ICCDP = 5.0E-07	ICCDP (1 Div. A Charger OOS) = 5.25E-09
	ICCDP (1 Div. B Charger OOS) = 1.46E-09
ICLERP = 5.0E-08	ICLERP (1 Div. A Charger OOS) = 3.18E-10
	ICLERP (1 Div. B Charger OOS) = 2.01E-10

Please see response to RAI Question 4 for an evaluation of the combined change request including all previous risk-informed amendments as outlined in RG 1.174.

NRC Question:

The licensee's average CDF estimate is based on a fuel cycle , not a year as referenced by RG 1.177, therefore, the potential exists to exceed the yearly ΔCDF, ΔLERF without schedule restrictions. Provide Plant Hatch policy with respect to maintenance completion times, scheduling, and TS implementation with respect to the proposed plant service battery and charger AOTs (i.e., planned maintenance is not entered unless the maintenance can be performed within half the proposed TS CT). Page E4-8.

Response:

The risk impact evaluation performed for the proposed, extended battery AOT was performed using an annual CDF and annual LERF. It was not on a per fuel cycle basis. The maintenance unavailability calculation was performed using the total out of service hours in a fuel cycle and the total power operation time in a fuel cycle. However, the maintenance unavailability calculated is unit less. It is, therefore, irrelevant whether the out of service hours and power operation time are on a per-year or per-fuel cycle basis. The unit for the CDF and LERF is determined by the unit for the initiating events, which are in terms of events per year.

The proceduralized guidance for planning work on Technical Specification components at Plant Hatch is to use only one-half of the associated AOT. The entire work out of service time is analyzed for risk by the on-line risk monitoring tool. In the case of station service batteries, the corresponding AOT will only be entered for emergent work and will not be invoked for any planned maintenance. Since the consideration of one-half of the AOT is only used for planned maintenance, it is not applicable to emergent work. For emergent work, the plant has no choice but to proceed with the corrective maintenance. If the affected battery cannot be restored to service within the associated AOT, the plant will be required by Technical Specification Action Statement to start the process for unit shutdown. Battery charger maintenance will be analyzed in a similar manner but can be pre-planned or emergent work.

NRC Question:

Enclosure E4-5. The ICCDP calculation shown uses the CDF (New Base) value. Per RG 1.177, this value should be the baseline CDF with current nominal expected equipment unavailabilities. Provide a discussion as to why the updated CDF (NEW base with revised unavailabilities) of the batteries is used in the ICCDP calculation instead of the original nominal expected unavailabilities. This would appear to minimize the ICCDP results. See also the ICLERP calculation on Enclosure E4-6.

Response:

The Incremental Conditional Core Damage Probability (ICCDP) as defined in Notes 2 and 4 of RG 1.177 is as follows:

ICCDP = [(conditional CDF with the subject equipment out of service) – (baseline CDF with nominal expected equipment unavailabilities)] (duration of single AOT under consideration)

It is the interpretation that the above equation is to estimate the maximum, total increased probability of core damage each time the subject equipment is removed from service for the full duration of the proposed, extended AOT (i.e., invoking the proposed, extended AOT). Since the proposed, extended AOT is being analyzed for the increased core damage probability, the condition and the basis assumed for this ICCDP analysis is that the proposed, extended AOT is already in place. Otherwise, the proposed, extended AOT would not be invoked for the analysis of the increased core damage probability. As such, the baseline CDF becomes the baseline CDF assuming the proposed, extended AOT is already in place. In other words, all three terms in the above equation have to be consistent with the condition and basis assumed, which is the proposed, extended AOT in this case.

Therefore, the first two terms in the equation is to calculate the increase in CDF from the average risk (assuming the proposed, extended AOT is already in place) due to the removal of the subject equipment from service (e.g., caused by equipment failure) and due to the entry into the proposed, extended AOT; i.e., the instantaneous CDF (with respect to the equipment removed from service) minus the average CDF (corresponding to the proposed, extended AOT). Since the unit for the first two terms is events per year, the third term in the above equation is estimated in the unit of "year." Multiplying the remainder from the first two terms with the third term results in the increased core damage probability for invoking the proposed, extended AOT; i.e., ICCDP.

Due to the interpretation as explained in the above, the "CDF (New Base)" value was used in the evaluation of the proposed, extended AOT for the calculation of ICCDP. For the same reason, the ICLERP was calculated in a similar manner.

NRC Question:

Page E4-16 of the submittal states that the Hatch Plant configuration risk management is provided by the maintenance rule 10 CFR 50.65(a)(4). Provide a discussion on the applicability of the Hatch 10 CFR 50.65(a)(4) based CRMP program meeting the additions and clarifications provided in RG 1.177 Section 2.3.7.2, Key Components 1 through 4.

SNC Response:

The risk-informed maintenance management program implemented at Plant Hatch is a procedurally controlled program that supports the implementation of the 10 CFR 50.65(a)(4) and all of the risk-informed AOT extensions requested by Plant Hatch. This program satisfies the additions and clarifications associated with all four key components outlined in Section 2.3.7.2 of RG 1.177 as described in the follows:

Key Component 1: Implementation of Configuration Risk Management Program (CRMP)

1. The maintenance scheduling and planning program implemented at Plant Hatch governs the scheduling of all operational and maintenance activities. In addition, it monitors and evaluates any configuration changes during operation

whether a surveillance/maintenance event is in progress or not. The scope of the equipment included in the quantitative and qualitative risk evaluation encompasses all of the components modeled in PRA and all of the Maintenance Rule functions that are outside the scope of PRA. This essentially includes all Structures, Systems, and Components (SSCs) considered high safety significance per Revision 2 of RG 1.160 that are not modeled in the PRA.

2. The risk management assessment tool used at Plant Hatch is the EPRI Equipment Out of Service (EOOS) risk monitor. At the heart of this on-line, computerized tool is the PRA model which is evaluated directly for quantitative assessment. This risk monitor also includes operator display panel, which performs risk evaluation deterministically based on qualitative fault trees developed for specific functions to be monitored. The results of the evaluations for both quantitative and qualitative assessment are also displayed in color to signify the resulting risk category.

3. Prior to entering the action statement of a system related AOT (including risk-informed AOTs) for any planned maintenance or operational event (e.g., surveillance/testing), a risk assessment will typically be performed using the EOOS risk monitor. This is usually performed by the work control and scheduling staff one or more times before the corresponding work week starts. In the event the equipment involved is not included in the EOOS risk monitor model, a qualitative risk assessment will be performed.

In addition to the risk assessment performed by the work control and scheduling staff for scheduled work items, all emergent conditions are also evaluated as they occur using the EOOS risk monitor if the equipment involved is included in the EOOS model or qualitatively if the equipment involved is not part of the EOOS model (with the assistance of the PRA staff on an as-needed basis). As such, for unplanned entry into plant configurations (e.g., due to emergent failures) described by a Technical Specification action statement associated with a system related AOT (including risk-informed AOTs), a risk assessment is typically performed immediately.

When in the plant configuration described by a Technical Specification statement with a risk-informed AOT, if additional SSCs become inoperable or nonfunctional, a risk assessment will also be performed immediately. If the risk acceptance criteria can no longer be met, removal of equipment from service for scheduled work will be delayed until those failed components are returned to service. In addition, work priorities may also be adjusted according to the risk importance.

4. The risk management program typically performs a risk evaluation for each emergent condition. If the equipment involved is included in the EOOS quantitative PRA model, identification of the risk-significant configurations will also be evaluated. As indicated in the submittal for the station service battery AOT extension, this proposed, extended AOT will only be used for emergent conditions and risk-significant configurations were already identified and will be avoided each time this proposed, extended AOT is entered.

Key Component 2: Control and Use of the CRMP Assessment Tool

1. Plant modifications and procedure changes at Plant Hatch are evaluated periodically. Information related to these changes is provided to the PRA staff prior to the actual implementation of these changes. The qualitative review of these changes by the PRA staff determines if these changes will affect the PRA and EOOS risk monitor models and results. Those changes that do impact the EOOS risk monitor model or results will be incorporated into a new revision of the model periodically. Prior to the actual implementation of the new revision of the revised risk monitor model, the effects of these changes on the assessment of configuration changes will be qualitatively considered.

2. Plant Hatch Administrative Control Procedure 90AC-OAM-002-0, Scheduling Maintenance, provides for the applications of the risk management assessment tools. Instructions are provided as to the qualitative risk assessment when the plant configuration of concern is outside the scope of the EOOS risk monitor model.

Key Component 3: Level 1 Risk Assessment

The EOOS risk monitor used at Plant Hatch includes both quantitative and qualitative evaluation models. The quantitative evaluation model is identical to the PRA model, which includes both the Level 1 and LERF aspects. The qualitative evaluation includes fault trees on major safety functions and selected Maintenance Rule functions including those that are not included in the PRA model. The Administrative Control Procedure used at Plant Hatch for scheduling maintenance also includes a risk matrix based on pre-existing calculations and evaluations.

1. At Plant Hatch, quantitative assessments using the EOOS risk monitor are performed whenever the equipment involved is included in the PRA model.

2. At Plant Hatch, qualitative assessment is used primarily when the equipment involved is not included in the PRA model. However, when qualitative assessments are performed, applicable existing insights from previous quantitative assessments are considered.

Key Component 4: Level 2 Issues and External Events

At Plant Hatch, for Level 2 issues that can be reflected by the assessment for LERF, quantitative evaluations using the EOOS risk monitor are performed. For all other Level 2 issues and external events, qualitative assessments are used.

NRC Question:

Enclosure E4-8 states that the present 2 hour AOT was used on-line for individual cell replacement approximately 3 times in 5 years. How often has the AOT been entered for all station service battery maintenance and surveillance? Does the risk metrics provided in the submittal reflect this frequency and resulting unavailability?

Response:

The 3 times in 5 years were the times that the Technical Specification was invoked. As indicated on Page E4-8 of the submittal, both the frequency and duration of station service battery removal from service were very conservatively represented in the calculation for the risk metrics presented in the submittal. The number of times the AOT was invoked is adequately reflected.

NRC Question:

Confirm that the station service battery reliability and availability will also be monitored and assessed under 10 CFR 50.65 consistent with RG 1.174 Section 2.3, Element 3, such that performance continues to be consistent with the assumptions used in the analysis for extended station service battery AOTs and 10 CFR 50.65 maintenance category.

Response:

The station service batteries are included in the scope of the Plant Hatch Maintenance Rule program. As such, performance criteria have been established for the reliability and availability of the station service batteries. The performance of these batteries will be monitored and assessed against the performance criteria established on a periodic basis to ensure the continued performance of the equipment within an acceptable set of limits. Since a conservative set of maintenance unavailability was used in the risk impact evaluation performed for the proposed, extended station service battery AOT, the conclusion of the evaluation would not change as long as the reliability and availability of the station service batteries do not substantially exceed their Maintenance Rule performance criteria established.

NRC Question:

Provide a discussion on the uncertainty/sensitivity to the proposed extended battery CT per the guidance of outlined in RG 1.177 section 2.3.5.

SNC Response:

The largest uncertainty associated with the proposed, extended battery completion time are the increased maintenance unavailability for the station batteries. However, the submittal was based on an analysis, which used very conservative assumptions regarding both the maintenance frequency and out of service duration for the station batteries given the proposed, extended battery AOT. It serves as, essentially or very close to, an upper bound risk level. As such, in terms of the sensitivity for the impact of variations in the assumed mean downtimes or frequencies, it is much more likely that the true risk increase would be lower than what was calculated and presented in the submittal. This is because both the maintenance frequencies and duration should be significantly less than the values used in the analysis.

Regarding the repair/maintenance policy, it is clearly stated in the submittal that the proposed, extended AOT will only be used for emergent conditions and no planned maintenance will be allowed during the period when this proposed, extended AOT is invoked. As such, the uncertainty in the repair/maintenance policy should be minimal.

NRC Question:

Is there a dedicated DC supply for breaker control?

SNC Response:

Breaker control for equipment supplied by the safety buses (e.g., 4kV) is provided by the emergency diesel generator battery system which includes five subsystems, each with a battery and two battery chargers (one charger is normally in service and one in standby). Control power for station service (non-safety) 4kV buses is provided by the station service batteries affiliated with this AOT.

Enclosure 1

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Questions and Responses

NRC Question:

Confirm that the battery chargers are capable of supplying safety loads independently without the batteries, including transient conditions. Enclosure E1-3.

Response:

The battery chargers are capable of handling the 125 VDC loads. The battery chargers alone, however, cannot handle all the 250 VDC loads. For example, the large Motor Operated Valves on the High Pressure Coolant Injection System, such as the pump discharge valve, cannot be operated alone on the chargers due to the large in-rush current. The batteries would be needed for this operation. This is also true for the Reactor Core Isolation Cooling System large MOVs.

NRC Question:

Internal Fires. The submittal states in Enclosure 4 that the fire area under consideration for the proposed completion time extension are those that may challenge the availability of the station service batteries. The IPEEE for Hatch noted that the risk dominant fire zones included the Division 1 station battery. THE IPEEE fore analysis screened on fire areas qualitatively and fire scenarios quantitatively. Confirm that fire areas and scenarios that require the station service batteries for mitigation were not screened and that these assumptions did not impact the fire analysis results provided in the submittal.

Response:

The fire PRA completed for Plant Hatch in the IPEEE program includes qualitative screening, quantitative screening, and detailed analysis. The qualitative screening was based on the consideration of the potential for initiating events occurrence and mitigation system failure/degradation due to fire damage. A plant location would only be screened if, given a fire event causing the loss of all of the equipment in the location, no initiating event would be induced or no damage to the accident mitigation equipment would occur. Due to the deterministic criteria and the very conservative assumptions used (assuming all equipment in the location is lost), the results of the qualitative screening would not be affected even with the proposed extension of the station service batteries.

For the quantitative screening analysis, a very conservative screening value of 0.1% of the IPE CDF for internal initiating events (2.1E-8 and 2.2-8 event/year for Unit 1 and Unit 2, respectively) was used to ensure that scenarios screened from further analysis were not risk significant. Due to this conservative screening value, only 16 and 25 fire zones were screened from detailed analysis. It must be noted that, although this step of the analysis is called "quantitative screening analysis," the scenarios that were screened were not thrown away in the Plant Hatch IPEEE fire analysis. These screened scenarios were still included in the final representation of the total fire-induced risk. They are only screened for detailed analysis, not screened for being included in the total fire-induced CDF calculation. Besides, due to the very small screening value used, it is not expected that the result of the fire PRA analysis would be significantly affected by the proposed, extended station service battery AOT. Therefore, the fire analysis results provided in the submittal are not expected to be impacted significantly due to the qualitative and quantitative screening performed in the Plant Hatch fire PRA.

Enclosure 2

Edwin I Hatch Nuclear Plant, Units 1 and 2
Response to Request for Additional Information on the DC Sources Technical Specifications Change Request

Peer Review F&O Comments and Observations

1. Accident Sequence (AS) Facts and Observations
2. Data Analysis (DA) Facts and Observations
3. Dependency Analysis (DE) Facts and Observations
4. Human Reliability Analysis (HR) Facts and Observations
5. Initiating Event (IE) Facts and Observations
6. Level 2 Analysis (L2) Facts and Observations
7. PRA Maintenance and Update (MU) Facts and Observations
8. Quantification (QU) Facts and Observations
9. Structural Analysis (ST) Facts and Observations
10. System Analysis (SY) Facts and Observations
11. Thermal Hydraulic (TH) Facts and Observations

ACCIDENT SEQUENCE (AS)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 1**

Process

Event tree description for the CAFTA model and the associated ESDs from the IPE are excellent methods of conveying the knowledge of the accident sequence process. However, the event tree descriptions for the CAFTA model are considered candidates for enhancement in the area of:

- Containment heat removal failure effects (i.e., unique dependency effects of PCS on SORVs, torus cooling requirement for HPCI and RCIC, vent effects on LPCI/CS, high DW pressure effects on SDC).
- The basis for the accident sequence end states and their correlation to NEI's functional accident classes.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Enhancing the documentation would provide a marked improvement in the guidance for future PRA analysts.

PLANT RESPONSE OR RESOLUTION

The revised Hatch model has more complete descriptions for the event tree (Hatch Unit 1 Rev.2 calculation). The referenced information has been included in more complete success criteria for suppression pool cooling for ATWS and Non-ATWS cases. A model to account for emergency venting the containment and its affects on low pressure ECCS has been added as well (AND Gate EMERGENCYVENT). Drywell pressure affects on Shutdown Cooling have been added to the model with OR Gate HDWP (High Drywell Pressure Signal Conditions) being placed under OR Gate, QS-COMMON. High drywell pressure will fail shutdown cooling.

Reflection of the sequences with NEI end state information is not necessary for model accuracy or use. This will be handled when the Level II model is upgraded and has no affect on core damage.

This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 4**
Event Tree Groups The event trees are grouped in a manner consistent with the best BWR PRAs reviewed.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID 1)	**Element AS**	**Subelement 5**

Return to Power

There is a node included in the transient event tree that allows the return to power. The derivation of this node is not discussed in the documentation. Specifically, there are conditional probabilities assigned to the failure to return to power that are not justified:

- MSIV closure ~ .3
- Loss of condenser vacuum ~ .2
- Turbine trip < .1

Historical data was used.

- This "data" is not current.
- The "data" does not show the reactor return to power.
- The "data" is for events that "could be" returned to power within 48 hours.
- The data includes MSIV closures and loss of condenser vacuum.
- The philosophy appears to be contrary to safe operation.
- This "recovery" is not consistent with any other BWR PSA reviewed by the BWROG.

This is atypical in the industry.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Remove this credit or provide detailed justification for the conditional probability assessments.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID 1)	**Element AS**	**Subelement 5**
PLANT RESPONSE OR RESOLUTION		
The RETURN TO POWER TOP EVENT has been removed from the Hatch model.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element AS**	**Subelement 5**
Credit for CST inventory is not well documented. Cannot validate.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Improve documentation to demonstrate CST inventory availability assumptions are consistent with as-built plant and analysis.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element AS**	**Subelement 5**
The LPCI inverters were removed from the plant and the model. This was a major load on the battery. Battery expected life should be significantly higher without this load. This will affect time allowed to recover AC when HPCI available on the battery.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Revise battery expected life based on LPCI inverters no longer a load. Based on revised lifetime of battery revise LOSP recovery factors when HPI is available.		
PLANT RESPONSE OR RESOLUTION		

Battery life without charging units for the station service batteries has been readdressed. In addition the LOSP recovery factors have been recalculated. This is discussed in the Rev.2 calculation for the Hatch Unit 1 model. The battery life increase affects RCIC; the LPCI inverters never were a major load. The indirect references for battery loading are provided in the Rev.2 calculation for the Hatch Unit 1 model. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	**Element AS**	**Subelement 6**

GT-9: CRD & Vent Is a Success

This sequence includes use of HPCI and RCIC and failure of torus cooling thereby leading to loss of HPCI & RCIC short term (~6 hours). The use of CRD at this point would allow the containment vent to be a successful containment heat removal path and would allow an additional success path currently not considered. (See accident sequence GT_ 9.)

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reduce the excess conservatisms in the model and include CRD as a useful injection makeup source.

PLANT RESPONSE OR RESOLUTION

CRD is used in the model as an injection source with regard to operator depressurization actions. If HPCI and RCIC failed at exactly 6 hours, CRD would indeed handle the water level—but containment heat load would still need to be removed. If the containment failed it is possible that CRD would no longer have an injection path. If HPCI and RCIC failed earlier CRD may or may not be enough to hold the level and the containment heat load would still be a consideration. The HATCH CRD system puts out a limited amount of flow and is not considered a viable injection source until the vessel is cooled significantly. Modeling the exact nature of CRD capability for low or high pressure injection is not feasible without defining a single sequence. This is beyond the worth of modeling and is certainly not an excess conservatism. In order to get CRD to put out maximum output (which is still only about 150 GPM) a considerable operator effort is required. The time involved as well as the difficulty of such HRA would provide a probability which would tend to offset any worth of CRD in this situation because it would fail the action.

CRD will be considered for the Level II model as a late injection source. Nevertheless CRD has been added as an injection source for sequences GT_4, GT_9, LOSP_3, and LOSP_6. The low capacity of CRD makes this a very poor injection source until approximately 6 hours after shutdown when the decay heat is around 10 or so megawatts or less. This comment is closed.

FACT/OBSERVATION REGARDING

PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	**Element AS**	**Subelement 6**

Critical Safety Functions

There is no vapor suppression node considered for the SORV/IORV or LOCA cases. Failure of vapor suppression could cause a LERF event; therefore even though it is low probability, it can adversely impact the consequences. This is acceptable if the PRA is to be used only for satisfying the IPE GL 88-20. If the PRA is to be used for applications, it would be necessary to ensure that the vacuum breakers, SRV tail pipe, check valves, and drywell spray are properly represented.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Add vapor suppression into the SORV/IORV and LOCA event tree. Remove this nonconservatism and allow addressing drywell spray in Level 1.

PLANT RESPONSE OR RESOLUTION

A new tree was developed called VAPSUPPRESSION (AND Gate). This accounts for limited suppression pool condensation affects from stuck open drywell to torus vacuum breakers. This model is used in Large and Medium LOCA or LOCA causing events.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element AS**	**Subelement 6**

ATWS

For a PSA to be used effectively for a broad spectrum of applications, the model should be robust in terms of accident sequences included--even if these seem relatively low in frequency in the base model. LOCAs and special initiators combined with a failure to scram do not appear to be evaluated quantitatively in the model (large and medium). In addition, to small LOCA the transient event tree is used despite the fact that the boron may wash out of the RPV for small water LOCAs.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Include the impact of LOCA initiators with a failure to scram in the sequences that are retained in the PSA model. Distinguish between LOCAs below the core and those for which boron would be retained in the RPV.

