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SUBJECT: Provides preliminary response to points raised at IPE info 971217 meeting re plant IPE,per GL 88-20.NUREG/CP-0132 encl also.

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SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO POINTS RAISED AT
IPE INFORMATION MEETING
PLA-4631 FILE R41-2

Docket Nos. 50-387
and 50-388

This letter is intended to provide preliminary responses to points raised at the IPE information meeting December 17, 1996 regarding the Susquehanna SES IPE. The attached responses address the issues identified by Staff reviewers in their prepared slides as well as additional oral questions asked during the course of the meeting and noted by PP&L attendees. The format of the responses is a statement of our understanding of each of the NRC questions followed by the PP&L position.

It may be noted that to address the NRC concerns typically associated with the Susquehanna IPE, specifically those of human error modeling, common cause failure, and phenomenological completeness, PP&L contracted with an independent reviewer to perform a review of the Susquehanna IPE. This nationally known PRA authority reviewed not only the SSES IPE, but also the record of correspondence between our Staffs. The review concluded that the SSES IPE method, although different from the norm, is consistent with the direction of GL 88-20 in that it provided insights regarding possible plant vulnerabilities and has, in fact, resulted in dramatic improvements to SSES robustness with regard to nuclear safety. However, the review suggests that appropriate sensitivity and uncertainty analyses be performed, with emphasis on human error and equipment common cause failure, to increase confidence in the IPE conclusions while providing your Staff with an acceptable basis for acceptance of the overall work. We would appreciate Staff concurrence that a proposed effort involving such studies is a possible method of resolving our differences.

A011

After review of the attached materials by the Staff, PP&L suggests a meeting with the reviewers to answer any additional questions and discuss the scope, schedule, and focus of additional work including the proposed human error/common cause failure sensitivity analyses.

Very truly yours,

R. G. Byram

Attachment

copy: NRC Region I
Mr. C. Poslusny, Jr., NRC Sr. Project Manager - OWFN
Mr. K. M. Jenison, NRC Sr. Resident Inspector - SSES

As Built-As Operated vs. IPE Basis

1. *To what extent were plant system walk downs and procedure walk throughs performed? In particular, was a flooding walk down performed?*

Plant walk downs were performed as needed to verify plant designs. Walk downs included: verification of primary containment vent pathway, location of CRD flow control skid, design of scram discharge volume and associated instrumentation, RHR system piping for susceptibility of water hammer. The primary containment was walked down with specific emphasis on severe accident concerns. NRC research personnel and their ORNL contractors accompanied us in this walk down to facilitate their work on the Containment Performance Improvement Program (CPIP). Walk downs were performed to verify implementation of Emergency Support (ES) procedures, such as alignment of the fire main system for RPV injection, or alignment of the mobile generator. Walk downs were performed to verify inputs to HVAC models of the Reactor Building, Control Structure and the Emergency Service Water Pump House. Additionally, temperature measurements have been taken to verify assumptions concerning electrical cabinet internal vs. outside ambient temperature differences (PLI-71824).

In addition to walk downs by IPE personnel, the IPE was reviewed by plant system engineering personnel, plant operations, and Licensed Operator training personnel to verify the veracity of the IPE model. The review process is summarized in Section 5.2 of the IPE and documented in PP&L recorded calculation EC-RISK-0513. Emergency Operating Procedures were walked down in the simulator and exercises were witnessed. The later exercises were witnessed by EPRI personnel as part of EPRI's Control Room Operator Response Study.

A plant walk down was specifically performed for the internal flooding study. The results of this walk down are documented in PP&L recorded calculation RA-B-NA-031, Internal Flooding Analysis for the SSES IPE.

2. *As SSES is a dual unit plant, were Unit to Unit interactions examined?*

Dual unit considerations were included in the Susquehanna IPE. The results are summarized in Appendix G, "Influence of Shared Equipment." The greatest impact of shared equipment is to reduce the conditional containment failure probability from 0.1 for a single unit to 0.01 when equipment is shared between the units. This is largely due to the ability to rely on either unit's DC control power to operate support systems and decay heat removal equipment.

3. *What is the status of the analytical studies identified in Section 6.3? Specifically, discuss the following three issues:*

   a) *Addition of the wetwell vent to be actuated in non-ATWS events, when there is no core damage.*

   Procedures for venting the primary containment (ES-1/273-003/4) have been put in place for venting. These procedures utilize existing SGTS duct work. Since the SGTS vent duct is expected to fail once the vent is opened, actions are specified in the procedure, (Section 4.2) to provide a source of core cooling that is independent of equipment (mechanical, electrical and I&C) located in reactor building. It is PP&L's intention to address the issues raised by the NRC in their review of the BWROG Accident Management Procedures concerning primary containment venting. Therefore the venting strategy is currently under evaluation as part of the accident management implementation project.

   b) *ADS SRV control during isolation events.*

   Control of ADS SRVs during isolation events has been dispositioned. The disposition depends upon whether the reactor is shut down or not. When the reactor is shut down sufficient time and facilities exist to dispatch an operator to either relay room to manually control the SRVs as needed. (The relay rooms are above and below the control room. All six ADS SRVs can be controlled from either relay room.) This action is proceduralized and the operators are tested on its implementation (Job Performance Measure 6.00.009.103, Perform Manual Operation of ADS Valves from Relay Rooms as Required by ON-1/200-109).

   In failure to scram events the RPV water level is very sensitive to SRV actuations when the RPV water level is being controlled around TAF. This is due to the relatively small free area at that region of the RPV and mass lost when an SRV opens. This problem has been avoided by changing the level control band for ATWS events. Unlike the BWROG procedures which instruct the operators to control RPV water level between Top of Active Fuel (TAF) and 2.5 feet below TAF, the Susquehanna level control band is TAF (-161") to -60" (about 2 feet below the feed water sparger) with a target band of -110 to -60. Using the procedure the operator has a 101 inch control band. Additionally the RPV free area in the PP&L target control band is about 300 ft^2 compared to 88 ft^2 at TAF and 15.6 ft^2 2.5 feet below TAF. Since the level response is inversely proportional to the free area, the PP&L procedure is much less responsive to SRV actuation than the BWROG procedures. The PP&L procedures have been approved by the NRC who have urged other BWRs to adopt our level control procedure.

   c) *LOCA load shed and high drywell pressure isolation.*

Several studies (e.g. EC-004-0522 provides bases for loading the D condensate pump and includes references to other work) were performed to evaluate plant voltage response to a LOCA signal. As a result of this study the Aux load shed scheme was modified to allow the operators to reload the D condensate pump on to the Aux bus and inject water into the RPV for vessel injection.