PLANT RESPONSE OR RESOLUTION

This comment will be addressed in time. Presently the ATWS case is not a consideration for MSPI criteria.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element AS**	**Subelement 6**

ATWS

The ATWS event tree assumes that the availability of the PCS precludes the need to ask questions related to boron injection (i.e., misses a critical safety function). This means that the model assumes successful PCS with a failure to scram reaches a successful end state despite not achieving a shutdown reactor condition. This would appear to be a nonconservative assessment of a failure to scram situation in which core oscillations, potential fuel damage, RPV water level instrument variability, and the ability to control the main condenser, condensate, and SRVs in this dynamic situation, are not treated.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

For turbine trip events with the main condenser initially available, address the safety function of boron injection and the potential for MSIV closure during the lowering of the RPV water level and boron injection.

PLANT RESPONSE OR RESOLUTION

Although this can actually be the case, the Hatch ATWS Event Tree was revised to remove this. Boron injection is now necessary to shutdown the reactor in the Hatch ATWS cases. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element AS**	**Subelement 6**
SORV The SORVs are treated in the General Transient Event Tree. 1, 2, and 3 SORV cases are investigated.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
This is a thorough examination with respect to SORV.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 6)	**Element AS**	**Subelement 6**

Need to consider the entire spectrum of transient scenarios and LOSP in the accident scenario evaluation. The impact of ATWS is probably under estimated by the selection of applicable initiating events

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Provide basis for exclusion of remaining initiators. Perform sensitivity to determine contribution to CDF.

PLANT RESPONSE OR RESOLUTION

Not necessary for MSPI

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 7)	**Element AS**	**Subelement 6**

SORV

The SORV investigation of 1, 2, and 3 SORVs does not have the following:

- Vapor suppression assessment.

  Failure of vapor suppression could cause a LERF event; therefore, even though it is low probability, it can adversely impact the consequences.
- The same high pressure functional fault tree HP-1 is used for all GT nodes regardless of whether there are 1 or 2 SORVs. There does not appear to be a technical basis provided for RCIC with two SORVs to provide adequate injection until low pressure makeup becomes available.
- The requirement to eventually use low pressure injection within the 24 hour mission time does not appear to be addressed.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Justify the nodal treatments for SORVs and HP

PLANT RESPONSE OR RESOLUTION

A vapor suppression fault tree has been added to the Hatch model, Rev.2. The Rev. 2 model has addressed proper failure and/or capability of HPCI and RCIC with failed SRVs. Success criteria for the Rev. 2 model now shows HPCI and RCIC ability. The revised event trees for the Rev. 2 Hatch Unit 1 model address low pressure injection requirements in a more straight-forward manner. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 7**

CRD

As more realism is included in the model, it may be necessary to more accurately reflect the benefit of CRD pumps, particularly as a makeup source to the RPV at extended times. This is important now that the EPG/SAGs are implemented.

The credit for CRD for long term RPV injection has not been included. This is judged to result in an increased level of conservatism in the model. The degree of conservatism is not considered sufficiently large to prevent adequate applications at the Grade 3 level. (See also AS-14, No. 1.)
f

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Add CRD as a means of high pressure injection to allow realistic assessment of end state classes and frequencies.

PLANT RESPONSE OR RESOLUTION
CRD has been added to the Core Damage PSA model for selected sequences as an injection source after HPCI and RCIC have finished their 6 hour mission time. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element AS**	**Subelement 7**
<u>RHRSW</u> Include RPV injection and Containment Flood Capability with RHRSW.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Level 2 may need to reflect the timing and capability of RHRSW for containment flooding.		

PLANT RESPONSE OR RESOLUTION

RHRSW has been added to the model as an alternate injection source. It will therefore be used in the Level II model as one means of covering corium leached from a failed vessel. The model is AND Gate RHRSWINJ. This comment is closed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 3)	Element AS	Subelement 7

LOOP - RPV Depressurization (DE)

There appear to be certain accident sequences under which the ability to depressurize the RPV is not asked. This would appear to prevent representing the importance of the SRVs and the operator action to use the SRVs for RPV depressurization within the Level 1 model. (See LOSP 2, 6, 7.)

This also prevents a representation of Level 1 end state that clearly delineates the RPV pressure status such that the Level 2 analysis can be tailored to address those sequences.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION
Ask the depressurization critical safety function in all Level 1 accident sequences where it influences Level 1 end states.

PLANT RESPONSE OR RESOLUTION
Sequence LOSP_2 allows HPCI and/or RCIC to naturally depressurize the vessel until low pressure injection is reached. Heat removal provides the failure point-if the HPCI/RCIC heat load cannot be removed from containment during this time. This sequence presumes that the Heat Capacity Temperature Limit Curve is "ridden" so to speak to stay within its confines by using HPCI and/or RCIC in pressure control (as required) mode rather that going straight to the required depress point. This is within the confines of the operator action for running HPCI and/or RCIC. Sequence LOSP_6 does consider depressurization with the top event #ADED. In fact this is the opposite case from LOSP_2 where the Heat Capacity Temperature Limit has been reached. Sequence LOSP_7 shows a successful manual depress (i.e. #DE as success). LOSP_10 shows it as failure. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element AS**	**Subelement 7**

Model does not include injection from external sources such as Fire System or RHRSW (in the EOPs).

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Include these low pressure alternate injection sources in the model or justify why not included in the model.

PLANT RESPONSE OR RESOLUTION

Fire Water injection and RHRSW injection trees have been added to the Hatch PSA model. These trees are modeled under AND Gate RHRSWINJ and AND Gate FIREWATERINJ. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	Element AS	Subelement 9

GT 39

Medium LOCA with PCS success assumed. The following items need to be addressed:

Inventory

- How can condensate maintain inventory (i.e., is hotwell makeup system adequate). The hotwell fill valve may be adequate but does not appear to be modeled.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Address the makeup capability of condensate from the hotwell given the 24 hour mission time and all medium LOCAs subsumed in the category.

PLANT RESPONSE OR RESOLUTION

This has been addressed by making #BVPR a large LOCA which in turn fails condensate capability. Two SRVs failed in the open position are also shown on the General Transient Event Tree. This is recoverable inventory loss and is readily made up by the condensate system which has 400000 gallons of CST available as well as several thousands of gallons in the hotwell. Consideration of the decay heat steam rate, injection requirements are in the 40 to 60GPM range after about 3 hours. Medium LOCAs fail the CST system for long term (24 hour injection) because of the potential for a break location where condensate injection cannot keep the core covered. The #DEHICO1 top event does allow for condensate to be used long enough to get reactor pressure to the point of using ECCS injection which can maintain core coverage as per design. This is referenced in the SUCCESS CRITERIA for the Hatch model. This comment is considered closed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	Element AS	Subelement 9

IORV

The IORV event tree correctly recognizes the need to have low pressure injection during an IORV. One area that could be considered for enhancement is the following:

- A potentially conservative approach in the IORV tree is that the RCIC system is assumed inadequate to provide RPV injection until the low pressure shutoff head of the LPCI/CS is reached.

LEVEL OF SIGNIFICANCE

B
POSSIBLE RESOLUTION
Consider resolving the potential nonconservative and conservative items by: • Confirming the success criteria regarding RCIC is an adequate RPV injection source for IORV until LPCI or CS can inject.
PLANT RESPONSE OR RESOLUTION
This has been addressed. RCIC is no longer failed for a single open SRV except for the station blackout case. The Hatch PSA model SUCCESS CRITERIA addresses this item. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element AS**	**Subelement 9**

<u>BOC</u> The BOC models are reasonable and complete in scope. Their inclusion is a superior technique in the PRA. Minor enhancements could be considered: • Inclusion of the break probability between the containment and the first isolation valve (currently neglected). • Inclusion of outboard valve body ruptures (currently neglected). • Inclusion of common cause failures of isolation signals from break logic, or in some cases, potential for single failures in break logic.
LEVEL OF SIGNIFICANCE
C
POSSIBLE RESOLUTION
Consider minor enhancements.
PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element AS**	**Subelement 9**

GT 42 & 46

These sequences are described as medium LOCA events in which core damage does not result. However, two potential items could be researched to ensure a realistic evaluation:

- The sequence does not ask if vapor suppression is available to ensure the containment operates successfully (nonconservative).
- The ability of RCIC to provide an adequate depressurization method during a medium LOCA is apparently assumed in GT 42 (nonconservative).

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Revise model if necessary.

PLANT RESPONSE OR RESOLUTION

The model has been revised and the #BVPR is now a Large LOCA event. Sequences 42 and 46 no longer exist. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 5)	Element AS	Subelement 9

It appears that the model includes sequences that are always should be deleted by the success logic. For instance GT_3 involves scenarios with the condenser available, but loss of decay heat removal. Since the success paths are not used this may be causing unnecessary conservative increase in CDF. Same comment applies to ATWS_3.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Perform sensitivity by running the specific sequences and performing delterm and evaluate whether these sequence result are being appropriately represented or subsumed in the cutsets for the entire model.

PLANT RESPONSE OR RESOLUTION

The success paths constructed by the Event Tree Editor are now used in the model. Any cutsets regarding this comment do not show up in the quantified range with or without ATWS_3 or GT_3. This comment is closed.

<table>
<tr><th colspan="3">FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS</th></tr>
<tr><td>OBSERVATION (ID: 6)</td><td>Element AS</td><td>Subelement 9</td></tr>
<tr><td colspan="3">Define the functional success criteria explicitly in the accident sequence document and correct the end state definition in the event tree. For instance, Sequence ATWS_38 was defined as no core damage; however, that does not appear to be correct since failure of SLC, Power Level control, and PCS were defined as unavailable.

Plant personnel demonstrated that the logic model would yield core damage for this scenarios since SLC logic is included under #LOWS.</td></tr>
<tr><td colspan="3">LEVEL OF SIGNIFICANCE</td></tr>
<tr><td colspan="3">D</td></tr>
<tr><td colspan="3">POSSIBLE RESOLUTION</td></tr>
<tr><td colspan="3">Verify the core damage end states in the event tree files, and edit accordingly.</td></tr>
<tr><td colspan="3">PLANT RESPONSE OR RESOLUTION</td></tr>
<tr><td colspan="3"></td></tr>
</table>

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 7)	**Element AS**	**Subelement 9**

Sequence ATWS_102, and 103 may need to consider that ATWS scenarios with a SORV may lead to core damage when SLC is unavailable. However, this would be considered a non-minimal sequence if one considers ATWS scenarios with SLC unavailable as a core damage end state.

LEVEL OF SIGNIFICANCE

C if the success criteria is amended per earlier comments.

POSSIBLE RESOLUTION

Review the functional success criteria and benchmark with that used at other plants.

PLANT RESPONSE OR RESOLUTION

The ATWS Event Tree has been revised. Standby Liquid injection is needed to prevent core damage. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 10**

Containment Heat Removal

The event tree models include all containment heat removal functions in one node at the end of the transient event tree. There tends to be a substantial interface between the type of containment heat removal system that is successful and the adverse impact that may be induced on RPV makeup systems.

An example of this potential adverse impact that does not appear to be captured is the following:

- LPCI or CS success AND CHR success could apparently yield success if containment venting were successful. However, for containment venting to be successful the containment may be depressurized during the vent. This could cause steam binding or loss of NPSH for LPCI and CS. This is apparently not accounted for.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Ensure that the functional dependency among systems is accurately modeled on the accident sequences.

PLANT RESPONSE OR RESOLUTION

The issue of problems with ECCS suction from the suppression pool is addressed with the addition of the trees EMERGENCYVENT and EMERVENTLOWLOSS. These trees address the possibility of NPSH problems with RHR or CS pumps during operation of the Hardened Vent. This issue is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element AS**	**Subelement 10**

The common cause failure of strainers is not included in the model.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Documentation should provide basis for exclusion of common cause failure of strainers during LOCA or include in the model.

PLANT RESPONSE OR RESOLUTION

This has been done by applying a new AND gate to the model that brings the probability of a LARGE LOCA and common cause failures of all low pressure ECCS strainers together. The Large LOCA value provided in a SYSTEMS F&O of 1E-4 is used. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element AS**	**Subelement 10**
Sequence ATWS_10B appears to be non-conservative since LPCI and CS may fail due to NPSH issues when the condenser is not available. Same comment applies to sequences with a SORV.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		

Address impact of low pressure injection pumps (NPSH) in ATWS sequences in which the condenser is not available.

PLANT RESPONSE OR RESOLUTION

This is addressed with the addition of the tree EMERGENCYVENT to RHR and Core Spray pump models. This tree accounts for NPSH problems during a need to use the Hardened Vent. NPSH issues regarding suppression pool water temperature are accounted for by using 260°F water temp. as a failure point for the containment which addresses the NPSH graph for the pumps. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	**Element AS**	**Subelement 14**

Loss of Heat Removal Sequences

There appears to be a need to further describe the basis for "assuming" core damage on loss of containment heat removal:

(1) Not all systems appear to have been asked; e.g., CRD could still provide long term RPV injection until containment breach.

(2) The differentiation between types of core damage events depending on the availability of different injection sources is not provided.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Consider adding detail to the model that:

a) provides CRD injection when there is a defensible technical basis, i.e., dependencies and flow rate are properly accounted for (see also AS-7)

b) clarify the Level 1 end states such that Level 2 analysis can appropriately address the plant conditions.

PLANT RESPONSE OR RESOLUTION

CRD has been added to sequences GT_4, GT_9, LOSP_3, and LOSP_6 to address issue a.). Issue b.) will be addressed during revision of the Level II model which is not necessary for MSPI.

FACT/OBSERVATION REGARDING

PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	Element AS	Subelement 14

End States

The Level 1 end states may not be sufficiently defined to allow binning into LERF categories. The following examples identify cases where the end states may have some ambiguity important to LERF determination:

- ATWS-6: This involves the failure of low pressure injection systems; however, the end state does not distinguish between too little water (Class IC) and too much water (Class IV).
- GT-9: RPV pressure could be high or low.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Clarify Level 1 End States; use the NEI functional binning scheme. (See NEI 91-04.)

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 3)	Element AS	Subelement 14

PDS

GT-9 is an example of a sequence in which the PDS is not resolved for the critical feature of RPV pressure, i.e., depressurization is not asked in Level 1.

LEVEL OF SIGNIFICANCE

D

POSSIBLE RESOLUTION

It is more typical of BWR PSAs to have determined critical aspects of the PDS affecting Level 2 in the Level 1 model. This could be considered in the future.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	Element AS	Subelement 17

Success Criteria The model was investigated and determined to differ significantly from the success criteria listed in the IPE (source of the overall success criteria). These areas all proved appropriately treated in the model, i.e., the written success criteria are considered non-conservative. The documentation should be modified. These areas of the success criteria documentation include: • ATWS: SBLC failure with level control and torus cooling is success. • ATWS: SDC with SBLC success is considered a success despite the failure of RHR suppression pool cooling • RHRSW: Not included as an injection source; this is not in the model; no technical support was identified for exclusion from the model. • LOCA: RHR in SDC is listed as a success. The model has appropriately eliminated this from the success path in the fault tree logic, but the referenced success criteria summary still includes it as a success.
LEVEL OF SIGNIFICANCE
C
POSSIBLE RESOLUTION
Place the success criteria that are used in the current PRA model in the Event Tree Notebook or in a separate notebook. Reference specific technical bases to support each success criteria. Explain implementations of the success criteria in the Event Tree notebook.
PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element AS**	**Subelement 17**

Containment Heat Removal

Containment venting or plant conditions leading to containment venting may introduce adverse impacts on RPV injection sources. These include:

- Back Pressure on RCIC (addressed)
- HCTL procedural requirements that lead to depressurization affecting HPCI and RCIC (addressed)
- Steam binding affecting LPCI and CS with suction from the torus (not addressed)

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

The treatment of RPV injection source failures due to containment conditions is considered a necessary and vital part of the sequence development process. Reflecting adverse impacts must be done to achieve the accurate reflection of sequence dependencies.

PLANT RESPONSE OR RESOLUTION

The tree EMERGENCYVENT has been added to RHR and Core Spray pump models to address the use of the Hardened Vent and the possible NPSH problems which these pumps may see during the venting. This comment is closed.

FACT/OBSERVATION REGARDING

PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 3)	Element AS	Subelement 17
Documentation does not specify the basis for auto-depressurization for the described sequences since this is not clear (see #ADED). Identifying the cause is desirable for understanding the scenario in which PCS is available and with no SORV.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Describe the causes that lead to #ADED. It is desirable to clarify the reason for depressurization in the scenario in which PCS is available and with no SORV		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 4)	**Element AS**	**Subelement 17, 19**

Containment Flooding not modeled.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Evaluate the need to model Containment Flooding from external sources.

PLANT RESPONSE OR RESOLUTION

Containment Flooding for Plant Hatch is the prime point of egress from the EOPs and an entry point for SAGs. The point for this action is the fact that you have not been able to get water on the core therefore flooding the area around the vessel will cover what leeches from the core—or will go into a possible break and allow core coverage. This is looked at as submergence of the corium on the floor of the containment during a potential LERF event. This is considered a Level II item because all methods of injection have been exhausted at this point and core damage is viewed as occurring in the Hatch PSA model. For MSPI this does not require immediate address.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element AS**	**Subelement 17**
ATWS_113 with 2 SORVs takes credit for failure to inhibit ADS for core damage. Provide the basis for this sequence since ATWS with two SORV may lead to a significant challenge with or without ADS inhibit.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Define the functional success criteria explicitly for each initiating event group.		
PLANT RESPONSE OR RESOLUTION		

ATWS sequence 113 no longer exists. The ATWS sequence that does have 2 SORVs stuck open now goes directly to core damage. Functional Success Criteria will be defined. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 19**

RPV Depressurization

The operator action to depressurize the RPV appears to have an extremely high HEP. This value is inconsistent with that developed for other BWRs and appears inconsistent with the clear definition in the EOPs, training and simulator exercises. (See also HR-10, 12, 15)

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Re-assess the interface of EOPs with the RPV depressurization modeling.

PLANT RESPONSE OR RESOLUTION

These HEPs as well as all other Hatch HEPs are being recalculated by an independent contractor. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	Element AS	Subelement 19

HRA

The following information is considered in the Team Review of AS-19:

1. Procedures have changed since that time (e.g., EOPs changed from EPG Rev. 4 to EPG/SAGs)
2. Operations Department interface on the PRA update for review of the HRA interface was not in evidence to the PRA Peer Review Team

It does not appear that the operating staff or training staff at the site are part of the PRA review cycle. This limitation may create an issue with the usability and fidelity of the model.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

The operating staff and training staff should be involved in the review of PRA updates.

PLANT RESPONSE OR RESOLUTION

As per SNC procedures the latest changes in the Hatch PSA are sent to the on-site Engineering manager for his dispersal. The keeper of the Hatch PSA is a former SRO and Hatch Operations Supervisor who maintains a constant contact with operations, work planning, engineering, and training as to what has changed and its effect. New HEP data used direct operator interviews for obtaining information. The latest revision to the model has considered changes to the ATWS EOPs and remodeled accordingly. The Hatch Simulator resides on the Hatch PSA lead engineer's computer to provide comparison as necessary. Changes are made because the operations, training, or engineering people have caused them to be made. Review of the necessary modifications to the PRA model to incorporate such items by operations personnel would not be that beneficial because they do not know the inner workings of the logic for failure in each case. There input comes from review questions regarding changes to be made. The training staff maintains the operator actions that are used in the PSA model as part of their various scenarios. This is evaluated during plant INPO visits. The complex nature of the review task is driven by the knowledge level of those requesting review. Direct procedures governing that operations and training review every change are not necessary. Indirectly, reviews are given by operations personnel by their use of the new models in the EOOS on-line risk monitor. Overall, the procedurally required information sent for all model revisions to the site engineering manager encompasses the required reviews. This comment is closed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	**Element AS**	**Subelement 20, 21**

Level 1 End States

The accident sequences transferred into the Level 2 LERF event tree are performed in a manner that allows for the potential to easily "misplace" or "disregard" Level 1 accident sequences. Specifically, the types of sequences transferred to Level 2 for the GT_7 sequence (loss of DHR) are only those with drywell failure due to overpressure failure. This means no accidents with wetwell airspace failure are included for the assessment of the shell melt-through for suppression pool bypass.

The model relies on the analyst to "know" that shell melt-through is treated as a non-LERF and that all core damage sequences are not explicitly evaluated for shell melt-through effects.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reconsider the Level 1 Binning approach.

PLANT RESPONSE OR RESOLUTION

This is a Level II concern and will be addressed with the revision of the LERF model. This is not necessary for MSPI.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 22**

MAAP Calculations

The core damage criteria used in the MAAP calculations for the IPE and the current model may differ, i.e., fuel melt versus 2200°F. This may result in longer available times from the older MAAP runs. Extreme care must be exercised in the implementation of time available for action in HRA and other assessments to ensure the appropriate recent criteria are used.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Confirm the appropriate core damage criteria are used to assess the timing of cues, actions and system required response.

PLANT RESPONSE OR RESOLUTION

This is done with the Hatch Success Criteria and the HRA up date (2005) calculation. In some cases such as station blackout core melt timing is used. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element AS**	**Subelement 24**
The documentation would be significantly improved if the failures that remove major systems from the sequence are identified or different functional gates are used in their appropriate sequences. GT_39 is an example where overpressure of the RPV causes a medium LOCA but PCS is still retained in the functional node and is defeated by inserting medium LOCA into failures of the PCS.		
LEVEL OF SIGNIFICANCE		
D		
POSSIBLE RESOLUTION		
Editorial		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element AS**	**Subelement 24**

ET Description: LOSP (p. 49)

The description of the HP node appears to be in need of clarification. It does not address:

- The mission time over which HPCI or RCIC can be functional under LOOP sequences.
- The HPCI operation with 1, 2 or 3 SORVs is not judged to be a success for a 24 hour mission time. This should be clarified or the basis for such a success criteria should be supplied.
- The RCIC system should be considered adequate to allow depressurization of the RPV for a single SORV until the RPV pressure is reduced sufficiently such that low pressure injection systems are adequate for injection and level restoration.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider the above model and text refinements.

PLANT RESPONSE OR RESOLUTION

DATA ANALYSIS (DA)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 4**

The explicit boundary discussion of component in the data collection and analysis documentation is a positive feature of the Hatch PRA.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

NONE

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DA**	**Subelement 4**

The Bayesian update analysis of generic data with plant specific data for a wide variety of components in the model is a positive feature of the Hatch PRA.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

NONE

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element DA**	**Subelement 4**
The SRVs failure to open (for RPV emergency depressurization) are modeled with a supercomponent basic event that apparently incorporates random failures, independent failures, etc.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Consider decomposing this supercomponent basic event into individual basic events (including pneumatic support, which does not appear to be modeled in the SRV fault tree logic).		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element DA**	**Subelement 4**
HPCI Failure Rate Consider updating data to ensure the latest available data reflecting current plant operation is included in the model.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element DA**	**Subelement 4, 7**

The data analysis involves a lot of good work, including:

- The number of components receiving plant specific data analysis
- Both failure rates and maintenance unavailabilities are derived using plant data
- The use of Bayesian analysis

However, the data analysis has not been updated since 1992. Update of the maintenance unavailabilities at least should be performed.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Update maintenance unavailabilities, at least, during the next PRA update, using Maintenance Rule data. If time and resources permit, also update key component failure rates.

PLANT RESPONSE OR RESOLUTION

The Hatch data has been updated through 2001. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 7**

Maintenance Unavailability

The maintenance unavailability data was not fed back from the MR recorded data.

It would be useful to incorporate the actual observed maintenance unavailability.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Include MR data.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DA**	**Subelement 7**
The maintenance unavailability data is generally pooled to represent average maintenance unavailabilities for like components. The analysis does appropriately group the 1A and 1C EDGs separately from the swing EDG.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
In the next PRA Update, use Maintenance Rule unavailability data and apply the component specific unavailability information to specific key components rather than pooling.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 8**
The Common Cause Failure Data was determined without the benefit of the NRC INEEL report.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Evaluate Common Cause Failure Data using the NRC INEEL report. (REF. NUREG/CR-6268)		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DA**	**Subelement 8**
The common cause data analysis is a positive feature of the Hatch PRA but has not been updated and was performed using common cause parameter information from the 1980's. Since that time, INEEL has recently released in the last couple years their analysis of 30,000 records and associated common cause parameters (considered to be the most commonly used cause parameter study to date).		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Consider using the INEEL CCF parameter information in the future.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element DA**	**Subelement DA-8, 14**

Generic numerical valves from PLG were used for all the MGL parameters. Use of generic numerical parameters from some reference, without a detailed evaluation of the basis of such parameters is not in conformance with NUREG/CR-4780. The values used from the PLG database may not be defensible in light of available common cause data. Now that NRC has developed a more complete data base on common cause, continued reliance on undocumented numerical values of these parameters on licensing submittals is likely to be unacceptable for regulatory use in the risk informed applications. More important, for such an important contributor as common cause, the analysis should be as realistic as possible.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Consider an updated common cause analysis using the procedures in NUREG/CR-4780 and NRC's new common cause database. See the paper by Frances Marshall at PSA'96 on the NRC/INEL common cause database or the NUREG/CR-6268 and NUREG/CR-5485.

PLANT RESPONSE OR RESOLUTION

Hatch common cause data has been totally updated using the INEEL data base (NRC INEEL report. (REF. NUREG/CR-6268)). This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 10**

The common cause failure (CCF) write-up should acknowledge that a CCF occurred at Monticello for the squib valves.

The common cause evaluation of SLC squib valves apparently did not address the operating experience in the industry related to these valves (see Attachment)

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Incorporate data or provide justification for rectification.

PLANT RESPONSE OR RESOLUTION

The value 0.014 (as provided in the Attachment) has been used for common cause failure of both SQUIB valves under basic event (CC-SL-12). This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DA**	**Subelement 10**

The number of common cause groupings identified and included in the models is a positive feature of the Hatch PRA. However, the following common cause groups are not included in the PRA:

- HPCI/RCIC common cause
- All site EDGs
- DC buses
- PSW and DGSW pumps

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Include the above groups in the model, if determined to be appropriate. Common cause failure of HPCI/RCIC should be included per INPO.

PLANT RESPONSE OR RESOLUTION

HPCI and RCIC are made by the TERRY Corporation and are steam turbines. HPCI has over 10 times the capacity of RCIC. The control features for startup are different between the turbines. RCIC has no auxiliary oil pump to open the throttle valve on startup, instead, it is open while in standby mode. HPCI has 2 pumps (a regular and a booster) in-line with each other and RCIC does not. HPCI trips on high water level, RCIC only has an isolation of the steam supply. HPCI needs room cooling while RCIC can operate without it. Common cause failure of the physical turbines is a low probability. Common cause failure of the control mechanism must consider the electrical and mechanical sides. The DC power supply for these machines is also different (Division I for RCIC and Division II for HPCI). These are different between the two machines. Common cause failure to start and failure to run have been included for HPCI and RCIC in the Hatch Rev. 2 model which considers the previous information.

Common cause modeling of Unit 1 and Unit 2 diesel generators has been incorporated in the Hatch model.

Common cause failure of the physical bus work for any switchgear is not modeled. These items are passive in nature (metal bar with wires). It is not modeled. Common cause failure of batteries is likewise not modeled because they are passive components. The frequent testing and inspection of the batteries tends to preclude catastrophic common cause failure. This is not considered to be a necessary function.

PSW pumps are modeled for common cause between each unit. These pumps do not share inter unit functions as is the case with the diesel generators and the supply to the LPCI buses (1R24S018A and B). The most logical case for common cause failure with regard to all (Units 1 and 2) PSW would be with a problem with the Intake structure. This is modeled with initiating event (i.e. &INTAKE).

The diesel generator standby service water pump is a smaller pump than the other PSW pumps and it serves only to cool the shared diesel. It has no common cause relationship with the exception of the blocking or loss of the Intake structure.

This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 11, 12**
The fact that HPCI/RCIC, PSW/DGSW, and all site EDGs have not been grouped is indicative that a formal assessment of potential appropriate asymmetrical common cause groups (i.e., cross-system, cross-unit, asymmetrical component design but other common features exist, etc.) has not been performed.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
When time and resources permit, perform a formal assessment of asymmetrical common cause groupings and include any new groups into the models.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 13**

CCF

The common cause failure probability of the SRVs to open for depressurization appears not to be based on operating experience. There have been precursors for such failures. It is judged that failure of sufficient SRVs to open is underestimated in the PRA. Operating experience should be reflected in the analysis.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reassess the CCF probability for SRVs to open for depressurization to use operating experience and the correct success criteria. (See 2 pages attached)

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS
The failure to open for SRVs is modeled as stated with common cause for the various number of SRVs needed to open in certain situations. The events PR1, PR8, PR8B, and PR10 model these cases. In order to evaluate the affectiveness of the attached information the PR numbers (except for PR10 which deals solely with ATWS) were raised to arbitrarily high values compared to those presently used in order to perform a sensitivity study. PR1 is 2E-6 and was raised to 3E-4. PR8 is 2E-4 and was raised to 3.4E-4 and PR8B is 2E-8 and was raised to 1.47E-4. The model was quantified and negligible change was noticed. Plant Hatch has had experience with SRVs and continually evaluates their performance. However, the attached information does not provide anything significant to SRV failure to open, in fact the information, does not provide failure data—only speculation. This data will not be included in the Hatch model. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 14**
The number of common cause groupings identified and included in the model, and their method of quantification, is a positive feature of the Hatch PRA.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
NONE		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DA**	**Subelement 19**
The basic event nomenclature is not uniform or consistent (e.g., operator actions basic events not uniform, some begin with OP, some do not; independent failure basic events begin with CC-, etc.).		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Update the basic event IDs to be consistent and uniform during the next update.		
PLANT RESPONSE OR RESOLUTION		

DEPENDENCY ANALYSIS (DE)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 1**

No overall discussion or guidance exists regarding the location of and treatment of the various aspects of dependent issues in the model (e.g., inter-system dependencies, human interactions, common cause, spatial interactions, etc.).

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Develop a guidance document or provide PRA documentation that directs/discusses the treatment of inter-system dependencies, human interactions, common cause, spatial considerations, etc.

It was discovered on Wednesday of the Review that a number of documents or calcs. that essentially represent guidance/methodology discussions (e.g., Wok Package H0) exist in the PRA filing cabinets. An effort should be made to go through these files to identify such discussions and to collect them in a binder in individual tabbed sections (potentially following the NEI Review Elements) to represent at least an initial start at a set of PRA Guidance Documents. Minimal effort can be expended in the blank sections to provide a one page summary of acceptable approaches (again, as a start in developing such a Guidance Document).

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 3**
System notebooks contain system to system and support system dependencies. The dependency matrix presents this information in one convenient location. Dependency matrix is very detailed with good use of notes to describe the effects on other systems.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 4**
The system to system dependency matrix and associated descriptive notes is a positive feature of the PSA. A useful enhance would be a similar matrix that summarizes system dependencies as a function of initiating event category.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Produce and document a initiator vs. system dependency matrix.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DE**	**Subelement 4**
Suction of ECCS is through a design that could be subject to steam binding when the containment fails with pool at elevated temperature.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Investigate the possibility of suction failures.		
PLANT RESPONSE OR RESOLUTION		

A special tree named EMERGENCYVENT has been added to the model to address the loss of NPSH for low pressure ECCS during venting of the containment. Typical failure temperature for the containment based on water temperature of the suppression pool is 260°F which is slightly above the NPSH curves for the low pressure ECCS pumps. Venting of the suppression chamber will occur prior to this temp. If venting fails in these cases (assuming no other form of containment heat removal is available) then the pumps and containment fail at 260°F. The EMERGENCYVENT tree takes into account the failure possibilities after you exceed the NPSH abilities and prior to 260°F. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element DE**	**Subelement 4**

Dependency Matrix HPCI/RCIC

The dependency matrix lists HPCI and RCIC as completely dependent on S.P.

The nature of this dependency needs to be explained.

The system notebook does not address this complete dependency on S.P.

It does not address the CST volume and the capability of the CST to be an adequate supply of water.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Clarify the RCIC and HPCI dependency on S.P.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 5**
The LOCA accident sequences do not question the Vapor Suppression function.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include Vapor Suppression in the MLOCA and LLOCA accident sequences.		
PLANT RESPONSE OR RESOLUTION		
A vapor suppression model is now included in MLOCA and LLOCA sequences. The model is an AND Gate called, VAPSUPPRESSION. This comment is closed.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 6**

Dependencies are modeled and listed in the System Notebooks as well as in the System Dependency Matrix.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DE**	**Subelement 6**
The detailed system-to-system dependency matrix and the linked fault tree model and their support in identifying and processing dependencies are a good feature of the Hatch PRA program.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
N/A		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 7)	**Element DE**	**Subelement 7**
The operator action basic event OPHEHP8 for failing to control level will appropriately fail HPCI and RCIC. However, based on the discussion with Hatch personnel regarding the Hatch trip earlier this year, it appears that this event should in some manner (i.e., either directly or with some other conditional event) also fail the main condenser (which it currently does not). Apparently, in the Hatch trip of earlier this year the operators failed to control level with HPCI, causing the operators to manually close the MSIVs.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Review the model and documentation with respect to the above discussion to determine whether the model appropriately models the Hatch trip of earlier this year.		
PLANT RESPONSE OR RESOLUTION		

An abnormal procedure is now used to address the case involving HPCI or RCIC not tripping automatically on high level. The operator only closes the MSIVs if the source of injection cannot be stopped. The operator action, OPHEHP8, accounts for the operator not manually stopping the injection source—it does not account for a runaway injection source. The steam lines as referenced in the abnormal procedure can take the water put in them and they can actually pass the moisture to the main condenser through the bypass valves (the main turbine is tripped in this case). HPCI and RCIC turbines can take water injection due to the strength of their blades and wheels. SRVs can operate with water in the steam lines as well. The MSIVs do not have to be closed because of overfill—if the source can be stopped.

It would difficult to put so much water in the steam lines that the main condenser was lost, but it is possible to degrade HPCI and RCIC (this is why HPCI is lost on overfill in the model). The possibility that the operators closed the MSIVs first then shut off HPCI is very small today for the event where the operators fail to monitor HPCI and water level like they should. However, during the referenced event, the level indication showed over 100 inches which by the old methods and procedures required the MSIVs to be closed—whether or not the source of injection had been stopped.

34AB-C32-001-1or 2 allows the MSIVs to stay open if the source is stopped. This change is made to purposely not have to shut the MSIVs just because water level is starting to spill into them---if you can isolate the source.

This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 8**
There is no common cause failure of HPCI and RCIC in the PSA even though an INPO evaluation shows that there is linkage between the two systems.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include HPCI/RCIC common cause term in the models.		
PLANT RESPONSE OR RESOLUTION		

Previous comments made in these F&Os regarding this issue show that HPCI and RCIC at Hatch have very little in common except they are TERRY turbines. Nevertheless a common cause basic event for failure to run and one for failure to start have been added to the HPCI and RCIC fault trees in the PRA model. HPCCR is common cause failure to run and HPCCS is common cause failure to start. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DE**	**Subelement 8**
Common cause data is from old database.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
During next update consider updating common cause data with most recent common cause data. Consider using INEEL's latest database.		
PLANT RESPONSE OR RESOLUTION		
This has been done for this update, Revision 2.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element DE**	**Subelement 8**
The room-by-room discussion of room cooling issues is a positive feature of the documentation. This assessment should be enhanced by direct reference to available plant analyses (e.g., Station Blackout Rule Coping Studies) and their key assumptions, details, and conclusions.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Provide such referencing in the documentation.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element DE**	**Subelement 8**
Previous Fact & Observations have commented re: missing terms for HPCI/RCIC common cause and common cause for EDGs across the both units. This may indicate that a formal investigation of appropriate asymmetrical common cause groups has not been performed.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
If resources are available, investigate cross-system asymmetrical common cause cases (e.g., cross-system, cross-unit) that may be appropriate to include in the models. For example, one plant with HPCS created common cause terms for HPCS and LPCS pumps because they are very similar in design and manufactured by the same company. Plants with EDGs of different design include common cause terms across the EDGs due to issues such as common fuel oil, etc. Also, it is common for multi-unit plants to include a common cause term for all EDGs across the site.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element DE**	**Subelement 8**

DG CCF

The identification of critical CCF potential failures assists in risk management.

All DG cooling water discharges to the same 30" discharge pipe to the river. Has the total blockage of the discharge pipe been considered as a CCF of all Diesels. (It has a 95 psig rupture disk in the 30" line also.)

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Ensure that this CCF source is examined and included numerically and as a separate basic event, if justified.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 6)	Element DE	Subelement 8

ECCS Suction Strainer

It is notes that the passive failure of ECCS suction strainers is included in the Hatch model:

- Each pump (RHR or CS) has its own strainer inside the torus and plugging is modeled for each of these strainers (basic events STPL1E11A (B, C, D) for RHR and STPL1E21L001A (B) for CS), each with a probability of 1.49E-4. There is no single plugging event modeled that fails all suction from torus.

There is however no CCF of all strainers due to debris clogging. This has been included in numerous BWR PSAs to model the extremely unlikely event of debris clogging. It is recognized that Hatch has modified the ECCS suction strainers to prevent this failure mode. Typical values are:

	CCF
Large LOCA	1E-4
Medium, Small LOCA	1E-5
Transient	1E-6

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider adding the ECCS suction strainer common cause failure

PLANT RESPONSE OR RESOLUTION

This has been done in Revision 2 of the model. The basic event is named CCFAILURE.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 9**
The number of common cause groupings identified and included in the model, and their method of quantification, is a positive feature of the Hatch PRA.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
N/A		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DE**	**Subelement 9**
NUREG/CR-4780 methodology was used for a systematic approach to provide plant specific grouping of similar system components for common cause treatment.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element DE**	**Subelement 10**
The level of effort and documentation regarding the internal flooding analysis is a positive feature of the Hatch PRA.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element DE**	**Subelement 10**
The room-by-room discussion of room cooling issues is a positive feature of the documentation. This assessment should be enhanced by direct reference to available plant analyses (e.g., Station Blackout Rule Coping Studies) and their key assumptions, details, and conclusions.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Provide such referencing in the documentation.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element DE**	**Subelement 10**
The room cooling discussion acknowledges that loss of room cooling in the DG Building 4kV room may lead to long-term loss of 4kV buses, but this is not modeled. The basis is tied to the assumption stated in the room cooling assessment that the operators would open bus room doors in the long-term. However, no such procedural direction exists in the AOPs.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Rectify this dependency issue and modify the documentation and/or models as appropriate. It is not uncommon in other BWRs to require 4kV room cooling in the long-term and to credit alternate room cooling activities (e.g., open room doors).		
PLANT RESPONSE OR RESOLUTION		
The item is being evaluated by SNC engineering and Bechtel engineering at present. It is not considered to be of such importance that the buses would fail. Until the evaluation is complete the model will stay as is.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element DE**	**Subelement 10**

Internal Flooding

The Turbine Building Flooding scenarios do not appear to be developed. It is believed that Turbine Building Flooding may be worthy of further consideration to address the following items:

Circulating water break in the Turbine Building of either unit could cause loss of all instrument air compressors. This results in loss of main condenser and degrades the support for the containment vent.

Batteries have water tight doors (no impact).

If "flood switches" in Turbine Building basement are miscalibrated and fail to trip circ water or isolate PSW, then continued flooding could lead to control building equipment damage. These scenarios should be assessed as to their frequency.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider T.B. flood scenarios either in the screening method or in an explicit quantification.

PLANT RESPONSE OR RESOLUTION

The condenser bay is isolated from the control building elevation 112 where the air compressors are. There may be seppage through the walls during a catastrophic line break but typically not enough to damage the compressors. The loss of circulation water would cause the condenser to loose vacuum and the MSIVs would close during the flood anyway. Since this has never happened (large scale circulating water line break) at Hatch the initiating event frequency would be low. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element DE**	**Subelement 10**
Intake Anomalies There doesn't appear to be any common cause treatment of intake failures that could interrupt PSW, RHRSW and circulating water. These could include trash blockage, low sever level, ice frazil.		
LEVEL OF SIGNIFICANCE		
POSSIBLE RESOLUTION		
Consider CCF intake failures.		
PLANT RESPONSE OR RESOLUTION		

The motors of PSW and RHRSW pumps are qualified for the environment of the intake structure. They have the ability to maintain operation with the fire sprinklers activated. Room cooling has been addressed within the PSA model by failing the PSW pumps on loss of intake structure fans. A sensitivity study was performed on the Hatch PSA model where a common cause basic event was inserted in the model for common cause strainer plugging. The value used was 1E-5 which considering individual failure of each strainer was left at 1.31E-4 is reasonably conservative. Core damage frequency change was in the low E-9 range. This does not warrant inclusion of a common cause value. Common cause failure of the strainers is more closely modeled by the initiating event , &INTAKE, loss of intake structure. The strainer condition and its affects on PSW is annunciated in the main control room. In addition at least once every 24 hours the strainer differential pressures are monitored and if necessary the strainers are manually rotated—they are typically in an automatic operational mode. Strainer failure would be no more than a failure of the rotating mechanism or the drain valve for the backwashing sequence. Both items can be operated manually if need be. Strainer plugging is not considered to be an instantaneous problem. In fact it is long term and would spotted prior to complete failure. Operator activity to repair the problem would be allotted enough time to make the HRA for such an event very small.

RHRSW is operated intermittently. Failure of the strainers is modeled with an operator action to swap them if necessary. Considering that dual trains of RHRSW exist with two strainers per train, common cause failure of the total mechanical system would be of low worth.

Common cause failure of the PSW and RHRSW strainers will not be modeled.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 6)	**Element DE**	**Subelement 10**
CCF for PSW and RHRSW Strainer clogging, room environment, or sprinkler spurious operation does not appear as CCF basic events for PSW or RHRSW.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include CCF of these by room environment, sprinklers, and strainer clogging, if applicable.		
PLANT RESPONSE OR RESOLUTION		

The motors of PSW and RHRSW pumps are qualified for the environment of the intake structure. They have the ability to maintain operation with the fire sprinklers activated. Room cooling has been addressed within the PSA model by failing the PSW pumps on loss of intake structure fans. A sensitivity study was performed on the Hatch PSA model where a common cause basic event was inserted in the model for common cause strainer plugging. The value used was 1E-5 which considering individual failure of each strainer was left at 1.31E-4 is reasonably conservative. Core damage frequency change was in the low E-9 range. This does not warrant inclusion of a common cause value. Common cause failure of the strainers is more closely modeled by the initiating event , &INTAKE, loss of intake structure. The strainer condition and its affects on PSW is annunciated in the main control room. In addition at least once every 24 hours the strainer differential pressures are monitored and if necessary the strainers are manually rotated—they are typically in an automatic operational mode. Strainer failure would be no more than a failure of the rotating mechanism or the drain valve for the backwashing sequence. Both items can be operated manually if need be. Strainer plugging is not considered to be an instantaneous problem. In fact it is long term and would spotted prior to complete failure. Operator activity to repair the problem would be allotted enough time to make the HRA for such an event very small.