4. *Have all improvements credited in the IPE been implemented? If not, what is the effect of those not implemented on the Core Damage Frequency?* (e.g., raising the suppression pool level to extend time to HCTL?)

All the improvements described in the IPE, plus additional modifications, have been incorporated into the units with the exception of the modification to the HPCI suction transfer logic. This modification will be installed in the units during the next refueling outage. Additionally, instead of adding a power supply to the condensate transfer pump, water hammer concerns have been alleviated via procedure changes and a second diesel fire pump is used to provide a backup source of low pressure water for extended SBO.

The impact of these modifications on the calculated core damage frequency has been evaluated using NUREG-1150 Peach Bottom analysis. The accident sequences and cut sets described in NUREG-4550 were amended to include the Susquehanna modifications not incorporated in the NUREG-1150 Peach Bottom design. As an example, a switch was installed to allow the operator to bypass the Rod Sequence Control system (RSCS) during ATWS events. Bypassing the RSCS allows the operator to manually insert control rods during ATWS. This method, as described in the attached Water Reactor Safety Information Meeting paper, represents a success path that is diverse from SLCS and was installed at Susquehanna to satisfy the severe accident defense in depth criteria described in the IPE. NUREG-1150 data and models were used in the sensitivity study when available to ensure modeling differences were not responsible for the risk reduction. This sensitivity study was reviewed by Dr. William Vesley. As shown in this sensitivity, the additional modifications made to the Susquehanna Plant significantly reduce the calculated core damage frequency. This sensitivity study is attached to this response.

Human Reliability Analysis

A. Pre-initiator Human Errors

5. *Pre-initiator human errors can lead to common-cause failures of important equipment; such failures will not be identified without a careful examination of plant practices. The SSES IPE assumed that equipment failures include the contribution of human errors performed during normal operations. It was not demonstrated that a systematic examination of plant procedures and practices was performed to confirm the applicability of the assumption. In particular, discuss how the following issues were addressed during the development of the SSES IPE.*

a. *Were the maintenance, test, and calibration procedures for the systems and components reviewed by the IPE analyst? Did the reviews include discussion with operations and maintenance personnel?*

Errors which lead to equipment unavailability or initiating events are incorporated into the equipment reliability blocks. The general method used to address these failures is provided in Section 2.3.6. Specific data is located in PP&L recorded calculation RA-B-NA-033, Analysis of Component Outage & Failure Data for Use in the SSES IPE. This calculation is 3079 pages long and can be reviewed at the PP&L corporate offices. An example from this calculation is provided as Attachment B. In this example, the human error associated with test and maintenance is "incorrect fuses were removed, resulting in Unit 1 HPCI inop." Unavailability associated with these types of operator errors are accumulated for each reliability block and incorporated into the accident sequence frequency calculation. Such errors are monitored as part of the company's condition report program. When they occur, corrective actions are taken.

In addition to the review of plant records to identify when such errors occurred, procedures were reviewed when the IPE analysts deemed it appropriate. Discussions with plant operations and I&C engineers were performed to understand frequency of testing and maintenance practices. Additionally, operations, systems engineers, maintenance and training personnel reviewed the IPE and provided many comments, including corrections/additions to equipment modeling. Examples of pre-initiator analysis is provided below.

Pre-initiator human errors were identified in the NUREG-1150 Peach Bottom analysis as dominant contributors to failure of SLCS and the Low Pressure Permissive (LPP) circuit. For the IPE, PP&L test and maintenance procedures were reviewed for these systems (RA-B-NA-033 for SLCS and the LPP and additionally IPE Section C.6 page C-132) due to their importance. Failure to restore SLCS was not found to be a big contributor for SLCS. A common cause couple of 0.01 was assigned to the low pressure permissive to account for possible pre-initiator human errors.

In addition to the evaluation of potential pre-initiator errors that would render these systems failed, modifications were installed that allow the control room operator to recover from these or other errors in these systems. For example, a switch was installed which allows the operator to bypass the RSCS and manually insert control rods (MRI) in case of ATWS. MRI is diverse to the SLCS system and is proceduralized and is initiated independent of the status of SLCS. A switch was also installed which allows the operator to bypass the failed low pressure permissive circuit. These modifications were motivated by

the HRA method employed by PP&L (see attached HRA paper presented at the ARS conference.)

In summary, the Susquehanna analysis evaluated identical errors as NUREG-1150, however modifications were installed at Susquehanna to ameliorate the impacts of these errors. The modifications were installed to comply with the defense in depth criteria that the Susquehanna IPE is based upon.

b. *Was the use in some cases of plant specific rather than generic data appropriately justified?*

Outages of equipment for all causes, including human error, was obtained from plant records. This included errors from improper implementation of the maintenance procedure as shown in the attached HPCI example. In some cases, as in the case of the low pressure permissive, a common cause failure was assigned to equipment to account for errors such as mis-calibration. Assignments of plant specific failure rates, including human error, are based on the plant data itself, supplemented by review of appropriate procedures or prior analysis. In the case of the low pressure permissive, initial work was performed by NUS corporation. Their initial estimate of failure of the LPP was 10^{-4} /demand. This value was reduced by a factor of 10 in the IPE for the following reason:

A key assumption in the NUS analysis was that all 4 switches were the same. In reality two different pressure sensors are used, a bordon tube and a diaphragm. Due to the difference in device types, separate procedural steps and tables are used for the bourdon tube and the diaphragm devices. Therefore they are less susceptible to common cause failure due to calibration error.

c. *Were recovery factors applied to pre-initiator human error? If so were the recovery actions justified?*

No credit was taken for recovery of failed equipment for any reason, unless explicitly stated. The equipment available for recovery includes: Offsite Power, Diesel Generators, and TBCCW. Additionally, credit was taken for manual aligning MOVs after loss of AC motive power (e.g. during SBO). The probability of recovery of this equipment is based on the length of time available and the type of recovery action anticipated (opening a valve vs. re-building a pump). Probability vs. time relationships are obtained from either plant maintenance data or from NRC NUREG/CRs -2886, 3154.

B. <u>Post-initiator Human Errors</u>

6. *A fundamental assumption in the Susquehanna IPE is that the operators will enter and follow procedures. What impact does this assumption have on: the core damage and containment failure frequencies, and the lessons learned from the IPE analysis?*

The rational for this assumption is provided in the attached paper on PP&L's HRA method that was recently presented at the Advanced Reactor Safety Conference. If one assumes that the operator fails to follow procedures, then the outcome of this error is unknown. However, one could make a credible argument that the plant damage frequency is on the same order as the procedural error probability. As an example, failure of the operator to follow the procedures to place the mode switch in shutdown and start suppression pool cooling will result in containment failure for all reactor trips. If the procedural error rate is determined to be 10^{-5} then the plant damage frequency will be on the order of 10^{-5}, since the trip frequency is about 1 per year.

Note that the SSES assumption of strict procedural adherence applies only to the scram procedure and EOPs. These are the relatively few, simple, symptom based procedures which are flow-charted on large boards in the control room and on which the operators are regularly trained and tested. Correct use of these procedures is an integral part of NRC requalification testing and a necessary condition for maintenance of operators' licenses. Based on the experience at SSES, strict procedural adherence to EOPs is justified.

7. *Errors associated with procedure execution were generally modeled as equal to 0, 1, or on the order of the equipment failure rate. This is a non-standard method of modeling human error. Provide the bases for the approach taken in the IPE. Additionally, what impact does using more conventional human error probabilities have on: the core damage and containment failure frequencies, and the lessons learned from the IPE analysis?*

The basis for this is described in the attached paper presented at the Advanced Reactor Safety conference. A summary of the rationale follows. The HRA approach used by PP&L has a requirement, that the human error probability be on the same order as the equipment failure probability. If this is not the case, then modifications to either the plant, procedures or training program are performed until this goal is achieved. It is for this reason that Susquehanna's EOPs are, for many key operator actions, significantly different than the standard BWROG EOP. This point is illustrated by the following Table. The procedure steps identified in Table 5.1 of NUREG-1560 DRAFT were found to be important in many BWR IPEs. Modifications to the plant, procedures or training programs have taken place to reduce the significance of these steps at Susquehanna. These actions are described below.

Action	Modification to Address Step or Assumed Operator Error Probability
Perform Manual Depressurization	Removed ADS inhibit step from RPV Control procedure. Therefore plant blowdown is automatic.
Containment Venting	Assumed vent fails during ATWS since time is insufficient to perform vent, and after core damage due to the undesirability to release fission products. Assumed 10% failure of vent due to complexity of venting issue.
Aligning Containment Cooling	Assumed operator as reliable as equipment based upon the routine nature of action and the length of time available to initiate containment cooling.
Initiate Standby Liquid Control	Modifications to the plant design and procedures extend the time to initiate SLCS from several minutes to 40 to 60 minutes. See Sensitivity 1 in the attached NUREG 1150 sensitivity study.
Level Control During ATWS	Susquehanna's level control band is -60 to -161 (Top of Active Fuel is -161) with a target of -110 to -80, rather than the BWROG band of -161 to -191. See Sensitivity 1. Calculations demonstrate level control is not required during ATWS given our level control band. The recent NRC SER on ATWS stability changes urges other BWR utilities to adopt our level control band.
Align/Initiate Alternate Injection Systems	Modified the plant to facilitate alignment of alternate injection systems and reduced the probability of their requirement. See attached Sensitivity 2.
Recover Ultimate Heat Sink	Developed a procedure to use RWCU to perform decay heat removal. This application of RWCU is fully capable of removing decay heat while maintaining the plant within design parameters.
Inhibit ADS	Step removed from RPV control procedure, ATWS level is controlled above ADS initiation set point.
Miscalibration of Pressure Sensors	PP&L installed a bypass of the low pressure permissive that allows the control room operator to quickly bypass miscalibrated switches. See response to question 5c and sensitivity 4.
Initiate Iso Condenser	NA
Control Feedwater on Loss of IA	Feedwater lost on loss of IA at Susquehanna.

Action	Modification to Address Step or Assumed operator Error Probability
Manually Initiate Low Pressure Core Spray Systems	Automatically initiate on -129, no operator action required.
Provide Alternate Room Cooling for Loss of HVAC	HVAC calculations show that alternate cooling not necessary, beyond opening approximately 12 equipment cabinet doors.
Recover Injection Systems	No credit for recovery of injection systems, unless loss is caused by loss of AC power and AC power is recovered.

Note that a recent independent review of the SSES IPE suggests further sensitivities be performed to judge the impact of varying human error on IPE results. PP&L plans to perform these additional sensitivities.

8. *Other IPEs have identified operator actions that must be performed under moderate to high time requirements or stress. Human error probabilities are generally adjusted in these situations using performance shaping factors. Such performance shaping factors were not discussed in the Susquehanna IPE. Explain how performance shaping factors were accounted for in the Susquehanna IPE and what impact they have on: the core damage and containment failure frequency and the lessons learned from the IPE analysis.*

The evaluation of operator performance, and how it is impacted by factors such as moderate to high time requirements is discussed in the attached paper on human reliability presented at the recent Advanced Reactor Safety meeting. The general approach to this potential source of operator error is to modify the equipment, procedures, or training to eliminate or reduce the source of stress until the desired performance is achieved, rather than that to accept larger operator errors caused by time induced stress. Two examples of this process appropriate to this issue are provided in this paper. The first deals with SLCS initiation and the second deals with realignment of the HPCI suction source. Initially the failure rate for SLCS initiation in 2 minutes was estimated to be 0.65 based upon simulator measurements. After significant procedure modification, derived from requirements of the procedural and interface defense in depth criteria, the SLCS initiation error rate was reduced to the order of the SLCS equipment failure rate or less. In the second case, transfer of the HPCI suction source to the CST represents a potentially high stress action due to time and environmental considerations. The plant is being modified to remove this action.