RHRSW is operated intermittently. Failure of the strainers is modeled with an operator action to swap them if necessary. Considering that dual trains of RHRSW exist with two strainers per train, common cause failure of the total mechanical system would be of low worth.

Common cause failure of the PSW and RHRSW strainers will not be modeled.

FACT/OBSERVATION REGARDING

PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	Element DE	Subelement 13
The System Dependency Matrix has review signatures for each system as well as a review signature for the System Dependency Matrix package.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

HUMAN RELIABILITY ANALYSIS (HR)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 1, 28**
The guidance for performing the various facets of the HRA are generally defined in two or more calculations and in the IPE submittal. No single HRA document exists.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
A single HRA document that covers the guidance and all the current HEP calculations in the PSA could be an enhancement. Such an approach to documentation is becoming a very useful, if not necessary, feature for applying and maintaining PSA models in the industry.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 2**
HRA SNC has chosen not to update the HRA for Hatch because of lack of consensus on acceptable methods of HRA. HRA remains the same as in IPE, which is 8 to 10 years old.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Consider an HRA update to reflect the latest procedures, training, hardware, and model changes.		
PLANT RESPONSE OR RESOLUTION		
Hatch HRA has been updated using the HRA Calculator by SCIENTECH. This comment is closed.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 3**

Process for calculating HEPs includes applying Monte Carlo sampling routines in RISKMAN. RISKMAN is no longer the PRA code in use by SNC. Will this process still apply?

This is an example of an area where the transition from RISKMAN methodology may make the current techniques insufficient to provide guidance in the future.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Clarify in future guidance documentation.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element HR**	**Subelement 3**

A variety of HRA methodologies exist to quantify human error probabilities. The FLIM method was employed for the Hatch post-initiation HRA and produces reasonable results. However, a key part of the FLIM method is the formation of a group of experts to provide qualitative rating for performance shaping factors.

Again, the method produces reasonable results, but the Review Team questioned the ease of updating the HRA or performing applications involving changes in action timing.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Ensure that the current methodology can be reproduced at will and is not hampered by the need to identify and poll a group of experts. If the FLIM process is not easily applied at will, consider the ASEP process or the EPRI Cause Based method.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element HR**	**Subelement 3**

EPU

The extended power uprate resulted in reducing the time available for operators to take action in a number of events such as Emergency depressurization. The FLIM HRA method is not very sensitive to the small timing changes calculated. The evaluation process is superior.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 4, 5**

<u>Pre-Initiator</u>

The identification of pre initiator HEPs is an excellent process.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 6**

Low Pressure Permissive

It is an excellent feature that the low pressure permissive is treated for miscalibration error.

The value derived of 1.3E-5(MIUNNS) appears lower than might be derived using THERP methods.

The recovery of the Low Pressure Permissive failure is described in the IPE as only available for cases where substantial time is available to manipulate the valves. The IPE also indicated that it is to be applied during testing. This restriction is not included in the PRA update. The recovery credit appears to be applied to all applicable cases except Large LOCA or cases w/o RPV high pressure injection.

This appears to neglect cases involving ATWS, medium and small LOCAs or IORV--all of which would appear to be clearly stressful situations with reduced times available.

In addition, the time available to bypass or "fix" the permissive is potentially only known or cued following RPV depressurization, so the time available can be very short. This limited time available is not addressed in the derivation of this recovery.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Re-examine the derivation of the HEP to account for the timing available to take action and the associated stress level. Ensure that the application of FLIM in the original HRA properly characterizes the conditions, i.e. permissive undergoing test or not.

PLANT RESPONSE OR RESOLUTION

MIUNNS has been renamed for each event that it applies to. In any case this is not a recovery. This is the probability that a miscalibration occurred and the new value, 2.7E-7, has recently been recalculated for the HRA update. The recovery of an instrument channel that is failed is now given a 1.0 or total failure value in the model—because of the reasons mentioned in this certification comment. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 7**

Pre Initiator

The calculation of the miscalibration of two separate trains of logic appears to consider the HEP for the two trains completely independently. This results in a relatively low HEP for miscalibration of 1.3E-5. Values from other methods including potential for common cause effects have led to HEPs of 8E-5 to 2E-4.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reassess the miscalibration HEP to ensure common effects (i.e., common error potential) are accounted for:

- Same crew
- Same day
- Same standard (calibration device)
- Same written procedure

PLANT RESPONSE OR RESOLUTION

The MIUNNS value has been reevaluated with the HRA update by SCIENTECH. Calibration is very rarely done except when a functional test of a trip channel shows this case to be necessary. Typical calibrations are performed once every outage as per Technical Specifications, but are allowed, if the channel under an FT&C test needs calibration. The functional test and calibration is typically performed on a set of channels in one card rack at one time. The FT& C as they are called can take upwards of 12 hours for a complete cabinet. This cabinet typically houses one division of several trips which would not cause a problem because the other channel in the division (in another cabinet) is still available. The FT&C has a range of acceptable values and involves one person to check alarms in the control room and a performer and checker at the panel. A consistent major error in one panel's FT&C would not inop the entire system. The FT&Cs are performed on a staggered basis at least one cabinet every 3 months or on an alternate basis with 2 cabinets every 3 months and the likelihood of the same crews doing all the cabinets and spreading their "error" is very low. Common cause miscalibration errors are very low. A sensitivity study was performed on the Rev.2 model on August 10 using a common cause value of MIUNNS inserted in every tree where the single failure (MINUNNSx) resides. MIUNNS was given the value 8E-5 which is considered very high. Core damage went for 7.91E-6 to 7.93E-6 which is considered negligible. The results of this sensitivity value and the previous discussion are used to negate the inclusion of calibration error common cause. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 10**

Training Interface

There are two items that would indicate a need to reassess the operations input to the PRA. These two items are:

- The PRA model has shown that RPV depressurization is a critical operator action of high importance.
- The HEP for these actions are quite high compared with other BWR PSAs.
- The operations input is documented to be biased by the postulated scenario and the impact of RPV level anomalies (perceived) at the time of the interviews.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Revise the HEPs and interviews for the critical HEPs to see if conditions at the plant and better definition of the scenario will modify the assessed HEP particularly for emergency depressurization so that the PRA will not have biased results.

PLANT RESPONSE OR RESOLUTION

These items are addressed in the PSA HRA revision using HRA Calculator by SCIENTECH. In addition items at the plant (in the control room) are now used to ease the stress on the operators for meeting level requirements (the use of SPDS). These items are taken into account in the new HRA numbers. The HRA revision is done for this model rev (Rev.2). This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element HR**	**Subelement 10,14**

Containment Vent

The availability of LPCI and CS for RPV injection following containment vent is assumed in the model. There is no documentation of the procedural or training guidance that would support this assertion. This is a major assumption and should be supported by operator crew input and a discussion of the configuration of the low pressure injection system suction pipe (e.g., steam binding potential).

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reconsider the LPCI/CS operability when venting is initiated. Provide justification for continued operation that addresses steam binding potential and loss of adequate NPSH.

PLANT RESPONSE OR RESOLUTION

A new tree called EMERGENCYVENT has been added to the model specifically to address failure probability of the low pressure ECCS during venting. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 12,15**

RPV Depressurization

The operator action to depressurize the RPV appears to have an extremely high HEP. This value is inconsistent with that developed for other BWRs and appears inconsistent with the clear definition in the EOPs, training and simulator exercises. (See also HR-10).

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reassess the RPV depressurization HEP.

PLANT RESPONSE OR RESOLUTION

These items are addressed in the PSA HRA revision using HRA Calculator by SCIENTECH. In addition items at the plant (in the control room) are now used to ease the stress on the operators for meeting level requirements (the use of SPDS). These items are taken into account in the new HRA numbers. The HRA revision is done for this model rev (Rev.2). This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 14**

HRA

The following information is considered in the Team Review of HR-14:

1. Operating staff was part of the HEP assessment with FLIM during the original HRA
2. Procedures have changed since that time (e.g., EOPs changed from EPG Rev 4 to EPG/SAGs)
3. Operation interface on the PRA update for review of the HRA interface was not in evidence to the PRA Peer Review Team

It does not appear that the operating staff or training staff at the site are part of the PRA review cycle. This limitation may create an issue with the usability and fidelity of the model.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

The operating staff and training staff should be involved in the review of PRA updates.

PLANT RESPONSE OR RESOLUTION

Operator interviews and training staff interviews were conducted for the recent HRA update by SCIENTECH.
As per SNC procedures the latest changes in the Hatch PSA are sent to the on-site Engineering manager for his dispersal. The keeper of the Hatch PSA is a former SRO and Hatch Operations Supervisor who maintains a constant contact with operations, work planning, engineering, and training as to what has changed and its effect. New HEP data used direct operator interviews for obtaining information. The latest revision to the model has considered changes to the ATWS EOPs and remodeled accordingly. The Hatch Simulator resides on the Hatch PSA lead engineer's computer to provide comparison as necessary. Changes are made because the operations, training, or engineering people have caused them to be made. Review of the necessary modifications to the PRA model to incorporate such items by operations personnel would not be that beneficial because they do not know the inner workings of the logic for failure in each case. There input comes from review questions regarding changes to be made. The training staff maintains the operator actions that are used in the PSA model as part of their various scenarios. This is evaluated during plant INPO visits. The complex nature of the review task is driven by the knowledge level of those requesting review. Direct procedures governing that operations and training review every change are not necessary. Indirectly, reviews are given by operations personnel by their use of the new models in the EOOS on-line risk monitor. Overall, the procedurally required information sent for all model revisions to the site engineering manager encompasses the required reviews. This comment is closed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	**Element HR**	**Subelement 15**

Recovery

The following recovery action is developed in the PRA documentation and is included directly in the model: QRA: Recover any DHR System.

This recovery is believed to be optimistic and the justification inadequate to support the assessed value. The following information is offered to assist SNC in understanding what other BWRs are doing in this area of recovery.

Typical approaches include the use of an exponential repair assumption over the 20 to 30 hour time frame of the loss of DHR accident. This leads to approximately 0.28 non-recovery probability instead of 0.1.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Modify the recovery problem and ensure it is not applied to #QT where recovery is needed by 6 hours. Ensure that QRA is not applied to the main condenser.

Specifically, QRA does not apply to the main condenser if the description of the RISKMAN QR split fractions still apply, i.e., are not superseded. The QR recovery is supposed to apply at a time after the vent pressure is reached, but the MSIVs cannot be opened with the containment pressure above the vent pressure.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element HR**	**Subelement 15**

Recoveries

Low Pressure Permissive Bypass

The recovery of the low pressure permissive failure is derived in the IPE documentation and is attributed to test recoveries.

The application to the CAFTA model applies the recovery to any failure of the low pressure permissive. This appears to be inappropriate given the derivation and what can reasonably be expected of the crew under severe stress associated with loss of all high pressure injection and the need for emergency depressurization.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Remove the recovery "credit" for low pressure permissive failures and miscalibrations.

PLANT RESPONSE OR RESOLUTION

There is no recovery credit for any item involving instrumentation at present. The restoration action is set to 1.0. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element HR**	**Subelement 15**

Recovery

The following recovery action is developed in the PRA documentation and is included directly in the model:

MCC: Bypass the MSIV Closure (applied during ATWS events to restore PCS)

This recovery is believed to be optimistic and the justification inadequate to support the assessed value. The following information is offered to assist SNC in understanding what other BWRs are doing in this area of recovery.

This recovery is close to 1.0 in all BWRs reviewed as part of the PRA Peer Review process.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Remove credit for this recovery unless there are plant specific procedures and training that make this viable under ATWS time restricted conditions.

PLANT RESPONSE OR RESOLUTION

MCC is no longer in the model. However, the value FCA, is used to describe the allowance to jumper around the MSIV low water level closure trip. This is only used during the ATWS case of turbine trip where there is the ability to feed the reactor enough water to prevent a rapid low level condition which would cause the MSIVs to close. The ATWS cases involving MSIV closure and loss of feedwater do not take credit for the ability to jumper the MSIV low reactor level closure because the water level closure point would be reached too quickly. The present value for FCA is 3.3E-2. A value for sensitivity was added equal to 0.7. The difference in CDF was within the E-7 range. The figures of merit for MSPI were evaluated and had negligible change with this number, therefore, it is concluded that this value does not affect MSPI. It does however change the ATWS contribution. Therefore based on engineering judgement the HRA calculated value for FCA will be changed to 0.7. Total failure of this event is dependent on the operating crew. It is possible to have a 100% ATWS and a turbine trip. If the bypass valves function and the Recirc Pumps are tripped, reactor power will be 40 to 50% without water level considerations. The terminate and prevent steps for the case where power is above %5, the suppression pool is above 110°F, water level is above -155, and an SRV is open or cycling will be reached. Providing the order to bypass the MSIV low water level trips was given prior to starting the termination of injection, it could be finished prior to reaching -101 inches (the trip or closure point). This is hard to evaluate consistently with regards to timing. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element HR**	**Subelement 15**

Recoveries

DC REC

The recovery of DC power is applied in the model to recover:

- Breaker failure
- Panel hardware failure

The application of the recovery has two potential items that are useful to provide additional information or modify the evaluation:

- The conditional probability of 0.01 does not appear reasonable based on comparison with other plant PRAs reviewed by the BWROG.
- The application of the recovery to hardware failures is inconsistent with the derivation.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Provide a better justified estimate for the recovery and do not apply it to portions of DC failures that are attributable to hardware failures.

PLANT RESPONSE OR RESOLUTION

DCREC1 is the recovery in question. It has been given a new value equal to 0.15. In addition it is now only used for recovery of a panel (125VDC, R25S001) which was at one time mistakenly deenergized during performance of a tag-out clearance. This eliminates the application to hardware failures referred to in the comment. The panel was recovered. DCREC1 is ANDed with the special initiator that models the failure of the panel. The recovery value for a panel which tripped on a fault as opposed to failure of a component is justified and fairly modeled. It was calculated by SCIENTECH as part of the HRA upgrade for the HATCH model.
This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element HR**	**Subelement 15**
The majority of the HEPs are best estimate calculations; however, a number of screening HEPs (e.g., start 2nd CRD pump, start 2nd CWS, X-tie Nitrogen) have also been included in the model.		
LEVEL OF SIGNIFICANCE		
C (The HEP events in question are non-significant contributions to the overall model results).		
POSSIBLE RESOLUTION		
When time and resources permit, consider performing realistic HEP assessments on the various screening HEPs in the model.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	Element HR	Subelement 16

Bypass the Level 1 MSIV closure Interlock

The model evaluation includes the possibility under ATWS conditions that the TBVs and MSIVs will be open for events such as:

EVENT	Probability
• TT	7E-3
• Loss of FW	4.2E-2
• Loss of Condenser Vacuum	<1.0

The assessment of these conditional probabilities is necessary to ensure that the model quantitatively reflects the plant and operating crew response.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reassess the conditional probabilities.

PLANT RESPONSE OR RESOLUTION

Basic events MCA and MCC which are what the above discussion is about have been removed from the model. FCA is the only device accounting for the MSIV low water level closure bypass. This is address in a previous comment. The following is a copy of that narrative.
MCC is no longer in the model. However, the value FCA, is used to describe the allowance to jumper around the MSIV low water level closure trip. This is only used during the ATWS case of turbine trip where there is the ability to feed the reactor enough water to prevent a rapid low level condition which would cause the MSIVs to close. The ATWS cases involving MSIV closure and loss of feedwater do not take credit for the ability to jumper the MSIV low reactor level closure because the water level closure point would be reached too quickly. The present value for FCA is 3.3E-2. A value for sensitivity was added equal to 0.7. The difference in CDF was within the E-7 range. The figures of merit for MSPI were evaluated and had negligible change with this number, therefore, it is concluded that this value does not affect MSPI. It does however change the ATWS contribution. Therefore based on engineering judgement the HRA calculated value for FCA will be changed to 0.7. Total failure of this event is dependent on the operating crew. It is possible to have a 100% ATWS and a turbine trip. If the bypass valves function and the Recirc Pumps are tripped, reactor power will be 40 to 50% without water level considerations. The terminate and prevent steps for the case where power is above %5, the suppression pool is above 110°F, water level is above -155, and an SRV is open or cycling will be reached. Providing the order to bypass the MSIV low water level trips was given prior to starting the termination of injection, it could be finished prior to reaching -101 inches (the trip or closure point). This is hard to evaluate consistently with regards to timing. This comment is closed.

This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 17, 20**
The post-diagnosis manipulation for the HEP calculations are not discussed and it is not clear if or how this issue is treated in the HEP calculations.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Ensure that the post-diagnosis manipulation times are part of the post-initiation HRA process. If they already are but the issue is one of documentation, clarify the documentation.		
PLANT RESPONSE OR RESOLUTION		

HRA for the Hatch Rev. 2 model has been done by SCIENTECH using HRA Calculator and the calculation produced accounts for timing. The typical scenario for an HRA uses the Post-Diagnosis part as the actual time to physically perform the event. This comment is considered closed because the HRA document (calculation) addresses this adequately.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 18**
The time available to perform post-initiation actions are generally discussed in most of the HEP calculations, but are often not clearly tied to MAAP runs or similar information. It is recognized that not all HEP calculations are directly related to MAAP run results and that judgments and qualitative discussions are appropriate in certain cases.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
When time and resources permit, and the HRA is updated, tie the HEP, when appropriate, to available MAAP runs or similar information.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element HR**	**Subelement 18**

The HEP basic event DEA (RPV Emergency Depressurization, non-ATWS) is documented as having a time available of 60 min for RPV water level to drop to -163" plus an additional +60 min to reach significant core damage. Based on comparison with the other BWR T&H calculations, two hours seems longer than can be justified. No calculation was presented to the Team to confirm the 2 hour time. A time available of 30-60 min is typical in other BWR HRAs for this action.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

During the next update, confirm the acceptability of the time available of 2 hours for the RPV emergency depressurization HEP; or, reassess as appropriate.

PLANT RESPONSE OR RESOLUTION

This comment is correct. DEA has been replaced by DE2 and timing can be up to 1 hour however the calculation uses less than one hour. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 19**

Level Indication

The FAI calculations cited as the basis for accident sequence timing do not provide a description of the RPV water level that the MAAP calculation corresponds to:

- Core level
- Shroud level
- Fuel zone indicated level

The perceived RPV water level in the control room is what will dictate the operator cues and the operator actions. The perceived levels may differ significantly from the MAAP calculated core level.

See the level correction procedure for the fuel zone instrumentation which is required for ATWS and emergency depressurization actions.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Provide a description of how to interpret the FAI MAAP calculations relative to what the operators will see in the control room.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 22**
A substantial amount of excellent work was involved in the HRA for the Hatch IPE submittal. However, the analysis was performed 10 years ago and plant procedures have changed in that time (e.g., to EOPs/SAGs) and training on certain scenario/procedures may also have changed.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
When time and resources permit, the HRA should be reviewed or updated against current procedures and training.		
PLANT RESPONSE OR RESOLUTION		

<table>
<tr><td colspan="3">FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS</td></tr>
<tr><td>OBSERVATION (ID: 1)</td><td>Element HR</td><td>Subelement 23</td></tr>
<tr><td colspan="3">The Hatch PRA generally does not include recovery actions unless a procedure is available. 4KV room cooling is not required in the model because it is not needed in the short term and in the long term. (Based on the Peer Review Team walkdown, this appears reasonable.) The model documentation states that the operator will open the room doors; however, no AOP procedural direction exists to open the 4KV room doors upon loss of room cooling. In addition, the opening of doors may exacerbate the problem because of its location next to the EDGs.</td></tr>
<tr><td colspan="3">LEVEL OF SIGNIFICANCE</td></tr>
<tr><td colspan="3">C</td></tr>
<tr><td colspan="3">POSSIBLE RESOLUTION</td></tr>
<tr><td colspan="3">In the next update, verify whether room cooling is required, and if required verify that such a procedural directive exists to open doors or that plant training obviates this course of action. Also, consider including a recovery HEP for loss of 4KV room cooling in the models.</td></tr>
<tr><td colspan="3">PLANT RESPONSE OR RESOLUTION</td></tr>
<tr><td colspan="3"></td></tr>
</table>

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 26, 27**

Recoveries-Sensitivity-Dependence

Certain "recoveries" are not included in the HRA sensitivity cases. These include:

- MCC
- QRA

Part of the problem with the sensitivity may have been that the model nomenclature does not allow an easy search for all basic events that are to be part of the sensitivity.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Include all HEPs, recoveries, and other operator interface actions (MCC, QRA) in the sensitivity assessment for HRA.

PLANT RESPONSE OR RESOLUTION

MCC and QRA were numbers from the original Hatch RISKMAN PRA model. Their basis is indeed more judgement than fact. These have been eliminated from the Hatch model because there is no sound basis for them. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element HR**	**Subelement 26, 27**

Operating Crew Actions and Dependence Evaluation

There are three issues that can be discussed relative to the treatment of dependent HEPs:

- Are the HEPs in the same cutset searched for and identified
  - For dynamic actions and certain recoveries, but does not include certain other actions (e.g., QRA, MCC)
- Are HEPs evaluated
  - Yes, with the above exception
- Is a "floor" on the lowest HEP or HEP combinations addressed:
  - No: This is inconsistent with the draft ASME PRA Standard

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Evaluate the HEP dependencies.