In summary the approach to performance shaping factors was to identify those conditions which increase the operator's propensity towards error and to remove them. Procedure reviews, simulator exercises and operator interviews are used to identify these sources of error. For the IPE, performance shaping factors are essentially determined, and reduced to acceptable values, during the analysis. That is, instead of reporting the high performance shaping factor and leaving it as an artifact in the IPE results, the PP&L risk management process, including satisfaction of defense-in-depth, requires that the positive safety changes needed are both identified and incorporated into equipment, procedures, and training. The IPE results then reflect the desired operator performance, based on incorporating the required improvements.

9. *Provided a explanation (or basis) for the times assumed to be available to perform an action.*

The times available to perform a particular operator action were derived from transient calculations. The times are provided in Tables in Volume 4 of the IPE. As an example Table F.1-4 presents times various actions must be complete to avoid a particular event tree transition. The time required to perform time critical actions was obtained from simulator exercises or based upon interviews with operations.

10. *How were dependencies among human actions considered? In particular, discuss the influence of the accident and previous human failures on human performance.*

Complete dependence between functionally equivalent steps is assumed. As an example, if the operator fails to initiate SLCS, on the mistaken belief that no ATWS exists, then there is no reason to expect the operator to manually insert control rods either. Failure to recognize ATWS necessarily implies that the operator fails to recognize the position of the control rods since the ATWS symptom is, " a valid scram signal and more than one rod greater than 00". On the other hand, if the operator assigned to initiate SLCS is slow at executing the SLCS initiation procedure, this should not impact the ability of the operator assigned to drive control rods to successfully complete the procedure. These are diverse methods of accomplishing reactivity control and precede independently once the existence of ATWS is understood.

These issues of dependence and performance shaping factors are discussed in the attached paper on HRA. They have been dealt with using a combination of deterministic and statistical methods to answer the following questions:

1. Does the operator understand the status of the plant?

2. Given that the plant status is understood, does he know what to do?

3. Given 1 & 2, what are the odds of successful EOP execution?

Questions 1&2 concern the non-algorithmic mental process of understanding. Therefore, PP&L relies on deterministic methods to evaluate them. Question 3 is amenable to statistical analysis. Therefore, simulator exercises are used to develop probability of response as a function of time curves for time limited operator actions. Essentially we assume that if an operator fails to perform an action because he has no idea what the plant status is or what to do, core damage and containment failure are assured. We fully recognize that an operator may intend to perform a correct action, but be unable to do so because of lack of time. This method has been used to identify and resolve deficiencies in the plant operating procedures and the operator interface.

11. *Failure probabilities for actions related to ATWS were assigned values ranging from 0.0 to 10^{-1}. Provide the basis for the human error data used for the ATWS model as well as other operator actions credited in the Susquehanna IPE*

The basis for the human failure probability is presented in the attached paper presented at the recent Advanced Reactor Safety Conference.

12. *In the ATWS analysis, why did PP&L use and what is the impact of assuming the human error probability of 1.0 for manual scram and primary containment venting? What is the impact of these assumptions?*

There are always at least two diverse trip conditions present when a scram signal is generated. Therefore any failure to scram is dominated by either failure of the scram relays or undetected blockage of the scram discharge volume. A manual scram will not rectify either of these conditions. There is insufficient time to reliably execute the venting procedure during ATWS, and the containment vent is insufficient for removing the large amount of energy deposited during the at-power conditions of ATWS. Therefore no credit was taken for either of these actions.

13. *Equipment repair in a PRA is typically limited to the recovery of offsite power for which there is adequate experience in nuclear power plants as well as established procedures and training. The success of equipment repair depends on many important plant-specific factors such as the type of failure, time needed for diagnosis, time needed for repair (which may range from a very few hours to several days), crew completing tasks under different accident conditions, and crew availability. The IPE takes credit for repair or recovery of components other than diesel generators (e.g., pumps and valves) without providing an adequate basis.*

Recovery actions are assumed to be perfect. Discuss how these factors have been taken into consideration in the SSES IPE recovery actions.

Credit for repair of equipment was taken in the Susquehanna IPE only in those instances where such credit could be justified. Specific cases are: diesel generators, offsite circuits, TBCCW pumps and valves. Additionally, credit was taken for manual alignment of equipment. Credit for manual alignment of equipment is necessary when incorporating the information identified in Supplements 2 & 3 of GL 88-20 since many of the proposed actions are designed to mitigate SBO sequences. No recovery action was assumed to be perfect.

Diesel generator recovery data is discussed in Section C.2.2.3. This recovery data was developed from plant data and represents the probability of recovering a diesel as a function of time. Additionally, credit was taken for tie in of the E diesel into the appropriate bus. Aligning the E diesel is procedurally controlled and a routine occurrence at SSES. The alignment consists of 3 breaker manipulations. Emergency lighting is available in the diesel bays to perform the alignments.

Credit was taken for repair of the Turbine Building Closed Cooling Water (TBCCW) valve and pumps. The discussion appears on pages F-10 & F-11. An exponential repair model was used with a 12 hour delay before repair was assumed to begin. The Mean Time to Repair (MTTR) for the valves was obtained from NUREG/CR-3154. The MTTR for the pumps was obtained from NUREG/CR-2886.

Credit was taken for recovery from offsite circuits. The Loss of Offsite Power (LOOP) model is discussed on pages F-5 through F-9 and based upon NUREG-1032. Specific recovery curves for each cause of LOOP are based upon NUREG 1032 analysis and were obtained from the NRC during September of 1985.

Additionally, credit was taken for performing the accident management measures identified in Supplements 2 and 3 of GL 88-20. In many cases these accident management strategies require that the operator manually stroke a motor operated valve. The failure rate used for a manual valve, "fails to operate" was 1×10^{-4}. This value was obtained from NUREG/CR-2728.

Plant modifications have been performed to eliminate operator actions in which the operators success rate may be less than equipment performance. These modifications were required to satisfy the PP&L Severe Accident Defense in Depth Criteria (see the Integrated Risk Reduction Study). Two examples follow.

A valve with a hose attachment was installed onto each division of the RHRSW piping. With this valve installed, the field operator need only connect one end of a pre-staged fire hose to the fire hydrant and the other end of the hose to the valve installed on the RHRSW piping. Fire water can then be injected into either the RPV

or the drywell by manually opening four valves. Two field operators perform this task per ES-013-001. The operators are periodically trained and tested (Fire Protection System Crosstie To RHR, ES-013-001 From Control Room is JPM 9.13.001.101; Fire Protection System Crosstie To RHRSW, ES-013-001 At ESW Pumphouse, is JPM 9.13.001.002) on this evolution. The tie-in procedure takes approximately 1 hour to perform and operators have at least 5 hours available (SBO) from the initiation of the procedure until fire suppression system injection is required. The valves are tested as part of the ISI program and the cross connect piping has been x-rayed to verify no sludge buildup. Therefore, we are confident that the operator can successfully perform this evolution. Prior to this modification the operator had to remove a blind flange from the RHRSW piping and install a flange with a valve. The operator would have to either let the piping system drain, which could require many hours or install the flange with water flowing out the opening. Neither prospect had a high confidence for success. Therefore, in order to satisfy the Interface Defense in Depth Criteria this modification had to be performed.

The second example involves failure of the ECCS low pressure permissive. Failure of two RPV pressure channels (sensors, transmitters, switches, etc.) will prevent all 4 ECCS injection valves from opening. Should this event occur following a RPV blowdown, the control room operator would have to dispatch a field operator to a core spray valve gallery and manually open a core spray injection valve. This action must be performed within about 10 to 20 minutes depending on the event to prevent core damage and 2 to 3 hours to prevent vessel failure. The likelihood for success, especially at preventing core damage, is low. Therefore, a low pressure permissive bypass switch was installed in the control room to allow the operator to override a failed signal. Therefore, the likelihood for success is high.

14. *Discuss the connection between all procedural improvements made and the IPE analysis results.*

The previous answers address this question. Further, the impact of procedure changes on the core damage frequency cannot be segregated from physical improvements made to the plant since many of the hardware changes were made to enhance operator reliability. See the attached NUREG-1150 sensitivity study.

15. *Discuss why the risk achievement worths for some actions are in excess of 10,000.*

Three actions are identified in IPE Volume 6 which have risk achievement worths greater than 10,000. They are: entry into EOPs, throttle feedwater flow, and initiate suppression pool cooling. The consequences of failing to enter the EOPs is assumed *a priori* to be core damage and containment failure, as discussed above. Failing to throttle FW flow in ATWS creates conditions of high power and core thermal-hydraulic instability leading to core damage. Failure to initiate suppression pool cooling eventually causes containment failure. Note that all these are manual in

nature. Failure of the operating crew to perform these actions will result in either core damage or containment failure. In all other actions there are alternate means of accomplishing the function, thus these actions have lower RAWs.

Front-end Analysis

16. *The objective of Common Cause Failure (CCF) analysis is to examine the plant to identify and quantify CCF events that have occurred or have the potential to occur. Use of plant specific data identifies only those CCFs which have occurred to date. What analysis and evaluation has been performed to evaluate CCF that have the potential to occur? This should include CCF due to maintenance (including equipment calibration), design, and wearout. CCF for the emergency diesel generators, batteries and other DC power components, HPCI and RCIC valves, condensate and RHR pumps, and the ESW system needs to be included in the SSES IPE.*

    CCF is specifically included in the IPE where applicable (Reference Table 2.2.1 of Volume 6). The CCF contributions are derived from both actual data and from engineering judgment applied to other reported CCF results. EDG start data from SSES was examined for CCF contribution. Because approximately 3600 starts of these EDGs have been recorded, sufficient data exists at SSES to provide confidence in our CCF couple for the EDGs. No battery failures have occurred at SSES and for this reason industry data was examined to determine an appropriate CCF couple. Perhaps most important, where significant potential CCFs have been discovered, they have been eliminated (e.g. low pressure vessel injection permissive).

    Because the equipment used at SSES has been regularly monitored, maintained, tested and/or used during operation for approximately 20 years, the potential for CCF from design misapplication and wear-out is judged to be minor. Maintenance should also be sufficient to detect "new"/unanticipated failure modes (e.g. microbial deterioration of stagnant piping, zebra mussle intrusion, etc.) Thus, CCF is expected to be dominated by errors introduced during maintenance, that is, by inadvertent human error. Such errors may arise through inadequate time, training, or procedures. These types of errors were not quantified in the IPE. However, recent discussions with I&C maintenance personnel at SSES indicate that gross miscalibration errors of multiple equipment has never occurred at SSES. Regardless, a recent independent review of the SSES IPE recommended sensitivity studies be performed to evaluate the impact of various "human error" CCFs on the IPE results. PP&L plans to perform these additional sensitivity studies and report the results.

17. *Failure data of some important systems (e.g., HPCI, RCIC, and Fire Pump) are lower by a factor of 5 to 1000 compared to generic or NUREG/CR-4550 data. Provide a basis of how SSES procedures, practices, or equipment justify this difference. In addition, explain why some failure types are omitted (e.g., ESW failure to start or DG failure to*

run). Why do some blocks of similar components have the same failure rate as blocks of dissimilar components?

The general approach to equipment failure data is reviewed in IPE Vol. 3 Section C.1. Specific treatment of diesel generators is presented in Section C.2. DC power components are presented in Section C.7. PP&L developed equipment failure data from in-house records. Generic data was used when in-house records were insufficient to develop meaningful statistics. A comparison of Susquehanna specific data with other IPE sources shows that in some instances the Susquehanna data is lower, e.g., core spray pump fails to start; 6.4×10^{-4} for Susquehanna vs. 3.5×10^{-3} for generic sources and in some instances, the Susquehanna data is greater, e.g., Loss of DC Bus Initiator, 0.026 for Susquehanna and 0.0015 for Limerick.

The process used to develop the in-house data base is based upon the SAIC proprietary document "How to Formulate and Use a Probabilistic Safety Assessment Data Base", provided under contract to PP&L from SAIC. The actual data development was performed in collaboration with the Idaho National Engineering Laboratory as part of the "Integrated Risk Assessment Data Acquisition Program" which was sponsored by the NRC. PP&L provided failure records to INEL. INEL provided PP&L with a DBASE data base and program to compute unavailabilities.