PLANT RESPONSE OR RESOLUTION

The Hatch HRA upgrade performed by SCIENTECH is calculated as per the ASME standard requirements. Two actions are being used here to generalize an entire system which is inadequate. Considering that these actions are removed and our HRA will be calculated as per ASME standard requirements, this comment, like its previous counterpart is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element HR**	**Subelement 26, 27**

Dependence

There generally is a floor applied to the HEP combination. This "floor" could vary from 1E-6 to 5E-7 depending on the justification.

Cutsets in Hatch have cases with OLA(2.4E-5) *OPHETBISO1(4.7E-3) *OPHEEPA(5.9E-3) or 7E-10 probability of combined HEPs in the cutset.

Another operator action cutset not addressed for dependency is the following:

• All cont heat removal fail to initiate	2E-5
• Vent (conditional)	0.1
• QRA (recovery)	0.1
TOTAL	2E-7

Justification for such events would generally be desirable. An increased "floor" value for the combination of basic events is also a feasible alternative.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Review HEP combinations to ensure dependencies are accurately reflected.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element HR**	**Subelement 28, 30**

EPU

The extended Power Uprate HRA evaluation indicates that it may be prudent to have an alternative HRA methodology that is both more in line with current HRA techniques and one that can be updated more easily than the FLIM method.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider use of the EPRI Cause Based methodology tied to a Time Reliability Correlation for the time stressor evaluation.

PLANT RESPONSE OR RESOLUTION

INITIATING EVENT (IE)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 1**
The initiating events notebook describes the process used for manipulating the data but does not provide criteria for categorizing plant specific events. Judgment is used to categorize events such as a manual scram prior to an automatic scram.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Provide criteria for consistent interpretation of plant data for initiating event analysis.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 2**

Although there are no standard "Industry Practices" there are some plants which have developed effective guidance documents which provide an important tool in training new employees and in providing continuity in performing updates of the PSA.

The existing PSA provides some instructive material for guidance of those performing updates of the PSA, but they are not sufficient for the uninitiated. Because of the normal turnover of personnel, guidance/instructional documents are an important resource for training and for providing uniformity in the quality of the PSA. Such documents are also helpful in briefing of management on the basics of PSA.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider developing Guidance Documents as part of any future updates of the PSA.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 3**

The Initiating Event Database states that the 09/30/92 scram was categorized as %SCRAM. The scram was manual at 12 mil displacement of the automatic turbine trip which should have tripped at about 9 mils. This should have been categorized as %TTRIP.

The Hatch units appear to have a high number of turbine trip related scrams. Careful consideration of each turbine trip is needed for an accurate initiating event frequency.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Re-categorize this scram

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element IE**	**Subelement 3**
No guidance document is provided, but the general IE identification, grouping, and calculational processes are generally defined in the various IE calculations.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
An IE guidance document or a single IE document that covers the guidance and all IE assessments in the PSA would be an enhancement.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 4**
The initiator list in the latest work package lists various ATWS scenario transfers as initiating events (the document correctly notes that ATWS is not an initiator).		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Remove ATWS events from tabulation in the IE calculations as if they are initiating events.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 5**
There is no Loss of Instrument Air Initiating Event. It is subsumed in Loss of MSIVs and Loss of Vacuum. There are other significant impacts to the plant other than MSIVs going shut and losing vacuum such as minimum flow valves on Condensate/Booster/Feedwater pumps failing open and the impact on containment vent.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Add a Loss of Instrument Air Initiating Event to the model.		
PLANT RESPONSE OR RESOLUTION		
A special initiating event called &LOINSTAIR (FAILURE OF INSTRUMENT AIR 1 YEAR EXPOSURE SPECIAL INITIATOR) has been added to the Hatch PSA model.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 7**

The list of initiating events is quite comprehensive. However, the following initiators are not included/address:

- RPV Rupture (Excessive LOCA)
- Manual Shutdown
- Ref. Leg Break

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Include RPV Rupture and Manual Shutdown initiating events and the associated accident sequence development. Assess significance of Reference Leg Break Initiator and include in accident sequence development as appropriate.

PLANT RESPONSE OR RESOLUTION

A model, @RPVRUPTURE, has been added to the Hatch model to evaluate Excessive LOCA. Reference Leg Break is a small LOCA and is adequately evaluated in the small LOCA models. Manual Shutdown is a consequence of some of the initiating events modeled such as loss of vacuum. It is assumed in these events that the operating crew manually shuts the reactor down or the event itself scrams the reactor. The consequences of manual shutdown are included in the reactor scram data for the model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element IE**	**Subelement 7**
Excessive LOCA should be considered in the analysis or a basis for exclusion should be provided in the documentation		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Add Excessive LOCA to model and document, or document why it should not be added to model.		
PLANT RESPONSE OR RESOLUTION		
This model has been included: @RPVRUPTURE.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 9**

Loss of MCR Cooling

The Hatch PSA includes the loss of cooling to the (MCR) main control room as an initiator.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

This is an excellent approach and shows the thoroughness and level of detail included in the model.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element IE**	**Subelement 9**

Large LOCA

The large LOCA initiator is said to include the spurious ADS event. Nevertheless, the frequency of the IE appears to be quite low.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Review the basis for the spurious ADS event frequency to determine if it is reasonable and a sound technical basis is provided.

PLANT RESPONSE OR RESOLUTION

The reference for the frequency of the spurious electrical actuation of all SRVs is Hatch IPE notebook, H61.5.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 11**
A closer look should be performed for ATWS initiating events. The small ATWS contribution probability is probably attributed to the choice of contributors (i.e., highest frequency, but also highest defense in depth). It is not clear that subsuming shows appropriate contribution.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Perform sensitivity analyses and incorporate appropriate contributors to ATWS.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element IE**	**Subelement 11**
The documentation should provide the contribution for each flooding scenario. It is not apparent how the remaining flooding events were subsumed or screened (i.e., initiating frequency or CDF contribution). The internal flood information is too general.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Perform sensitivity study for flooding initiators and determine if more initiators need to be added to the model and then document the results.		
PLANT RESPONSE OR RESOLUTION		

Hatch IPE notebooks which still pertain for the flooding evaluation are exhaustive in information regarding screening and combining initiating events. These notebooks are H95, H96, and H97. The final results show that internal flooding is insignificant in contribution to core damage. The contributions for the flooding scenarios modeled can be retrieved via CAFTA modeling tools. The flooding information is more than adequate to explain the significance of the screened initiators. These previously mentioned notebooks, H95, H96, and H97, serve as resolution to this particular comment.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element IE**	**Subelement 14**

The Hatch ISLOCA evaluation provides a through discussion of the events that could lead to the ISLOCA and a realistic quantification of the frequency of such events including the failure of pipes due to overpressure.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

N/A
PLANT RESPONSE OR RESOLUTION

LEVEL 2 ANALYSIS (L2)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 1, 3**
Guidance The Level 2 documentation is not considered to be sufficient to support updating and reproducing the analysis without significant input from the current PSA analysts.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Provide additional guidance or documentation to support the Level 2 evaluation process.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 4**

Success Criteria

The success criteria currently inferred from Level 2 are:

- CsI < 10%
- Release Time < 6 hrs. after RPV breach
- MAAP evaluation

It would be preferable to provide a success criteria for each functional node in the Level 2.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 4, 5**
The Level 2 success criteria (e.g., systems & flow rate of in-vessel post-core damage recovery and ex-vessel debris cooling, methods for post-core damage RPV depressurization, etc.) are not as clearly discussed as they are in the Level 1.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Discuss the Level 2 success criteria in the LERF model documentation.		
PLANT RESPONSE OR RESOLUTION		

This will be addressed in Revision 2 of the Hatch PSA. Water systems for post core damage are essentially the same as pre-core damage. They may be less in number or inject in unique places but when the vessel goes to core damage due to lack of water coverage all sources have been exhausted prior to.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 5**
The analysis does not acknowledge the extensive NRC studies that support quantification of severe accident phenomena.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include consideration of the NRC studies.		
PLANT RESPONSE OR RESOLUTION		
Severe accident phenomena and phenomenological studies for the Hatch model have been extensively addressed in documentation provided by Fauske and Associates (who were involved in many of these extensive, but rather limited result studies).		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 5**

Loss of Makeup Sequences

The LERF determination for loss of makeup sequences include the following:

- Must fail DW sprays
- Must fail venting
- Must fail RPV depressurization

The latter two are believed not necessary to fail to have the possibility of releasing a large magnitude of radionuclides, i.e.,

- Despite successful vent, shell failure will occur at elevated cont. pressure. The release is expected to be large.
- Despite RPV depressurization, shell melt-through will occur and the release will depend on the reactor building effectiveness in release mitigation.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Consider removing the non-minimal functions from the Level 2 LERF assessment.

PLANT RESPONSE OR RESOLUTION

This comment will be considered in Revision 2 of the Hatch model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 7**
Missing Transfers ATWS_3 appears to be missing from the LERF evaluation.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Reassess this sequence and determine if there are other accident sequences that may also not be transferred to the LERF model.		
PLANT RESPONSE OR RESOLUTION		
This will be addressed in Revision 2 of the Hatch model.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 7**

Sequences Transferred

GT_21 and GT_37 are core damage sequences from Level 1 that have many of the characteristics needed to be addressed in LER_OT fault tree. However, it is not included in the LER_OT top. This is believed to be an oversight and could be an example of other sequences that have not been transferred correctly.

It is noted that LER_OT includes in its written definition that depressurization has failed. However, this is not true. GT_9 is included in LER_OT and it has not had depressurization failed.

GT_21 and GT_37 have SORVs, but this should not affect the treatment in Level 2 if VDPR is performed correctly.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Verify sequences are transferred correctly to Level 2.

PLANT RESPONSE OR RESOLUTION

This will be addressed in Revision 2 of the Hatch PSA model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 7**
One of the more dominant ATWS core damage sequences, ATWS-3, is not transferred into the Level 2 LERF model.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Verify that it is appropriate to exclude sequence ATWS-3 from the LERF model. If not, include this sequence transfer into the LERF model.		
PLANT RESPONSE OR RESOLUTION		
This comment has been previously addressed.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element L2**	**Subelement 7, 8**

A realistic treatment of DW shell failure consistent with the approval taken by Theofanous may require additional realism in the treatment of shell failure mitigation, specifically the treatment of DW sprays.

LER_VD appears to be overly conservative in the treatment of:

- GT_9
- LOSP_4

In that DW sprays are not asked and they could result in reducing release below Large.

It is noted that LO and QT have failed, but these do not guarantee DW spray failure.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Eliminate over conservatisms.

PLANT RESPONSE OR RESOLUTION

This will be addressed in Revision 2 to the Hatch PSA model. When #LO is failed in the Hatch model, there are no pumps to run drywell spray. I think this comment provides some overconservatism on the part of the one making the comment.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 8**

Shell Failure

Liner melt-through failure mode is discussed in IPE Sections 4.4.9 and 4.7.5.3.A. It is also discussed in FAI/98-95. The result is that FAI position paper asserts that there is little impact due to debris shell interaction.

The shell failure evaluation by FAI appears to be contrary to accepted technical analysis relative to the shell integrity under degraded core conditions. The evaluation does not address any of the following items:

- The release of substantial quantities of debris in excess of ½ of the core debris.
- The voiding of the debris due to CCI products in the debris causing higher "volumes" on the drywell floor and in the drywell sumps.
- The potential for debris spreading in a directed location instead of evenly distributed.
- The potential for a large shell failure size.
- The modification to include the EPU core (more debris, higher decay heat).

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Shell failure needs to be reevaluated to consider the major body of technical work. This includes the NRC research NUREG/CR-5423 (Theofanous) and NUREG/CR-5623 (G.A. Greene) and other BWR PRAs. Reassess the shell melt-through failure mode to be consistent with NRC and industry assessments unless there are unique Hatch features that preclude shell melt-through considerations leading to LERF potential.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 8**
PLANT RESPONSE OR RESOLUTION		
This will be reassessed but since Fauske was involved with this "major" work, it is doubtful that this will change. If the reviewer will read the entire works they would see that the general conscensus was vague on drawing any conclusion from these writings.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	Element L2	Subelement 8, 10

Phenomena

The phenomena that are not addressed quantitatively in the Hatch Level 2 include the following:

In-Vessel Interactions

- H_2 Production
- Steam Explosion
- Recriticality
- Bottom Head Failure

RPV Breach by Debris

- Direct Containment Heating (DCH)
- RPV Blowdown and Containment Pressurization
- Debris Temperature and Containment Susceptibility

Ex-Vessel Interactions

- RPV Blowdown
- Ex-Vessel Steam Explosion
- Core Concrete Interaction
- Drywell Shell Failure
- Hydrogen Burn
- Containment Heat Removal
- Vapor Suppression Failure

It is noted that the failure of the ring header due to contact with cryogenic fluid has occurred at Hatch. Therefore, an explicit discussion and quantification of this would be desirable to include in the Level 2 (and Level 1).

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 8**
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Ensure that the phenomena that may contribute to determining the importance of accident mitigation SSCs are quantitatively addressed. Include phenomena or provide a means to address in applications such as applications affecting de-inerting time, H_2 Analyzer tech specs, SRVs		
PLANT RESPONSE OR RESOLUTION		
This occurred due to lack of operator knowledge regarding the characteristics of liquid nitrogen. The plant now has a vaporizing system for large nitrogen flow such as drywell inerting and the flows are limited physically and by procedure. This problem has been addressed.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 8**

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element L2**	**Subelement 8**
Discussion of recriticality impact on Level 2 analysis for non-ATWS conditions does not appear to be included. The attached discussion based on NRC efforts in NUREG/CR-5653 could be added to the Level 2 documentation.		
LEVEL OF SIGNIFICANCE		
C -- Quantitative treatment is not significant therefore, it will not have an impact on results.		
POSSIBLE RESOLUTION		
Add the attached documentation to Level 2.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	Element L2	Subelement 8

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element L2**	**Subelement 8, 10**

Energetic Failure Modes

The "Position Papers" provide valuable input to the PRA. However, they appear to have been used to the exclusion of research performed by the NRC that indicates that there are probabilities, however small, that can cause energetic drywell failures. These small failure probabilities include:

- In-vessel steam explosion NUREG-1524 and NUREG/CR-5030
- Ex-vessel steam explosion.
- DCH.
- Hydrogen deflagration when de-inerted.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Incorporation of LERF potential from these failure modes is judged important to include in the Level 2. All accident sequences could be subject to energetic failure modes of this type, i.e., including the overpressure failure modes that currently assume Bellows-only failures.

PLANT RESPONSE OR RESOLUTION

This will be reviewed for Revision 2 of the Hatch PSA model. However, the Fauske papers provided for the Hatch model are complete on the subject.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 6)	**Element L2**	**Subelement 8**
Position Papers The Level 2 qualitative evaluation is supported by an extensive set of "Position Papers" that describe phenomena and provide deterministic calculations to support the assessment of phenomena such as: • Steam explosions • DCH • Shell melt-through		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 11**
LERF Cutsets Nos. 1 and Nos. 3-7 The LOSP cutsets do not take additional credit for AC power recovery during the Level 2 time frame. There could be approximately 1 hour after core damage and before RPV failure to credit additional AC power recovery.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Verify that no additional AC recovery can be applied for the Level 2 time frame to ensure results are not overly conservative.		
PLANT RESPONSE OR RESOLUTION		

The same recovery approach is applied for the CDF and LERF models. There is no additional power recovery time for Level 2.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 11**

RB

The reactor building node is crucial for the accurate assessment of the shell melt-through. It should be included explicitly in the probabilistic CET model to reflect:

- Bypass of the RB
- H_2 combustion effects
- Uncertainty in MAAP treatment of the fission product retention mechanisms given the R.B. flow paths.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Model the R.B. using a deterministic code such as MAAP to assess its mitigation capability for dominant containment failure modes.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 11**

The LERF model does not explicitly consider:

- SORVs caused by adverse environment due to post-core damage extremely high gas temperatures.
- ADS/SRV failure due to post-core damage high drywell temperature.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider including the above issues in the LERF model, and explicitly consider the issue of post-core damage RPV depressurization--both, positive and negative impacts of the additional time to perform the depressurization, but also the higher environmental stress on the SRVs and the SRV solenoids.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 12**
The LERF model and documentation do not clearly discuss Level 2 actions. Operator actions propagating through the Level 2 appear to be the same action HEPs calculated for pre-core damage conditions.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Document and assess Level 2 HEPs to consider the effects of the post-core damage context.		
PLANT RESPONSE OR RESOLUTION		

In general this is correct. Level 2 is a progression of Level 1. There is no break in between; all actions have been done initially to prevent the situation of Level 2. There are very few extra items that can be done that were either not done or failed to be done and resulted in the situation that will become Level 2. Recoveries are a possibility but are difficult to evaluate for Level 2. This comment will be further explored in Revision 2 of the Hatch model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 12**

Level 2

DW Sprays

The implementation of the DW sprays in Levels 1 and 2 does not address the DW Spray Initiation Limit Curve (DWSIL). There are no MAAP calculations that are used in conjunction with the Level 2 assessment to determine whether the DW sprays would be allowed. The ability to use DW sprays is contingent on meeting DWSIL and having a cue to initiate the sprays. Neither of these two items from the EOPs/SAGs are discussed or evaluated in the analysis.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Address the DW spray initiation under various severe accident conditions to mitigate against DW shell failure.

PLANT RESPONSE OR RESOLUTION

This will be done for Revision 2 to the Hatch model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 12**

<u>Vent</u>

The containment treatment in Level 2 calls the same node as needed in Level 1. The assumptions apparently used in Level 2 are:

- Operator action is not adversely impacted by potential radiation release.
- The vent will completely depressurize the containment such that a low differential pressure across the DW shell exists.

Both assumptions appear questionable:

- Operating crew response with the TSC manned is expected to be more cautious and may delay or prevent the vent.
- There is no procedural guidance to depressurize the containment.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Update the vent treatment in the Level 2 to address these two issues.

PLANT RESPONSE OR RESOLUTION

The Revision 2 Hatch PSA model will evaluate this, but-as previously stated-, actions are fairly well exhausted in the Level I portion of the model in order to prevent getting to Level II. Venting containment is done at pressures that are prior to containment failure. Operator actions to vent with the "hardened vent" which is in question here, are the same. It is an emergency action only and stress levels are high already. There is time however for preparation to vent so the overall action is not high in failure probability. Procedural guidance for venting was provided to the certification team if and when they asked for it. This comment has very little merit in that too many assumptions of failure are made with little knowledge on the subject. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element L2**	**Subelement 12, 24**

The following Level 2 "recovery actions" do not appear to be explicitly addressed:

- AC recovery
- RPV depressurization prior to vessel failure
- In-vessel Injection recovery

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Address the above recoveries explicitly in the LERF models.

PLANT RESPONSE OR RESOLUTION

AC Recovery is addressed in the Level I model section. This is carried through for the Level 2 conditions. RPV depressurization makes up the one of the largest contributions to the model and is certainly addressed prior to vessel failure.
HATCH does not consider In-Vessel Injection Recovery. This is strictly an engineering judgement and is not considered in keeping with what the new ASME standard on PSA quality wishes to invoke.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 13**

Containment and System Functional Failures

Some systems have been treated ultra conservatively by assuming the systems are completely ineffective in the severe accident core melt progression:

- RHRSW cross tie for containment flooding and in-vessel recovery.
- LPCI/CS injection to the RPV following RPV depressurization due to RPV breach
- No FPS cross tie is included for RPV makeup.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

It would be prudent to consider all "effective" system capability in the PRA to avoid a biased risk spectrum that could distort SSC importances.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 13, 24**
Alternate injection sources (e.g. RHRSW and DFP crosstie) are conservatively not credited in the model. These alignments may be credited for in-vessel core melt arrest and for containment flooding.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include the above alternate injection alignments explicitly in the LERF model.		
PLANT RESPONSE OR RESOLUTION		
These systems are included in the Revision 2 PSA model and are not worth very much in providing a reduction in core damage.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 13, 24**

It is not clear as to whether the LERF model credits LP ECCS injection after RPV melt-through during high pressure core damage scenarios. The LP ECCS systems are most likely available and will flood into the RPV upon RPV melt-through and provide debris cooling and prevention of drywell shell melt-through.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Verify that the LERF model credits LP ECCS injection following RPV melt-through during high pressure core damage scenarios. If not, explicitly include in the LERF model.

PLANT RESPONSE OR RESOLUTION

These systems are credited but there is not a specific model node to address late injection. This comment will be considered for the Revision 2 model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 15**

Class 4 Containment Failure

The definition of containment failure during an ATWS and its size and location should be identified. The attached discussion of ATWS-induced dynamic loads is included for your use in considering the plant specific evaluation. Attachment L2-19 provides some consideration regarding containment failure modes that may require consideration under ATWS conditions.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

The containment failure mode for failure to scram events is key to LERF assessment and should be assigned consistent with the Southern Nuclear evaluation of ATWS. Based on Mark I hydrodynamic loads associated with high pool levels, it is suggested to modify the containment failure locations and probabilities to be 0.5 in the wetwell air space and 0.5 in the wetwell water space.