Failure data for each reliability block is presented in Volume 2 by system. Both failure on demand and failure to run is presented. For standby systems, the failure probability used in the PRAC calculation was a combination of the failure on demand probability and the failure to run probability; $p = pd + \lambda T$. This is a standard method of combining probabilities. This may cause confusion, however, because entries for both standby and operating equipment appear in the PRAC input. A value of 6.9×10^{-3} was used for the probability of an ESW pump failure. Specifically, the ESW data can be found on A-181 and F-47. Using the above equation, the failure probability for ESW pump is computed below.

$$p = pd + \lambda T = 3.4 \times 10^{-3} + (4.8 \times 10^{-5}/\text{hr}) \times 72 \text{ hours} = 6.9 \times 10^{-3}$$

Diesel generator failure to start as well as run was accounted for in the IPE. All diesel failures are listed in Vol. 3 Section C.2 Table C.2-1. When computing the failure probability, all failures were assumed to occur at the time of the LOOP. This is conservative since it increases the probability of SBO.

18. *Provide justification for the following hypotheses used in the SSES IPE:*

- *ATWS calculations predict, in most cases, core damage but no core melt,*
- *Calculations of loss of HVAC in the control building appear to neglect the potential new failures that may come to play at elevated temperatures,*

- *Procedures to increase the suppression pool inventory to prolong the time until HCTL are based on in-house calculations but are counter to BWROG EPGs.*

Responses to each question are provided below:

- ATWS calculations predict, in most cases, core damage but no core melt

The most likely cause of damage during ATWS is due to large reactivity insertions not loss of cooling. There are a multitude of systems for achieving vessel water injection at SSES. Several of these (e.g. LPCI, CS) require vessel depressurization for success. However, vessel depressurization during ATWS puts the BWR core into thermal-hydraulic instability, with the potential for severe power spiking. This spiking can result in clad rupture and local melting, but not global core overheat and meltdown. Thus, ATWS fuel failures are expected to result more in gap release than overheat/melt release of fission products. The large insertions of reactivity during ATWS which could result in clad rupture and localized melting have been observed in calculations performed by General Electric using their TRAC-G code (NEDO-32164, Dec 1992). ATWS calculations performed using the PP&L SABRE code identified this potential form of core damage during the 1989 to 1990 time frame as part of the industry efforts to deal with reactor thermal-hydraulic instability issues. Therefore this form of damage was included in the Susquehanna IPE. We are unaware if other IPE's have accounted for this form of damage.

- Calculations of loss of HVAC in the control building appear to neglect the potential new failures that may come to play at elevated temperatures.

Calculations for loss of control structure HVAC show that the I&C equipment is expected to function adequately for the 72 hour mission time of the IPE. The temperature response to loss of HVAC is a gradual increase, not an immediate shock. Temperatures remain within their EQ envelopes for hours after the onset of any accident with loss of HVAC. It is recognized that increased temperatures may cause increased drift in instrumentation/monitoring electrical equipment. However, the equipment used to cope with accidents does not rely on fine control for successful function. For example, the equipment used for vessel injection and decay heat removal (pumps and valves) are typically started/opened and then require no further operator interaction. Large control bands are specified in procedures (e.g. vessel level is controlled between -161" and +54"). High pressure vessel control is limited by mechanical springs on SRVs, and low pressure after blowdown is not specified. Rod position indication is not required, other than control rods be fully inserted. Thus, while increased control structure temperatures may cause problems for routine power operation, such temperatures are not expected to significantly hinder response to reactor accidents.

- Procedures to increase the suppression pool inventory to prolong the time until HCTL are based on in-house calculations but are counter to BWROG EPGs.

Procedure to increase mass is consistent with the BWROG EPG Rev. 4. Mass addition to the containment is only limited by the Maximum Primary Containment Water Level Limit (MPCWLL) which is 49 feet at Susquehanna.

Back-end Analysis

19. *Following containment failure, the SSES IPE assumes that core damage can be prevented by utilizing the diesel driven fire pumps and opening the SRVs with the aid of the mobile diesel generator to supply DC power to reactor building components. Discuss the consideration that was given to the consequential damage to, for example, the HPCI, RCIC and /or core injection lines due to the containment failure and a hot suppression pool. Also, explain how operators can align the required equipment locally under these extreme conditions.*

Containment failure by itself is considered a form of plant damage in the Susquehanna IPE. Events that result in containment failure prior to core damage were not analyzed beyond containment failure. However Susquehanna procedures include steps to align systems external to the reactor building to provide for RPV injection prior to loss of containment integrity to enable the operator to respond to containment failure. These procedures were instituted due to the lack of confidence that reactor building equipment will reliably operate after loss of containment integrity.

Alignment of required equipment located in the reactor building is performed prior to containment failure. All reactor building actions are to be complete prior to the containment pressure exceeding 65 psig. Containment failure is not expected until 140 psig. Therefore the reactor building environment is not extreme when the actions are taking place. After containment failure, the only equipment required operable in the reactor building is piping and already open valves. HPCI and RCIC are not considered operable after containment failure. Because the injection source for the fire suppression water used for vessel injection is external to both the reactor building and containment, there is no impact on injection from a hot suppression pool.

20. *How were the impacts of uncertainties associated with in-vessel and ex-vessel recovery modeling included in the IPE results? Were sensitivities performed? Some vulnerabilities (due to uncertainties in code prediction, procedure descriptions, and operator actions) may not be identified from the PP&L methodology.*

In the 1986, the FAI company performed calculations for PP&L to address invessel debris recovery using the MAAP 3B computer code. These calculations demonstrate that given the MAAP modeling, vessel failure will occur 30 minutes after the initiating event.

PP&L first reported a conditional probability of vessel failure given core damage of 0.3 in our IDCORE IPE (1986). This value of 0.3 was derived by first determining a success

criterion for terminating the damage progression in vessel and assessing the probability of satisfying that criterion. The criterion established by FAI associates was to restore injection prior to the peak core node exceeding 3000 °F. However MAAP was designed for evaluating source terms rather than examining the in-vessel damage progression with the goal of terminating such damage.

Because of this inability on the part of MAAP, PP&L contacted the NRC via a letter on 8/3/87 concerning the possibility of terminating the damage progression in-vessel. In this regard PP&L received the BWRSAR code developed by ORNL under contract to the NRC. BWRSAR is specifically written to model the in-vessel core damage progression for the BWR. Based upon BWRSAR calculations (and corroborated by recent MELCOR calculations) core damage will be terminated in vessel provided that the bottom head doesn't dry out. This result is corroborated by experimental results reported in NUREG/CR-6133, "Fragmentation and Quench Behavior of Corium Melt Streams in Water" (See the response to question 21 for discussion of ex-vessel phenomena.)

Although no sensitivities were performed on the details of the core damage progression per se, that is, in terms of re-writing the computer coding, sensitivity analyses were in effect performed during the construction of the BWRSAR input. For the IPE, hundreds of BWRSAR runs were performed. Where results did not match expectations, input and coding details were examined, in consultation with the code developers, to understand the interactions and to improve the input data. Thus the BWRSAR calculations are judged to be the most accurate reflection of the BWR core damage progression available.

21. *How was the impact of major containment phenomena such as high-pressure melt ejection/direct containment heating and steam explosions included in the IPE results? Discuss the effects of ex-vessel debris coolability (e.g., documentation of the geometric details of the cavity configuration to justify assumptions of coolable debris bed).*

Phenomenological Bases for the Safety Function Success Criteria

The phenomenological bases for the success criteria for the core, vessel and containment are provided in Appendix E and in Volume 6. These success criteria address the most likely causes of core, vessel and containment failure. Effort was directed at deriving operational success criteria that could be used to judge the ability to prevent a particular form of damage. This process was seen as a very real way of dealing with uncertainty in the risk calculations and is based upon the following observation. The uncertainty associated with maintaining core cooling is much less than the uncertainty associated with terminating the damage progression in vessel. The uncertainty associated with terminating the damage progression in vessel is much less than the uncertainty associated with terminating the damage progression on the drywell floor. The uncertainty associated with terminating the damage progression on the drywell floor is much less than the uncertainty associated with estimating the released source term. Finally the uncertainty associated with estimating the released source term is much less than the uncertainty associated with

estimating the latent cancers from the released source term. In short the uncertainty in the risk calculation explodes as the damage progresses. For this reason PP&L chose to concentrate on using the IPE to identifying mitigating measures that terminate the damage progression as early as possible to the calculation uncertainty. A discussion of the specifics follows.

Fuel Coolant Interactions

When performing the Susquehanna IPE, PP&L believed that vessel or containment failure from thermal attack was far more likely than from fuel coolant interactions. Therefore this failure mode was not included in the containment evaluation. This view has been supported by recent NRC publications concerning Fuel Coolant Interactions, NUREG 1529 and the paper by Basu and Speis, "An Overview of Fuel-Coolant Interactions (FCI) Research at NRC." Based upon this paper one can conclude that FCI is of little or no significance to the overall risk from a nuclear power plant operation. Furthermore, this paper points out that, "Steam explosion was not observed in experiments involving prototypic melt composition at all water subcooling levels considered."

High Pressure Melt Ejection (HPME & DCH)

At the time the Susquehanna IPE was being performed, no analysis methods existed to allow for meaningful evaluation of either HPMC or DCH. We gained this understanding through our own investigations and our cooperative efforts with ORNL in the Mark II CPIP and other programs. This state of affairs was reflected as recently as November 21, 1994 by R. C. Schmidt and M. M. Pilch of Sandia in their Letter report to the NRC, "Assessment of the Importance of High Pressure Melt Ejection Events in the BWR Plants." They state:

Unfortunately, our current understanding and modeling capabilities for these important physical processes preclude deterministic calculation that can accurately predict the importance of HPME induced loads on BWR containment failure.

Additionally they point out,

A necessary precursor to containment failure caused by HPME loads is vessel failure at high pressure.

And finally,

the importance of the first part of the HPME equation, i.e., the conditional probability of vessel failure at high pressure given core damage, should not be overlooked. Gaining a clear understanding of why this conditional probability remains relatively high in current BWR PRA studies and finding ways to reduce it would provide an obvious benefit.

PP&L arrived at the same conclusions while performing the IPE. Based upon this conclusion, PP&L decided to reduce the likelihood of the necessary precursor to containment failure caused by HPME. This approach resulted in a real reduction in plant risk and uncertainty. Specifically, PP&L has:

Installed a self contained mobile generator that can continuously supply power to DC loads such as the SRVs, HPCI, RCIC and instrumentation.

Deviated from the generic BWROG EPGs by not inhibiting ADS in the RPV Control Procedure.

Additionally, the gas supply to the SRVs are backed by nitrogen bottles with a minimum 3 day supply. The bottles can be re-supplied if nitrogen is required beyond 3 days. Dose calculations have been performed to verify that the doses to the maintenance workers replacing the bottles are acceptable.

Based upon the modifications to the plant design and procedures and the availability of nitrogen to the SRV, the probability of containment failure and the associated uncertainty from HPME is considered remote.

Thermal attack of corium on containment structures

The Susquehanna Mark II containment is designed with a flat floor and no drains or downcomers in the inner pedestal region. The downcomer rises 18 inches above the drywell floor which allows a shallow pool on the drywell floor. The drywell floor in the Mark II containment represents a pressure boundary. For this reason, the drywell floor is steel lined.

At the time PP&L was performing the Susquehanna IPE, no analysis methods existed to allow for credible analysis of corium on the drywell floor for containments with Susquehanna's design. This state of affairs was also realized by the NRC contractors performing the Mark II CPIP calculations who state in NUREG/CR-5565 (5/91),

The choice of this design (deep-cavity design) was dictated by current CORCON and MELCOR code modeling limitations which preclude credible analyses of designs in which debris would be allowed to spread or flow outward from the in-pedestal to the ex-pedestal region of the drywell floor.