PLANT RESPONSE OR RESOLUTION

Having read the reports regarding the containment of the Mark I vintage failure locations are vague to say the least. There are many reports but they do not address failure with regards to the characteristics of the PRA. This comment will be addressed, if possible, in Revision 2 of the PRA.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 18**
The LERF model apparently considers all containment failures to be ruptures (except the shell melt-through); that is, the model does not explicitly question whether the containment failure is a leak or a rupture.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
When time and resources permit, consider whether the primary containment failure is a leak or a rupture failure mode.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 19**

The Level 2 LERF model does not quantitatively address the following primary containment failure modes and locations:

- DW shell melt-through
- DW head seal failure
- WW water space failure
- Dynamic torus loading and failure during unmitigated ATWS

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Address the above containment failure modes and locations in the LERF model.
PLANT RESPONSE OR RESOLUTION
These items will be considered in the Revision 2 model, however, their quantitative evaluation may be questionable due to limited insight on the subject failures.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 21, 22**

LERF

The "Level II Process," FAI/98-88, implies that the PSA Application Guide for definition of LERF is used. The LERF is defined as a large early release frequency and is the figure of merit recommended for measuring Level II activities. The PSA Applications Guide (EPRI 1995) defines the LERF as both a large and early release. Large is defined as a rapid, unscrubbed release of airborne aerosols and early as occurring prior to effective implementation of off site emergency response. The end states representing an unscrubbed release are OPD, over pressure failure of the drywell, CB, a containment bypass, VD drywell venting and OT, over temperature. The time between vessel and containment failure, less than 6 hours would be considered early. Thus, all CB and VD end states would be LERFs and OPD cases where there is RHR injection and OT cases where there were no drywell sprays or vessel depressurization. There, however, is not a description of how the Emergency Action Levels (EALs) for Hatch are used to distinguish the LERF end states.

Specifically, long term loss of DHR sequences are identified as LERF potential, i.e., LER_OPD. This interpretation would appear to be overly conservative and should be reevaluated. (See EAL Procedure 73EP-EIP-001-0S, p. 42 of 46 (attached).

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Ensure consistency of LERF definition with implementation using the Hatch EALs.

PLANT RESPONSE OR RESOLUTION

This is in absolute keeping with the emergency action guidelines. This specific topic was discussed with this reviewer early in the morning of one of the review days. The Hatch facilities are manned within the hour after accident declaration. State and Local Authorities are notified of any predicted release and recommendations regarding same shortly thereafter. This does not mean that the release will not be early. The action recommendation to shelter, evacuate, or whatever is a state function. Plant Hatch does not control this. If the first recommendation that Hatch provided was shelter, and the state and local authorities implemented this, it would take far longer than one hour to accomplish.

Because the Hatch function is addressed in one hour does not make the release not early. The time frame of 6 hours was discussed with Southern Nuclear Emergency Planning personnel and it was agreed that this can be considered early.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 24**

CET

The CET structure is quite unusual:

- Usually in-vessel recovery is treated in the CET. The FAI event tree has replaced this with a question regarding previous injection. This node does not appear useful.
- Containment isolation failures and energetic containment failures (both LERF contribution) have been eliminated from the LERF CET. This seems quite unusual.
- DW spray injection to cool debris and prevent shell failure does not appear to be addressed.
- The shell melt-through, the dominant failure mode of concern in a steel Mark I containment, is not addressed in the CET.
- Energetic failure modes such as cited in L2-8 are not quantified.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Modify the Level 2 CET to explicitly quantify potential dominant contributors to LERF.

PLANT RESPONSE OR RESOLUTION

This will be addressed in the Revision 2 Hatch PSA model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element L2**	**Subelement 24**

Containment Isolation

This is not in the L2 model. For GL 88-20 evaluations of vulnerabilities, this could be acceptable.

For applications, it is desirable to include containment isolation assessments.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Include CI into the L2 model or be aware that applications affected by this function may need to be treated explicitly with compensatory measures.

PLANT RESPONSE OR RESOLUTION

Containment Isolation is included in the Level 2 model. Containment isolation of 2 inch and under piping is not considered however, because of limited contribution.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element L2**	**Subelement 24**

CET

The Level 2 CET structure does not adequately address the use of drywell sprays or containment flooding to preclude shell melt-through.

The drywell sprays are critical to the assessment of preventing shell failure in a number of accident sequences. The CET is not structured to ask DW sprays under certain CET sequences. This means that assumed OK sequences (CET_CN, CET_VW) are really shell failure cases unless DW sprays are asked and shown to be successful.

Because of the concern expressed relative to the shell failure treatment (see L2-8), the assertion that any shell failure impacts are minimal should be revisited.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Restructure the CET to properly account for potential shell failure cases.

PLANT RESPONSE OR RESOLUTION

This will be addressed in Revision 2 of the PSA model.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element L2**	**Subelement 24**
Containment flooding is not explicitly considered in the LERF model.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include containment flooding in the LERF model.		
PLANT RESPONSE OR RESOLUTION		
Late injection is considered now, however, there is not a specific node for it in the Level II model. This comment will be addressed.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element L2**	**Subelement 28**
The Level 2 analysis is described both in the IPE and the LERF work package. There are substantial technical bases provided in the IPE that are not reiterated in the LERF model documentation, and it is not clear as to the treatment of these in the LERF model.		
LEVEL OF SIGNIFICANCE		
D		
POSSIBLE RESOLUTION		
Include the critical technical bases and assessments of the IPE Level 2 into the current LERF documentation.		
PLANT RESPONSE OR RESOLUTION		

PRA MAINTENANCE AND UPDATE (MU) FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 3**

There is sufficient detail in the procedures to reproduce the evaluation, however the various procedures are not linked together.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Revise the procedure for Maintenance and Update to direct the user to all the procedures that will need to be used.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 4**
The Maintenance and Update procedures REES 2-2 and REES 2-4 do not direct that the following areas be evaluated as part of data collection for an Update: 1. New or revised Engineering Calculations. 2. Changes in the Severe Accident Guides (SAGs). 3. Changes in the E-Plan. 4. Maintenance Rule Unavailability Database. 5. Industry Operating Experience other than NRC information.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Evaluate the above listed areas as part of the next PRA Update.		
PLANT RESPONSE OR RESOLUTION		

The PSA department procedures address items 1, 2 and 4. Changes in Severe Accident Guidelines are very rarely PSA related nor are changes in the Emergency Plan. The PSA personnel do review these items (all items affecting PSA) periodically to ensure anything affecting PSA is credited.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element MU**	**Subelement 4**
Equipment unavailability data and SAG information was not incorporated into the latest revision of the model.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Update the model with plant specific equipment data and SAG operator actions and modeling.		
PLANT RESPONSE OR RESOLUTION		
Hatch has a data update for the Revision 2 model. SAG actions are reviewed. There are none necessary for the PSA that are not already included in the model.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element MU**	**Subelement 4**
Neither the REES procedures nor the PSA Data Update Guidelines require a review of Engineering Department calculations for consideration in model updates -- nor do they require reviewing the plant emergency plan.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
These items should be added to the maintenance and update procedures.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 5**
Plant specific initiating event data was used in updating the hatch model; however, the equipment data was not updated. Equipment data may reflect better maintenance practices used in the industry.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Update the plant specific equipment data.		
PLANT RESPONSE OR RESOLUTION		
Revision 2 of the Hatch model has a data update.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 6**
Computer code base model security maintained as discussed in the REES procedures is a strength.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element MU**	**Subelement 6**
Procedure 4-2 seems to address computer code controls including acceptance testing, but the procedures for performing applications do not appear to address benchmarking of the code prior to use in applications.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 7**
Procedure REES 2-2 does not refer to procedure TS 1-5 which controls computers code development and control.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Revise REES 2-2 to reference TS 1-5 for control of the PRA Computer Model.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 8**
There is a process defined in the REES procedures for maintaining the PRA. As the latest update of the model was reviewed, it became clear that the procedures allowed for updates without important elements of an update, i.e., equipment data and SAG modeling.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
The procedure should be revised to require certain elements for an update or provide criteria for not updating if not needed.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element MU**	**Subelement 8**

Procedures are in place for maintenance and distribution to the plant of each of the main products such as MR risk significance, EOOS. Engineering products do not appear to be reviewed prior to implementation.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Ensure engineering modification procedures include PSA screening, and consider PSA as an impacted group prior to implementation.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 9**
A schedule for update has been defined, but it allows for much interpretation. After each outage and within 6 months a decision is made regarding whether to conduct an update. At least every 3 years an update needs to be considered. What is missing is objective criteria for management to make the decision to update or not the PRA.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Revise the REES procedure and provide objective criteria for making a decision relative to updating the PRA.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 10**
The REES procedures do not require review or evaluation of a PRA update outside of REES personnel. It is considered good industry practice to have model update results reviewed by a panel composed of a broad range of backgrounds.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Have the Revision 1 model results reviewed by an expert panel.		
PLANT RESPONSE OR RESOLUTION		
The review of the model is defined by the procedures for calculations used at SNC. Expert panel is not used.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element MU**	**Subelement 10**

Procedural guidance is provided in REES 2-4, and qualifications are defined in REES 2-7, but no guidance is available for addressing the threshold for screening risk significance, or the cumulative impact of non-risk significant modifications.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Define a screening criteria and evaluate periodically the cumulative impact of multiple non-risk significant modifications.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 11, 12**

There is no provision in the REES procedures for a qualitative review of past applications based on a revision to the PRA model. There is no evidence that applications have been evaluated based on revision 1 of the model.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

There should be a formal review of any applications in the licensing process with a model revision.

PLANT RESPONSE OR RESOLUTION

Licensing applications are reviewed with respect to every PSA model revision.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 14**
It is clear that there is an independent review of REES generated documents. This is considered a good practice in the industry.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element MU**	**Subelement 15**
The current process for maintaining records may not allow for future configuration control needs. There appears to be limited traceability related to some model input reviews/considerations. For example, there is no traceable evidence that industry events were evaluated, and that their disposition relative to Hatch in a format similar to initiating event data was evaluated.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
REES procedures should consider incorporating requirement to track model input evaluations.		
PLANT RESPONSE OR RESOLUTION		

QUANTIFICATION (QU)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 1**

The guidance used for quantification "SNC-HI-98-005" describes the method for solving the previous version of Level 1 model. It needs to be updated to describe the current single top version using a fault tree recovery file.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Update documentation to reflect current process.

PLANT RESPONSE OR RESOLUTION

The quantification process is described in the individual model calculations as it will be for Revision 2 of the Hatch model. This hardly warrants a level B comment. This will be addressed as revisions occur.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 3**
Documentation of steps required to reproduce the CD cutsets is very good. All required files are documented as to their content.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element QU**	**Subelement 3**
The manner in which recoveries are applied are not easily discernable from the description in the quantification notebook. For example, in Cutset #4, describe the basis and the methodology for appending basic event GRA2&3 (prob=0.27).		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Please provide additional description for the methodology incorporated for the Recovery file.		
PLANT RESPONSE OR RESOLUTION		

The Recovery Tree is what was used with the Rev 1 model that was reviewed by the Peer Certification Team. This was explained and it was also explained that these details resided in SNC Calculations which serve as the official QA record. This calculation number is PSA-H-00-024 Rev.1. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 4**
Propagation of NOT Logic The CAFTA Code requires some manual intervention to ensure the proper NOT logic is included in the accident sequences. This is particularly important for the Hatch Event Trees which are multi-page event trees. The NOT logic is not automatically created for the initial page(s) of the Event Tree.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Create the proper NOT logic for accident sequences that transfer among multiple pages. For PRAQuant use only.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element QU**	**Subelement 4**
The mutually exclusive file is very comprehensive. The mutually exclusive file inappropriately removed a valid cutset at 2.535E-7. Discussions revealed this was known and corrected in current model.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Review Mutually Exclusive File with plant personnel to verify its validity. Update list accordingly and rerun model to verify reasonable results.		
PLANT RESPONSE OR RESOLUTION		

The Mutually Exclusive File is reviewed for Revision 2 of the Hatch model. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 6**
The fact that the practice at Hatch is to re-quantify the full PRA model for applications is a positive feature of the Hatch PRA program.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 8**
Cutset # 27		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
SBO (LOSP * CCF of DGs to start) appears to be ANDed with non-recovery convolved over 24 hrs. How can this condition survive beyond 2.5 hrs when RCIC and battery fail? This can be compared with cutset # 4 which includes identical recoveries when the failure is a failure to run (FTR) of the D/Gs. The AC recovery should <u>not</u> be the same for these two cutsets.		
PLANT RESPONSE OR RESOLUTION		

Grid Non-Recovery Probabilities integrate the time frame from Time=0 to Time=24 hours with RCIC available and with RCIC not available. This means that the whole spectrum of time which includes T=0 or fail to start to T=24 which includes fail to run that may have occurred at various times for various diesels over the 24 hours is covered. This manual integration is an acceptable approach to the more stiff arithmetic method of attempting to arrange a value for fail to start separately from fail to run. It is doubtful that very much value is lost in the integration. In addition a start failure recovery is included based on plant data and work experience which prevents a significant contribution to this concern in the cutsets. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element QU**	**Subelement 8**

CDF quant cutsets Nos. 1, 2 & 3

There appear to be two possible conservatisms in the model that may bias the results. The following two items are identified for consideration:

- Confirm that all FW is failed due to loss of DC switchgear S016.
- Verify HEP for fail to depressurize. This HEP appears to be approximately a factor of 10 higher than similar BWRs.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Consider removing conservatisms in the Hatch model.

PLANT RESPONSE OR RESOLUTION

All Depressurization values have been reevaluated for the Revision 2 model using HRA Calculator. The vendor performing the work, SCIENTECH, adheres to the ASME standard guidance for their calculations.

The loss of S016 prevents the station service buses from transferring to their alternate power sources which are the startup transformers. This will cause a loss of vacuum in the main condenser, MSIV closure as a result, a loss of condensate and condensate booster pumps which in turn will prevent restart or operation of the reactor feed pumps.

This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element QU**	**Subelement 8**

Bypass the Level 1 MSIV Closure Interlock

The model evaluation includes the possibility under ATWS conditions that the TBVs and MSIVs will be open for events such as:

Event	Prob
TT	7E-3
Loss of FW	4.2E-2
Loss of Condenser Vacuum	<1.0

These estimates are substantially lower than other BWR PSA assessments.

The assessment of these conditional probabilities is necessary to ensure that the model quantitatively reflects the plant and operating crew response.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reassess the conditional probabilities.

PLANT RESPONSE OR RESOLUTION

This comment has been previously addressed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 9**
The methodology used to address common cause is comprehensive, but some potentially significant contributors such as common cause failure of HPCI/RCIC and ECCS strainers do not appear to be included.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Perform sensitivity and provide basis for exclusion or include in the model.		
PLANT RESPONSE OR RESOLUTION		

Common cause failures of ECCS strainers and HPCI and RCIC are now included in the model. The basic events for HPCI and RCIC are, HPCCR for Common Cause Failure to Run and HPCCS for Common Cause failure to start. For ECCS the basic event for Common Cause failure of the strainers is under the AND gate STRNCCFAILURE. The strainer concern is only postulated to be a problem during the Large LOCA condition therefore Common Cause failure is ANDed with the Large LOCA initiator.
This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element QU**	**Subelement 9**
Common Cause Data Battery common cause failure is not included into model.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Consider the incorporation of common cause battery failures.		
PLANT RESPONSE OR RESOLUTION		

Batteries are passive boxes of fluid and lead. The continuous maintenance program at Hatch for these units prevent anything that could lead to a disasterous or total failure common cause such as: all batteries having a massive jug crack at one time. The only common cause items faced at Hatch to date are terminal corrosion and minor jug cracks. These issues are closely monitored and cells will be replaced prior to any potential of their failing to be able to perform. Outage discharge tests provide a high degree of confidence—along with very conservative load calculations—that a common cause failure will not prevent the batteries from performing as necessary during an accident. Battery common cause failure is therefore, not included.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element QU**	**Subelement 9**
HPCI FTS/FTR events have a F-V of ~0.27. RCIC FTS/FTR events have a F-V of ~0.13. This appears extremely high compared to other BWRs.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Review modeling of HPCI/RCIC and their assigned FTS/FTR probabilities to ensure that there are not overly conservative values assigned, use recent data to characterize equipment performance.		
PLANT RESPONSE OR RESOLUTION		

Hatch data was recently updated. All failure data from 10 years back have been Bayesianed into the present failure data. This includes early years of poor service prior to some modifications recommended by GE that seemed to help. The modifications are such that it is not believed that the old failure data can be totally omitted.
This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element QU**	**Subelement 9**
Given that HPCI and RCIC are so important, identify the CCF of HPCI/RCIC as a dominant contributor to CDF and include as a separate common cause group.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Review CCF of HPCI/RCIC modeling and the assigned probability.		
PLANT RESPONSE OR RESOLUTION		

A common cause value has been added to the HPCI and RCIC models. It is difficult to group these machines because their main similarity is in the fact they are made by TERRY Corp. The control schemes for start are different and the physical difference in pumping power reguires more components on HPCI that RCIC. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 10, 17**

Operating Crew Actions and Dependence Evaluation

There are three issues that can be discussed relative to the treatment of dependent HEPs:

- Are the HEPs in the same cutset searched for and identified
  - For dynamic actions and certain recoveries, but does not include certain other actions (e.g., QRA, MCC)
- Are HEPs evaluated
  - Yes, with the above exception
- Is a "floor" on the lowest HEP or HEP combinations addressed:
  - No: This is inconsistent with the draft ASME PRA Standard

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Evaluate the HEP dependencies.

PLANT RESPONSE OR RESOLUTION

The recent update of the Hatch HRA by SCIENTECH using HRA Calculator addresses these issues. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 11**
ATWS contribution appears to be low. Comments in the accident sequence section discuss some potential non-conservative sequences involving the selection of initiators for ATWS, and LPCI and CS availability in scenarios with torus heat up (i.e. SORV).		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Re-evaluate the success criteria used for the specific issues discussed in the accident section and/or provide a comparison with similar BWRs.		
PLANT RESPONSE OR RESOLUTION		

The ATWS Event Tree has been totally redone. Success criteria for suppression pool cooling as well as the need for injection of standby liquid (boron) have been addressed. BWR contributions for ATWS range from low to as much as 60%. A principle cause for ATWS, mechanical failure of control rods to insert, has been reevaluated by General Electric and is in the E-6 range. This has significantly reduced the overall contribution from ATWS. A sensitivity study will be done on the Hatch model to address the selection of ATWS initiators. This comment does not affect MSPI.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	Element QU	Subelement 11

The SBO event (failure of offsite power and the failure of D/Gs to start) would appear to have an AC nonrecovery probability limited by the RCIC operability time plus the time for boiloff. This time is approximately:

- 2.5 hours (battery life) plus 2 hours according to EQE
- Other plants have found this time for boildown to be closer to 1 hour.
- No calculation for the EPU plant was presented to the Peer Review Team to support the 2 hour boildown time.

AC recovery from "typical" industry data at these times are as follows:

Time	AC Nonrecovery Prob.
3.5 Hrs.	0.18
4.5 Hrs.	0.13

The AC non-recovery probability value used in the subject Hatch cutset is .057 composed of two separate non-recoveries, .21 and .27

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

The AC non-recovery probability for cutsets with a failure to start of the D/Gs should be a factor of 2 to 3 higher than currently being used in the model. This would change if the battery life is reassessed or HPCI credit is included in the model.

PLANT RESPONSE OR RESOLUTION

HPCI will not be given SBO credit due to ventilation problems. Hatch in the revision of the model used for peer review has no 0.057 non recovery value. Boil off time is approximately one hour. RCIC has for this case 2.5 hors of run time. If it is able to operate the whole time then 2.5 hours plus and extra hour for boiloff is used for the SBO case. In reality after 2.5 hours of RCIC ops. boiloff would be longer. The Hatch model has changed its battery availability to 5 hours thus the grid recoveries have been changed. Nevertheless this comment appears to bear a lack of understanding or a failure to completely read the Hatch documentation regarding grid non-recovery factors for Hatch. In addition "typical industry" non recovery factors is a mis nomer. Every plant does this item differently and just about every plant has a unique electrical configuration which makes commonality a improbable goal.
This comment has been adequately addressed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 3)	**Element QU**	**Subelement 11**

LERF Cutsets Nos. 1 and Nos. 3-7

The LOSP cutsets do not take additional credit for AC power recovery during the Level 2 time frame. There could be approximately 1 hour after core damage and before RPV failure to credit additional AC power recovery.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Verify that no additional AC recovery can be applied for the Level 2 time frame to ensure results are not overly conservative.

PLANT RESPONSE OR RESOLUTION

The Level II model is being redone and this comment will be addressed. This is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 13**

The asymmetric CDF contribution for 4160V Buses E, F, G initiators is not discussed in the accident quantification notebook. Similarly, the asymmetry between Buses C and Bus D should be discussed.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Discuss asymmetries in the calculation for results.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 14**
The designation of circular logic gates with the prefix "L-" and treatment of circular logic loop should be considered a strength for model maintenance.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 25**
Given that the CDF is quantified under "OR" gate @H1CDFTOP, how are the success paths accounted for as described in Section 3.0of the Quantification Notebook? Are "NOT" gates used in the CDF fault tree model?		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Explain how the success paths are accounted for in the CDF fault tree @H1CDFTOP.		
PLANT RESPONSE OR RESOLUTION		
NOT gates are used in the fault tree model. The success paths are addressed by forming the failure sequences with PRAQUANT. PRAQUANT provides the success paths as well for the initial calculations.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element QU**	**Subelement 25**
Was a comparison performed between the PRAQuant sequence quantification results and the single top Core Damage quantification results? If the success paths were not explicitly modeled for the single top Core Damage fault tree, did this lead to different cutset results?		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Perform a comparison of the merged sequence cutsets with the Core Damage quantification cutsets.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 26**
The flag files (Appendix B) are not included in the quantification notebook.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Ensure that the flag files are included in the controlled copy and their use explained.		
PLANT RESPONSE OR RESOLUTION		
The flag files is in the model calculational files. This is the controlled copy.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 29**

<u>Uncertainty</u>

The PRA is well constructed and robust.

The model has been examined and produces reasonable results to support applications.

A separate explicit evaluation of potential contributors to uncertainty has not been performed.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

It is considered important to provide a qualitative search for uncertainties in the model.

The nature of unique plant features that could substantially alter the results is considered an important insight. This could include the treatment of

- Return to power
- ATWS mitigation without Boron injection
- Containment failure location in DW not bellows (i.e., bellows much stronger than modeled).
- Battery life
- RHRSW X TIE capability

PLANT RESPONSE OR RESOLUTION

This will be done at a later date. This has no affect on MSPI.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 31**
Document does not describe the dominant sequences or display the contribution by accident type (chart) in the summary document. In addition, should consider including some sensitivity results in the quantification document.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		
This is included in the calculational files. These are the QA records. Sensitivity results will be addressed at a later date. This will not affect MSPI.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element QU**	**Subelement 34**
The common cause events should provide more descriptive detail than just the basic event IDs that are considered in the CCF event (e.g., basic event CC-VM-21____I in cutset #7). This would provide additional clarity to describe which events are considered.		
LEVEL OF SIGNIFICANCE		
D		
POSSIBLE RESOLUTION		
Editorial		
PLANT RESPONSE OR RESOLUTION		

STRUCTURAL ANALYSIS (ST)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element ST**	**Subelement 4**

RPV Capability Success Criteria

Documentation for PRA success criteria should be consolidated into a document with other critical success criteria:

- FSAR gives 1250 psig as design pressure
- PRA says 1375 psig in the success criteria
- EPU uses 1500 psig service Level C for ATWS success criteria

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Clarify documentation that is to support the PRA analysis.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 1)	**Element ST**	**Subelement 5**

Plant Specific Differences

The bellows are significantly different between Hatch Unit 1 and 2 (50% thicker for Unit 2). This difference is not discussed in the documentation and is judged to potentially make a significant potential difference in the assessed failure probability of the containment.

In addition, the Hatch Bottom of the Torus is not as thick (-16%) as the CB&I analyzed plant. This means that the torus water space failure probability could be significantly increased.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Consider the differences between Hatch (both Units) and the analyzed plant.

	Torus Shell Thickness (min.)		Bellows	
	Top	Bottom		Ply-Thickness
Peach Bottom 2 & 3	0.604"	0.675"	Out	2 - 0.08"
Monticello	0.633"	0.584"	Out	1 - 0.08"
Hatch 1	0.640"	0.599"	X	2 - 0.05"
Hatch 2	0.640"	0.607"	X	2 - 0.078"

PLANT RESPONSE OR RESOLUTION

This does not affect MSPI. This is LEVEL II model concern.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2	**Element ST**	**Subelement 5**
Vent Bellows and Plant Differences The technical basis for the assessment of the bellows as the weak point of the containment should be identified. Hatch has double bellows seals these have been evaluated at other Mark I plants to have higher pressure capability than assumed here. In addition, the differences between Unit 1 and 2 could be significant and should be addressed in the evaluation. It is noted that plant differences documentation does not appear to address the differences.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Include discussion of containment differences between Unit 1 and 2. Include technical basis for pressure strength assessment of Unit 1 and 2 bellows.		
PLANT RESPONSE OR RESOLUTION		
This is not an MSPI concern. This is a LEVEL II model issue.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element ST**	**Subelement 5**

Containment Failure Pressure

The basis for the Hatch containment failure curve has the following related items that could be reconsidered:

- The bellows failure mode differs significantly from the CB&I Mark I study and other evaluations where 2 ply bellows are predicted to have capabilities in the 128 to 200 psig range -- not the 84 psig assumed in Hatch.
- Torus shell failure modes appear to be completely neglected. Any torus shell failure above the equator crack cannot be assumed to remain above the torus equator once the crack initiates and starts to "run".
- Containment failure under dynamic loads associated with ATWS do not appear to be addressed. These conditions are likely to increase the torus failure probability (see L2-15)

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reconsider the bellows failure mode as a dominant contributor to the containment failure modes. The torus, the drywell closure, and hatches are prime candidates for consideration. Under ATWS conditions, the hydrodynamic loads on the torus need to be examined to assess the ATWS induced containment failure location.

PLANT RESPONSE OR RESOLUTION

This is not an MSPI concern. This is a LEVEL II model issue.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element ST**	**Subelement 8**

The Hatch Level 2 PRA conservatively does not credit the Reactor Building in reducing release magnitude, despite the fact that MAAP runs cited in the IPE and associated documentation show that the Hatch Reactor Building would reduce releases in certain primary containment failure scenarios to the next lower release category.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Include the Reactor Building in the Level 2 PRA, as appropriate, if it is reasonably and justifiably assessed to aid in the reduction of release magnitudes (as the IPE documentation states).

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element ST**	**Subelement 9**

ISLOCA

The Hatch ISLOCA evaluation provides a through discussion of the events that could lead to the ISLOCA and a realistic quantification of the frequency of such events including the realistic assessment of failure of pipes due to overpressure.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element ST**	**Subelement 10**
The level of effort and documentation regarding the internal flooding analysis is a positive feature of the Hatch PRA.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

SYSTEM ANALYSIS (SY)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 1**
No guidance or ground rules are available for nomenclature, or criteria for failure modes included in the system models		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Develop ground rules for system modeling and specify nomenclature.		
PLANT RESPONSE OR RESOLUTION		

The nomenclature for failure modes is documented in the data update for the rev.2 model. It is not necessary to have a pre-set naming convention. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 1**

Guidance/Ground Rules

There is wide variation in the System Notebooks. The excellent AC Power System Notebook should be the model for other notebooks. The Containment Vent notebook is an example of an area that should be updated to be consistent with the precedence set by the AC Power System Notebook.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Update System notebooks to have consistent format and level of detail.

PLANT RESPONSE OR RESOLUTION

Notebook updates at this time remains questionable. Time restraints and overall net worth are the deciding points. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 2**
The original system notebooks seem thorough and reviewed by the plant, but these have not been updated. The system notebook need discussion on operator actions, common cause failures, and flag settings. These should be spelled out explicitly in the notebook.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

These items are spelled out explicitly in the calculations done for the model revisions, HRA update, Data update, and so on. System notebook updates will come as time permits. This is hardly a B level finding. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 4**
The system notebooks, conversion notebooks, and computer model was available for team review.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 4**

<u>Fault Tree</u>

Excellent model.

LEVEL OF SIGNIFICANCE

S

POSSIBLE RESOLUTION

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element SY**	**Subelement 4**

SLC Fault Tree

The use of fault trees provides both a quantitative measure of a system's failure probability and a logic model description of the system. The SLC Fault Tree does not provide sufficient descriptive information for each basic event to allow a review to be conducted. This is considered a desirable aspect of the logic models.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Include descriptive material regarding each basic event in the logic model descriptions of each basic event.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 5**
Venting The interaction of venting with CS/LPCI could cause the CS/LPCI NPSH to be violated or steam binding of these low pressure pumps. Other BWR PRAs have attributed failure of CS/LPCI pumps when venting occurs.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Consider the impact of containment vent on continued operation of CS/LPCI pumps.		
PLANT RESPONSE OR RESOLUTION		

This affect is addressed by the adding of the new tree EMERGENCYVENT. This is used to address the NPSH head issues for ECCS pumps taking suction from the suppression pool during use of the Hardened Vent. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 5**

ECCS Suction Strainer

It is noted that the passive failure of ECCS suction strainers is included in the Hatch model:

- Each pump (RHR or CS) has its own strainer inside the torus and plugging is modeled for each of these strainers (basic events STPL1E11a(B,C,D) for RHR and STPL1E21LOO1A(B) for CS), each with a probability of 1.49E-4. There is no single plugging event modeled that fails all suction from torus.

There is no CCF of all strainers due to debris clogging. This has been included in numerous BWR PRAs to model the extremely unlikely event of debris clogging. It is recognized that Hatch has modified the ECCS suction strainers to prevent this failure mode. Typical values are:

	CCF
Large LOCA	1E-4
Med, Small LOCA	1E-5
Transient	1E-6

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider adding the ECCS suction strainer common cause failure.

PLANT RESPONSE OR RESOLUTION

This has been done by applying a new AND gate to the model that brings the probability of a LARGE LOCA and common cause failures of all low pressure ECCS strainers together. The Large LOCA value provided in this F&O of 1E-4 is used. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element SY**	**Subelement 5**

ADS/SRVS

The system modeling of SRVs and their required pneumatic supplies is quite limited. It is judged prudent to include the power supplies and pneumatic supplies (accumulators and Nitrogen backup) to the SRVs.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Add detail to the depressurization model to ensure that the required support systems, including pneumatic supplies, are included.

PLANT RESPONSE OR RESOLUTION

This has been accomplished by the addition of fault tree, SRVREMOTEOP. This models the motive force for remotely opening (from the control room) an SRV which is by the way: nitrogen—there is no backup. In addition it models the power supplies and the drywell pneumatic system which provides the pathway for the nitrogen. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element SY**	**Subelement 5**

LPCI Inject Valves

The power supplies for LPCI inject valves are still described as from inverters (see SN).

This should be corrected.

LEVEL OF SIGNIFICANCE

D

POSSIBLE RESOLUTION

Editorial

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element SY**	**Subelement 5**

DW Sprays - System Notebook

Discuss the ability to use DW sprays under accident conditions. This includes the following:

- Use of DW sprays for vapor suppression failure, i.e., elevated containment pressure.
- Use of DW sprays from external water source.
- Use of DW sprays in SAGs for elevated radiation levels.
- Use of DW sprays in conjunction with containment heat removal as the only heat removal pathway.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Consider adding the accident response of DW spray description to the RHR system notebook.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 6)	**Element SY**	**Subelement 5**

Return to Power

Historical data was used.

- This "data" is not current.
- The "data" does not show the reactor return to power.
- The "data" is for events that could be returned to power within 48 hours.
- The data includes MSIV closures and loss of condenser vacuum.
- The philosophy is contrary to safe operation.
- Not consistent with any other BWR PSA.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Remove the return to power mode from the model.

PLANT RESPONSE OR RESOLUTION

The RETURN TO POWER top event has been removed from the model. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 7)	**Element SY**	**Subelement 5**

Steam Condensing Mode

The documentation should identify whether the steam condensing mode is operational and/or procedurally allowed.

LEVEL OF SIGNIFICANCE

D

POSSIBLE RESOLUTION

Editorial

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 8)	**Element SY**	**Subelement 5**

Containment Vent

The availability of LPCI and CS for RPV injection following containment vent is assumed in the model. There is no documentation of the procedural or training guidance that would support this assertion. This is a major assumption and should be supported by operator crew input and a discussion of the configuration of the low pressure injection system suction pipe (e.g., steam binding potential).

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Reconsider the LPCI/CS operability when venting is initiated.

PLANT RESPONSE OR RESOLUTION

This has been addressed by the addition of fault tree, EMERGENCYVENT, to the rev.2 PSA model. This accounts for NPSH head concerns for low pressure ECCS after operation of the Hardened Vent. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 9)	**Element SY**	**Subelement 5**

Low pressure Permissive

It is an excellent feature that the low pressure permissive is treated for miscalibration error.

The value derived of 1.3E-5(MIUNNS) appears lower than might be derived using THERP methods.

The recovery of the low pressure permissive failure is described in the IPE as only available for cases where substantial time is available to manipulate the valves. However, the recovery credit appears to be applied to all applicable cases except Large LOCA or cases without RPV high pressure injection. Cases involving ATWS, medium and small LOCAs or IORV, and transients--all of which would appear to be clearly stressful situations.

This appears to neglect the fact that the time available to bypass or repair the permissive is potentially only known or cued, following RPV depressurization, so the time available can be very short. This is not addressed in the derivation of this recovery.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Re-examine the derivation of the HEP to account for the timing available to take action and the associated stress level.

PLANT RESPONSE OR RESOLUTION

MIUNNS has been renamed for each event that it applies to. In any case this is not a recovery. This is the probability that a miscalibration occurred and the new value, 2.7E-7, has recently been recalculated for the HRA update. The recovery of an instrument channel that is failed is now given a 1.0 or total failure value in the model—because of the reasons mentioned in this certification comment. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 10)	**Element SY**	**Subelement 5**
MSIV High Radiation The high radiation trip of the MSIVs has been removed by Hatch. This differs from the dependency matrix notes. This should be modified to more accurately reflect the plant.		
LEVEL OF SIGNIFICANCE		
D		
POSSIBLE RESOLUTION		
Editorial; ensure plant model and documentation reflect the current plant configuration.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 11)	**Element SY**	**Subelement 5**
Model does not include injection from external sources such as Fire System or RHRSW in accordance with EOPs.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Include these low pressure alternate injection sources in the model or justify why not included in the model.		
PLANT RESPONSE OR RESOLUTION		
These sources are included in the revision 2 model. This comment is closed.		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID:12)	**Element SY**	**Subelement 5**
The model does not include system interfaces for SAG implementation. One specific case is the modeling of alternate injection sources. It is understood that providing this modeling may not have a significant effect in reducing CDF or LERF; however, not including them in the model can hinder the use of the model in performing applications associated with the systems that would have been modeled.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Review the EOP and SAG and provide model changes consistent with the procedures.		
PLANT RESPONSE OR RESOLUTION		

Alternate injection sources of RHRSW and Fire Water have been included with the normal model injection sources. This pretty well takes care of all on-site sources, SAG and/or EOP related. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID:13)	**Element SY**	**Subelement 5**

LPCI/CS

Keep Fill

The LPCI/CS keep fill system may not be available during LOOP events. It is judged useful to include potential for water hammer in the LPCI/CS systems when keep fill may have allowed discharge pipe draining.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Probabilities of 1E-2 to 1E-4 are typically used for these systems when dry discharge pipes could exist.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 7**
Model includes various passive components such as check valves and strainers.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 8**
CCF of 2 Diesels on Unit 2 The 2 Unit 2 EDGs are not included in the model for Unit 1 even though RHR injection valves for Unit 1 have dependencies on Unit 2 buses.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Include D/G CCF to fail 2 Unit 2 D/Gs and all supports for LPCI injection valves.		
PLANT RESPONSE OR RESOLUTION		

The Unit 2 diesels and Unit 1 diesels are included in both models now. Common cause failure is included. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 8**
Common cause of ECCS strainers is not included in the model and the basis for exclusion is not included in the documentation.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Provide basis in documentation.		
PLANT RESPONSE OR RESOLUTION		

This has been done by applying a new AND gate to the model that brings the probability of a LARGE LOCA and common cause failures of all low pressure ECCS strainers together. The Large LOCA value provided in a SYSTEMS F&O of 1E-4 is used. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 9**
Model does not include any modules.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 10**
IA IA is not dependent on PSW. Ensure the current model reflects the change in plant configuration.		
LEVEL OF SIGNIFICANCE		
D		
POSSIBLE RESOLUTION		
Editorial		
PLANT RESPONSE OR RESOLUTION		

It does indeed. The air compressors have their own cooling system which is not related to PSW. This was information was retrieved, evidently, from historical record as opposed to the current information. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 10**
Page H3-1 of 4160-V AC Power distribution notebook under support systems states that loss of HVAC would not lead to bus problems. This is based on a walk down of plant and the site system engineer input. No calculations exist to confirm this judgement. May need a calculation to support this assumption.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Consider performing a heat up calculation to support this assumption. Including addressing the sliding fire doors that automatically close on high temperature.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element SY**	**Subelement 10**
System Notebook, H-5 DC Power Systems, has a justification for room coolers and battery exhaust fans as not being required for battery and battery charger success. The justification implies other AC and DC components do not require room cooling. This justification could be clearer as to which components are included in the evaluation.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
The inputs to the evaluation (such as design calculations and equipment evaluated), the process used, and the conclusions should be clearly stated and documented so that an independent reviewer could reach the same conclusions.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element SY**	**Subelement 10**
HPCI operation is assumed to fail if no room coolers are operating. This appears to be an overly conservative assumption because mitigating actions such as opening the room door may provide sufficient cooling for prolonged system operation		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Conduct a room heat up calculation for HPCI.		
PLANT RESPONSE OR RESOLUTION		

Failure of room cooling is a very small (negligible) cause of HPCI failure in the model at present. The accepted engineering recommendation and backed by calculations (BH-1-M-V005-005 and BH2-M-0352) is that HPCI does not function properly without room cooling. HPCI qualification temperature is 148°F with the room coolers operational. This is based on HELB events outside the HPCI room and General Electric recommendations. There is an isolation of the steam supply to HPCI when the temperature in the cooler area exceeds 165°F. This will most sure occur with HPCI operation without room cooling. This trip is allowed to be bypassed by jumpers as per EOP instructions. The various components in the HPCI room have EQ temperatures that can exceed 200°F for a few minutes but in all 148°F for 12 hours is the basis. There is no temperature limit that one can base operation on without reworking the EQ packages for HPCI. The vendor (TERRY) bases the function of the HPCI units at Hatch on our EQ profiles. Operations does not have procedures for operating HPCI without room cooling and any type of manual activity in the room without the coolers in service during HPCI operation would be extremely harsh for personnel. HPCI operation will not be considered without room coolers in service for the PRA. This comment is closed.

References: Memo, Darryl Howard to Gary McGaha, May 11,1994, Max. Temperature in ECCS Area File: ST-90012A. Memo (email) Deep Ghosh to Duane Brock, Thursday Feb.25, 1999, 7:36AM Hatch General Area Temperatures
EQ files SECTION D QDP51 Correspondence (This is the entire HPCI EQ package correspondence)

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 5)	**Element SY**	**Subelement 10**

HPCI Room Cooling

The HPCI room is quite large and the time to HPCI high temperature failure in the room is believed to occur at greater than one hour. This is expected to delay the need for offsite AC power recovery by several hours if only HPCI is available, and by many hours if HPCI or RCIC could be operated in series on an as-needed basis.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Add HPCI capability to the model for SBO mitigation. This is expected to increase the time allowed for offsite AC recovery. This may be needed when the offsite AC power recovery probability is "fixed."

PLANT RESPONSE OR RESOLUTION

Calculations BH1-M-V005-0005 and BH2-M-0351(HPCI ROOM HEAT LOAD 12/3/02) are the Units 1 and 2 HPCI room heat up calculations. These show the effectiveness of a HPCI Room Cooler to keep the room at 105°F in a HPCI standby mode and 148°F in an operating mode. The cooler is close to its capacity to maintain the room temperature at 148°F. This is based on heat load being put into the room on an hourly basis. The door openings to the room would not be able to hold the temperature in check without room cooling. Based on the cooler load it is engineering judgement to say that the HPCI Cooler Outlet Temp. Steam Line Isolation would be invoked at 165°F without cooling. If the operators attempted to run HPCI and shut HPCI down to hold the cooling load, battery capacity (without chargers) would not allow the repeated start cycles and HPCI would be inoperative anyway. It is known that HPCI operation could provide support in SBO cases, but modeling this capability with any certainity is not possible. As a result HPCI is not considered for the SBO case.

Failure of room cooling is a very small (negligible) cause of HPCI failure in the model at present. The accepted engineering recommendation and backed by calculations (BH-1-M-V005-005 and BH2-M-0352) is that HPCI does not function properly without room cooling. HPCI qualification temperature is 148°F with the room coolers operational. This is based on HELB events outside the HPCI room and General Electric recommendations. There is an isolation of the steam supply to HPCI when the temperature in the cooler area exceeds 165°F. This will most sure occur with HPCI operation without room cooling. This trip is allowed to be bypassed by jumpers as per EOP instructions. The various components in the HPCI room have EQ temperatures that can exceed 200°F for a few minutes but in all 148°F for 12 hours is the basis. There is no temperature limit that one can base operation on without reworking the EQ packages for HPCI. The vendor (TERRY) bases the function of the HPCI units at Hatch on our EQ profiles. Operations does not have procedures for operating HPCI without room cooling and any type of manual activity in the room without the coolers in service during HPCI operation would be extremely harsh for personnel. **HPCI operation will not be considered without room coolers in service for the PRA. This comment is closed.**

References: Memo, Darryl Howard to Gary McGaha, May 11,1994, Max. Temperature in ECCS Area File: ST-90012A. Memo (email) Deep Ghosh to Duane Brock, Thursday Feb.25, 1999, 7:36AM Hatch General Area Temperatures
EQ files SECTION D QDP51 Correspondence (This is the entire HPCI EQ package correspondence)

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 6)	**Element SY**	**Subelement 10**

600V AC BUS 1C

It appears that loss of cooling for the 600V 1C bus compartment could lead to failure of the critical 600V bus given the following.

- Loss of normal HVAC which is completely dependent on IA (see comment on the loss of air initiating event).

AND

- Failure of the operator action to establish natural cooling (stack effect). (It is assumed there is a procedure for this action.)

OR

Failure to prevent the sliding fire door on the compartment from closing.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Review the model to ensure that success of 600V AC Bus 1C includes the above potential failure modes or provide clear justification.

PLANT RESPONSE OR RESOLUTION

Operation of this equipment without cooling is being evaluated by SNC engineering at present. Until the evaluation is complete the need for ventilation is not included in the model.

r

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 11**
Degraded Environments Pool adverse impact on LPCI and CS is not addressed.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Re-establish the basis for LPCI and CS operation at high pool temperatures, especially under venting conditions. Correlate the failure modes with system capability.		
PLANT RESPONSE OR RESOLUTION		

LPCI and CS operation at high suppression pool temps follow the NPSH curves as per the EOPS. Pool temperature and the model sequences are looked at to show pool failure at 260°F. Emergency Venting affects on LPCI and CS operation are accounted for with the inclusion of the EMERGENCYVENT model tree in the rev. 2 model. These discussions are in the model Success Criteria and in the model calculation sequence description.
This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 11**

The ADS/SRV note book quotes reference U (a GE letter) which identifies a 50 psia air (nitrogen) pressure across the actuating diaphragm to open the SRV and 45 psia to maintain it open. After looking at the pressure suppression pressure curve and the PCPL curve, it is not clear if either or both have any parts of the curve based on SRV operability. The calculation did not easily reveal this information either.

This appears not to be a concern with the PRA since the accident sequences would run as modeled; however, the SRVs would shut earlier than anticipated if the pressures are calculated incorrectly. This may be an EOP calculation problem.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Examine the calculation basis for the two curves and determine if the differential pressure needed to open the SRV is correctly accounted for.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 12**

System Notebook

Establish whether a support system is critical to the system operation. This is a place where the dependency matrix can be discussed and any clarifications identified.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Enhance the System Notebook Description to specify the degree of dependency on support system.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 12**
Support systems are modeled and documented in system notebooks and dependency matrix.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 13**

RCIC SN

- Mission time six hours vs. 24 hours, p. 9.
- Room cooling required, pp. 3 & 8 No
- Room cooling required, p. 9 Yes (Unit 1)

The event tree analysis does not appear to assume a 6 hour mission time.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Resolve the mission time assigned to RCIC operation, its supports, and its use in the model.

PLANT RESPONSE OR RESOLUTION

RCIC and HPCI mission times are 4 hours and 6 hours respectively. This is not continuous operation. Room cooling is required for HPCI. This comment is closed with the failure rates of RCIC and HPCI being adjusted for 6 and 4 hours respectively.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 13**
The LPCI inverters were removed from the plant and the model. This was a major load on the battery. Battery expected life should be significantly higher without this load. This will affect time allowed to recover AC when HPCI available on the battery. Same comment under AS 5.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Revise battery expected life based on LPCI inverters no longer a load. Based on revised lifetime of battery revise LOSP recovery factors when HPI is available.		
PLANT RESPONSE OR RESOLUTION		

This comment has been addressed and all has been done. Battery life is now 5 hours with RCIC operation allowed during an SBO. LOSP Recovery Factors have been recalculated. This is part of the rev. 2 model calculation and also part of the diesel AOT extension calculation performed for Hatch (see PSA-H-01-003).

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 14**
RHRSW model includes valves not defined in the simplified system boundary schematic. MOV1E11F003A and MOV1E11F0047A are not shown in the diagrams.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Ensure consistency of the model with the boundaries defined in the schematic		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 14**
PSW model includes a standby pump not defined in the simplified system boundary schematic.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Update schematic		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element SY**	**Subelement 14**
The Condensate and Feedwater System Notebook (H31) is not defined (shown) in the simplified system boundary schematic.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Update schematic.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 15**
The RHRSW model does not include any short cycle pump train failures.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Incorporate failure mode during next update.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 18**
The basic event nomenclature is not uniform or consistent (e.g., operator actions basic events not uniform, some begin with OP, some do not; independent failure basic events begin with CC-, etc.).		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Update the basic event Ids to be consistent and uniform during the next update.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 19**

RPS

Scram failure probability is composed of:

- CCF of mechanical scram components 1E-5
- Fault tree for electrical portion (~ 2E-6)

The mechanical portion of the scram failure probability is consistent with the current state of other technology. A slightly lower value (~ 2.5E-6) could be used based on the INEEL report on scram failure probability. This is not recommended until the BWROG agrees with the use of this report.

The electrical common cause failure probability appear underestimated, but is consistent with INEEL. (NUREG/CR-5500, Volume 3)

LEVEL OF SIGNIFICANCE

D

POSSIBLE RESOLUTION

No action required.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 19**
System models are detailed and include electrical switchyard.		
LEVEL OF SIGNIFICANCE		
S		
POSSIBLE RESOLUTION		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 23**
The system notebooks do not address severe accident conditions. A specific example is the SRV/ADS notebook.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Now that the plant has implemented EPG/SAG the notebooks should reflect operation of systems under these conditions.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 25**
Documentation is in several places.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Consider updating process documentation such as HO and providing an overview of where process documentation exists for common cause, dependence, initiating event impact, etc.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element SY**	**Subelement 27**

Containment Vent

The Calc H44.1 and CAFTA conversion notebook for containment vent were reviewed.

- No simplified diagram (see IPE for a figure).
- System boundaries are not discussed.
- Rupture disk failure is not included (rupture disk setting is not identified).
- No impact of initiation.
- Support systems not discussed.
- Procedural interface not discussed.
- Throttle capability not addressed.
- Control band not addressed.

The fault tree model appears excellent and reflects pertinent failure modes, system interfaces, and support systems.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Improve containment vent documentation.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element SY**	**Subelement 27**

All System Notebooks

The System Notebooks have a substantial amount of vital information to support the PRA.

Five areas where the System Notebooks that could be enhanced are the following:

- Success criteria are not clearly defined.
- Mission time is not defined for all applicable cases.
- System boundary not defined.
- Simplified figures could be redrawn (vent missing completely).
- Nomenclature is atypical.

LEVEL OF SIGNIFICANCE

Priority:

B - Success Criteria
C - Mission Time
D- Simplified Figures
C- Boundary definition
D- Nomenclature

POSSIBLE RESOLUTION

Update the System Notebooks to clarify the above items to ensure consistent interpretation of the PSA results.

PLANT RESPONSE OR RESOLUTION

Success Criteria is being updated for the Rev.2 Hatch PSA model. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element SY**	**Subelement 27**

SLC

Conversion Work Package

P. 5 of this document appears to provide a "shortcut" description of the top gates--B1.

#B1 - Failure of boron injection when boron injection initiation temperature is exceeded.

#TINJ - Failure to terminate all high pressure injection and lower vessel level to top of active fuel when required.

LEVEL OF SIGNIFICANCE

D

POSSIBLE RESOLUTION

These descriptions are incorrect:

- SLC is to be initiated before exceeding the BIIT. SLC failure is not assumed if BIIT is exceeded.
- Terminating high pressure injection does not necessarily stop at top of active fuel.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 4)	**Element SY**	**Subelement 27**
The system notebooks do not address severe accident conditions. A specific example is the SRV/ADS notebook. Documentation does not provide direct reference to plant specific analysis such as MAAP.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Update documentation to reference plant specific analysis..		
PLANT RESPONSE OR RESOLUTION		

THERMAL HYDRAULIC (TH)

FACTS AND OBSERVATIONS

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 1**

A description of the approach to be used for determining the need for thermal hydraulic (T&H) calculations and the type of T&H calculation to perform along with the output needed is desirable.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

The incorporation of documentation on the T&H approach for future updates may not be needed for most Grade 3 applications but it would be desirable to include if resources permit. It may by useful to provide a basis for the following:

- Tabulation scheme for calculations to identify specific deterministic runs with an ID
- Limitations of codes
- Code comparisons
- Areas where realistic codes may be suspect

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 3**
The RPT success criteria is judged appropriate (i.e., 2 pumps must trip, RPT and ARI are required – not just ARI by itself); however, the bases for the RPT success criteria are not provided. The specific short term issue is the RPV pressure response given a failure to scram.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
Provide a thermal hydraulic reference basis for RPT success criteria. It is generally found that a computer code such as REDY or ODYN will yield pressure responses of approximately 1600 psig or higher within 9 seconds if RPT fails. Existing GE analyses are documented NEDE-24708A.		
PLANT RESPONSE OR RESOLUTION		

The success criteria for the ATWS trip is shown in the Appendix K Thermal Power Optimization Report for Hatch Units 1 and 2. GE-NE-0000-000308305-01, Rev. 0, September 2002. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 4**

Success Criteria

The PSW success criteria for shutdown operations are based on FSAR, training, and judgment. A single PSW pump is considered adequate in each division (this is explicitly supported by FSAR 107.3), or any two PSW pumps if Turbine Building loads are to be used. This success criteria appears reasonable and appropriate.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

It is desirable to provide any operating experience insights into the determination of the PSW success criteria that are available to provide additional confidence in the success criteria, and to explicitly discuss this in the system notebook.

PLANT RESPONSE OR RESOLUTION
Success criteria for the PSW system is 1 pump in each division with the turbine building isolation valves closed based on an LOSP and loaded diesel operation. The calculation number is SMNH-05-002 for REA04243560. In addition at least one Intake Structure Fan must be in service for PSW pump operation.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element TH**	**Subelement 4**

Success Criteria

SNC cited the IPE as the basis for the success criteria. In the IPE, SDC success is cited as feasible for accident sequences in:

- Tables 3.1-3 (events with successful scram)
- Tables 3.1-4 ATWS with SBLC

In both cases, these are accident sequences in which RPV water level can drop below Level 3 and cause SDC isolation. The use of the SDC system with RPV water level below Level 3 should be clarified in the success criteria discussion.

The discussion of SDC for heat removal should account for the potential to be interrupted or presented by low RPV level or high RPV pressure interlocks.

LEVEL OF SIGNIFICANCE
D
POSSIBLE RESOLUTION
Ensure description accounts for SDC limitations. Model has been examined to be correct. Additional SNC verification could also be performed.
PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element TH**	**Subelement 4**
No success criteria are provided for the Vapor Suppression function (important during MLOCAs and LLOCAs). For example, such success criteria would address the number of stuck-open WW-DW vacuum breakers that may be allowed during a blowdown and if initiation of DW sprays is credited.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		

Develop and document such success criteria and include this critical safety function in the appropriate accident sequences.

PLANT RESPONSE OR RESOLUTION

The requirement for the allowed number of stuck open drywell to torus vacuum breakers is one (1) which is shown in the vapor suppression tree (VAPSUPPRESSION) in the rev.2 model. Two or more stuck open vacuum breakers leads to containment problems. Drywell spray or normal venting must fail before the damage occurs. Containment failure during the largest postulated medium LOCA (0.4 FT^2) occurs very late. Containment failure for the large LOCA occurs early. This is assumed due to the wide range of large LOCAs. Success criteria are provided with revision 2 of the Hatch PSA model for the medium LOCA case. The large LOCA case is based on engineering judgement for the circumferential recirc suction line break. Drywell volume (free space included suppression chamber air space) is approximately 268000 FT^3. If the blowdown data for the large LOCA from the FSAR chapter 6 analysis is used and 20% is assumed to flash to steam, a 5 second blowdown will reach over 100 psia in the containment. Engineering judgement is therefore used to postulate failure of the containment and subsequent core damage for 2 stuck open drywell to torus vacuum breakers with any large LOCA (greater than 0.4 FT^2). This allows for error and considers that the calculation for the medium LOCA case alone reaches over 60psia in the drywell.
This comment is considered closed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID 4)	Element TH	Subelement 4

The Power Uprate MAAP runs recently performed to provide bases and to “re-confirm” the existing PRA success criteria cite MAAP Run F as the basis for the success criteria of 3 SRVs needed for RPV Emergency Depressurization. However, MAAP Run F is a SORV case with 2 SRVs stuck open at time zero. Considering that the Emergency Depressurization human action modeling in the PRA requires waiting over an hour before initiating ED, this case is not directly applicable.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Considering that MAAP Run F uses 2 SORVs and the selected success criteria for RPV ED is 3 SRVs, the model is almost assuredly not erroneously non-conservative. However, a discussion should be provided that address the issues why this case does not model the appropriate scenario but it can be used as a surrogate to provide a basis for the 3 SRVs success criteria. Alternatively, run a directly applicable MAAP case in which initiation of RPV ED (using 2 or 3 SRVs) is put off as long as possible (e.g., about an hour, until level reaches 1/2 - 1/3 core height, etc.).

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING

PSA TECHNICAL ELEMENTS

OBSERVATION (D: 1)	Element TH	Subelement 5

T & H

The success criteria and their bases should be compared with other sources to confirm the realistic nature of the analysis and to provide an indication of where limitations in the particular code are potential areas to exercise caution when performing applications.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Provide comparison of T & H code results to identify areas of potential code limitations. A specific example would be a comparison of the Hatch results with NEDO 24708A results for similar plants.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	**Element TH**	**Subelement 5**

<u>Success Criteria</u>

RCIC is assumed in the Hatch PRA inadequate under SORV cases. The RCIC system with an SORV, however, in NEDO 24708A is shown to allow RPV depressurization with adequate injection until low pressure systems are available. Therefore, Hatch is conservative relative to the available generic T & H calculation.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Revise the success criteria for SORV cases to allow RCIC success until low pressure injection systems can provide makeup.

PLANT RESPONSE OR RESOLUTION

The model has been revised to allow RCIC success with a stuck open SRV—except for the station blackout case. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 7**
MAAP cannot be used in the evaluation of early overpressure failure of the primary system. Codes such as REDY, ODYN, or TRAC-G are capable of providing this deterministic input.		
LEVEL OF SIGNIFICANCE		
B		
POSSIBLE RESOLUTION		
Formulate the success criteria for Hatch based on qualified T & H codes or clearly understand and document the limitations of the codes as applied to specific accident sequences.		
PLANT RESPONSE OR RESOLUTION		

General Electric prepared the ATWS analysis for our Appendix K uprate. It is this information that determines the number of SRVs required to keep primary pressure below the ATWS upset pressure limit. The report uses codes such as those mentioned since they are GE codes. The report number is GE-NE-0000-003-8305-01, Rev. 0 Sept. 2002 Thermal Power Optimization. MAAP does not evaluate accurately failure in the first 5 seconds of an accident which is what the referenced codes can do. However, MAAP is acceptable for the first hour of the transient for PRA usage. The ATWS case is the only item that seems to apply to this comment. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION (ID: 2)	**Element TH**	**Subelement 7**

Level Indication

The FAI calculations cited as the basis for accident sequence timing do not provide a description of the RPV water level that the MAAP calculation corresponds to:

- Core level
- Shroud level
- Fuel zone indicated level

LEVEL OF SIGNIFICANCE
C
POSSIBLE RESOLUTION
Provide a description of how to interpret the FAI MAAP calculations relative to what the operators will see in the control room.
PLANT RESPONSE OR RESOLUTION
All MAAP calculations are shroud level. Any case where the true core level is used is now defined. This comment is considered closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 8**
Intake Ventilation No calculation is available to support PSW operation with no fans operating during summer months. Peer Review Team walkdown confirmed that there appears to be sufficient uncertainty regarding PSW pump room cooling to question operability of PSW, RHRSW, and DGCW pumps.		

LEVEL OF SIGNIFICANCE
C
POSSIBLE RESOLUTION
Consideration should be given to the incorporation of the need for fan ventilation during hot periods of the year.
PLANT RESPONSE OR RESOLUTION
A fan failure tree has been incorporated into the Hatch PSA model. This is an OR Gate named INTSTRUCTUREFANS. This comment is closed.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element TH**	**Subelement 8**
There are no room heat-up calculations for HPCI. It is assumed that if room cooling is not available, HPCI fails. This is overly conservative and can distort the results by 1) making HPCI less important and 2) increasing the CDF and LERF. This was also confirmed as potentially conservative during the walkdown by the Peer Review Team members by discussion with plant Hatch staff.		

LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Conduct HPCI room heat-up calculations and consider crediting operator action to open room doors if a procedure and training are provided.
PLANT RESPONSE OR RESOLUTION

Calculations BH1-M-V005-0005 and BH2-M-0351(HPCI ROOM HEAT LOAD 12/3/02) are the Units 1 and 2 HPCI room heat up calculations. These show the effectiveness of a HPCI Room Cooler to keep the room at 105°F in a HPCI standby mode and 148°F in an operating mode. The cooler is close to its capacity to maintain the room temperature at 148°F. This is based on heat load being put into the room on an hourly basis. The door openings to the room would not be able to hold the temperature in check without room cooling. Based on the cooler load it is engineering judgement to say that the HPCI Cooler Outlet Temp. Steam Line Isolation would be invoked at 165°F without cooling. If the operators attempted to run HPCI and shut HPCI down to hold the cooling load, battery capacity would not allow the repeated start cycles and HPCI would be inoperative anyway. This is the LOSP case where HPCI operation on battery power could supplement RCIC. Cases where HPCI actually fails due to room cooling are extremely low worth in the model. The other option of removing the equipment hatch to the HPCI rooms requires a large outside crane. Modeling items like this is more speculative than factual. The calculations set for the room heat up exist. The plant personnel are well aware of them and in an extreme emergency all that could be done to operate HPCI—if no other source of water were available—would be done.

Failure of room cooling is a very small (negligible) cause of HPCI failure in the model at present. The accepted engineering recommendation and backed by calculations (BH-1-M-V005-005 and BH2-M-0352) is that HPCI does not function properly without room cooling. HPCI qualification temperature is 148°F with the room coolers operational. This is based on HELB events outside the HPCI room and General Electric recommendations. There is an isolation of the steam supply to HPCI when the temperature in the cooler area exceeds 165°F. This will most sure occur with HPCI operation without room cooling. This trip is allowed to be bypassed by jumpers as per EOP instructions. The various components in the HPCI room have EQ temperatures that can exceed 200°F for a few minutes but in all 148°F for 12 hours is the basis. There is no temperature limit that one can base operation on without reworking the EQ packages for HPCI. The vendor (TERRY) bases the function of the HPCI units at Hatch on our EQ profiles. Operations does not have procedures for operating HPCI without room cooling and any type of manual activity in the room without the coolers in service during HPCI operation would be extremely harsh for personnel. **HPCI operation will not be considered without room coolers in service for the PRA. This comment is closed.**

References: Memo, Darryl Howard to Gary McGaha, May 11,1994, Max. Temperature in ECCS Area File: ST-90012A. Memo (email) Deep Ghosh to Duane Brock, Thursday Feb.25, 1999, 7:36AM Hatch General Area Temperatures
EQ files SECTION D QDP51 Correspondence (This is the entire HPCI EQ package correspondence)

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element TH**	**Subelement 8**

<u>Room Cooling Calculation</u>

The lack of room cooling calculations and the observations of the Peer Review Team members during the plant walkthrough both indicate a reconsideration of the modeling of the intake room cooling modeling is prudent.

The PSW and RHRSW operability under loss of ventilation may be questionable at certain times during the year. The pump intake building (a single congested compartment) contains 8 PSW pumps and 8 RHRSW pumps in a confined location. During an event that challenges RHRSW and PSW (e.g., SCRAM, failure to SCRAM with loss of condenser vacuum, loss of offsite power, dual unit loss of offsite power, and dual unit loss of air (turbine building flood) occurring during hot times of the year) it is believed the temperature in the intake structure could exceed the sprinkler temperature. Adverse impacts from sprinkler operation do not appear to be addressed. (It is noted that information provided in a LERF indicates actuation of the sprinklers is a "good" event for room temperature control. However, the impact on pump motors and "surge packs" is not addressed. It must be noted that the sprinkler head appears to be directed at the pump motors but it is highly unlikely that power surge packs would be affected.)

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Include the intake structure ventilation fans in the model. (See attached model suggestion.) If found appropriate, include the fire protection system and sprinklers in the model as either a beneficial cooling method or potential failure mode as appropriate (or both).

PLANT RESPONSE OR RESOLUTION

The intake structure ventilation fans have been included in the model. The tree is represented in the model as OR Gate INTSTRUCTUREFANS.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 9**

Success Criteria

The model was investigated and determined to differ significantly from the success criteria listed in the IPE (source of the overall success criteria). These areas all proved appropriate in the model, i.e., the written success criteria are considered nonconservative. The documentation should be modified. These include:

- ATWS: SBLC failure with level control and torus cooling is called a success
- ATWS: SDC with SBLC is called a success.
- Vapor Suppression: No success criteria identified.
- RHRSW: Not included as an injection source; this is not in the model; no technical support was identified for including in the model.
- LOCA: RHR in SDC is listed as a success. The model has appropriately eliminated this from the success path in the fault tree logic.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Explain all implementations of the success criteria in the event tree notebook.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 2)	**Element TH**	**Subelement 9**

The ability to perform MAAP runs in-house is a good capability and should be taken advantage of as often as practical (to promote the skill and to provide bases for modeling). However, many of the MAAP run packages are not documented well as they could be and contain varying levels of output documentation.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Provide standard cover sheets on all MAAP runs that summarize, at a minimum, the following:

- Run No.
- Run Title
- Purpose(s)
- Name of individual performing run
- Date of run
- Run input details
- Key results
- Key conclusions

Ensure that each MAAP run package contains similar input and output in the package and has at least a minimum number of key output plots of key parameters.

PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 3)	**Element TH**	**Subelement 9**
The success criteria and supporting thermal hydraulic calculations are located in various locations.		
LEVEL OF SIGNIFICANCE		
C		
POSSIBLE RESOLUTION		
When resources become available, it would be desirable to collect all related calculations into a single coherent set of volumes. Other plants take this approach with such documents as Level 1 and 2 Success Criteria, and/or Deterministic Calculations Notebook.		
PLANT RESPONSE OR RESOLUTION		

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS		
OBSERVATION (ID: 1)	**Element TH**	**Subelement 10**

Documentation Reflects Process

The success criteria are summarized in the IPE. This should be updated and converted into a current, living document.

The use of MAAP to support individual success criteria is considered excellent; however, there should be a direct correlation between the success criteria and the specific MAAP cases being used to support the success criteria and the HRA timing.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

Document the success criteria basis by referencing the specific calculations.
PLANT RESPONSE OR RESOLUTION