Additionally, CORCON could not model the steel plate on the drywell floor which prevents concrete degassing and subsequent vapor interaction with corium provided its integrity is maintained. Therefore, PP&L decided that if the liner plate could be preserved, CCI could be substantially reduced (substantial reduction in chemical energy) and containment failure from thermal attack could be prevented. In house calculations based upon work performed by Fred Moody of GE demonstrated that if the drywell floor

was flooded, the initial metallic pours would be quenched preventing metal water reaction. Continual supply of water from the drywell sprays was sufficient to remove both sensible and decay heat from the mixed oxides. Therefore, flooding the drywell floor and maintaining a continuous supply of water was determined to prevent containment failure from thermal attack. This result was "verified" given the above qualifications, using CORCON. Subsequently, Theofanous corroborated that the presence of water on the drywell floor precludes containment failure from thermal attack, NUREG/CR-5423 (1991) and NUREG/CR-6025 (1993). Recently, the NRC asked the BWROG to confirm that the recently issued AMGs provide for drywell spray to provide for corium quenching.

We have installed valves with threaded attachments on both RHRSW piping to improve the probability that water can be sprayed on the drywell floor during a severe accident. Additionally, two diesel fire pumps are installed at Susquehanna that automatically initiate on low header pressure. Finally, the Susquehanna Mark II design promotes debris coolability. Therefore, it is reasonable to expect that flooding the drywell floor will preclude containment failure from thermal attack of corium.

Summary

When we were performing the Susquehanna IPE, analytical methods did not exist to perform a credible evaluation of the containment challenges such as HPME, DCH, steam explosions, etc. PP&L decided to direct effort toward hardware and procedure changes that would reduce the probability that these containment challenges would occur. Subsequent research by the NRC has confirmed this judgment, that is, avoid vessel failure especially at high pressure and if vessel failure cannot be avoided, spray the drywell. While calculations may broaden the range of success, our plant is demonstrably safer having performed these modifications. This course of action is fully consistent with the GL 88-20 goal of identifying areas of poor containment performance and fixing them.

22. *Were source term calculations biased by the low core damage frequency (only one sequence meets the IPE screening criteria)?*

The sequences were chosen to represent a broad spectrum of accident types so that surrogate source term calculations would exist for any potential accident sequence. The goal was to select source terms spanning large early releases to late small releases with intermediate accidents in between.

23. *Explain the basis for the assumption of a containment failure step function at 140 psig instead of the distribution of failure pressures as requested in Generic Letter 88-20.*

The bases of the 140 psig is provided in Section C.4. A distribution of failure pressures will be included in a future sensitivity study and this information will be factored back into the results.

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SUSQUEHANNA
SOOR SUMMARY REPORT BY SYSTEM # - UNIT 2
SORTED BY UNIT, SYSTEM & SOOR #
P M I S

SYS	SOOR #	REV STAT	LER #	OCCUR DATE	CAUSE	DESCRIPTION	TREND TYPE	RESP GROUP
52	2-87-068	1		4/22/87	B	HPCI BOOSTER PUMP IMPELLER HAS NOT "DESIGNED" FOR SSES.		PENG
52	2-87-071	1		4/26/87	B	HPCI DISCHARGE CHECK VALVE WOULD NOT SEAT DURING SO-252-002	VLVFUN	PENG
52	2-87-078	1		5/02/87	B	LOUD NOISES HEARD IN CONTROL ROOM DURING HPCI RESTORATION	WATERH	PENG
52	2-88-008	1	LRR88001-0	1/27/88	B	HPCI LUBE OIL SAMPLE HAD EXCESSIVE AMOUNTS OF WATER PRESENT		MECH
52	2-88-013	1	LRR88001-0	2/03/88	X	HPCI LUBE OIL SAMPLE CONTAINED 7000 PPM WATER, LIMIT IS 5000 PPM		MENG
52	2-88-015	1		2/08/88	B	HPCI AUX OIL PUMP OPERATIONAL SHUTDOWN LED TO HPCI TRIP ALARM		PNSS
52	2-88-025	1		2/21/88	B	'HPCI OUT OF SERVICE' ALARM RECEIVED DUE TO ALARM RELAY FAILURE		ELEC
52	2-88-117	1		5/09/88	X	HPCI EXHAUST VACUUM BREAKER ISOLATION HV-2F075 DID NOT CLOSE		I&C
52	2-88-126	1		5/18/88	X	HPCI FV-25612 PISTON CUP SEALS DID NOT MEET EQ REQUIREMENTS.		PENS
52	2-88-133	1		5/27/88	B	HPCI STOP VALVE HYDRAULIC CYLINDER DOES NOT MEET EQ EL -80-II		PENS
52	2-88-135	1		5/27/88	B	HPCI SHAFT DRIVEN OIL PUMP COUPLING DOES NOT MATCH EQEL-80-II		PENS
52	2-88-136	1		5/27/88	X	HPCI REMOTE SERVO HYDRAULIC ACTUATOR HAS REPLACED WITH NON-EQEL-		CMPL
52	2-88-190	4		7/29/88	B	SPEED OSCILLATIONS OCCURED DURING HPCI OVERSPEED TEST TP-252-021		PNSS
52	2-88-191	7		7/29/88	B	HPCI ROOM FLOODED DUE TO CONDENSER CONDENSTE PUMP DISCHARGE LEAK		MECH
52	2-88-239	1		10/24/88	B	HPCI TROUBLE ALARM INVESTIGATION FOUND A FAILED RELAY IN 2D274		ELEC
52	2-88-247	1		11/04/88	X	SUSPECTED WATER HAMMER DURING HPCI QUICK START, SO-252-002.		PNSS
52	2-88-248	1		11/05/88	D	HPCI STEAM LINE HANGER DBB-214,H23 HAD TRAVEL STOPS INSTALLED.		MECH
52	2-89-091	1		8/05/89	X	HPCI STOP VALVE FV-25612 OPENED 20% AND THEN CLOSED DURING SURV		PNSS
52	2-89-092	4		8/10/89	X	HPCI STOP VALVE OPENDED TO 30%, CLOSED AND THEN OPENED FULLY		PNSS
52	2-89-137	7		9/21/89	X	PRESS TRANSMITTER INSTALLED ON HPCI PP DSCH NOT RECONNECTED		OPS
52	2-89-141	1		9/29/89	X	PCV-256-F035 FAILED AND LEAKING, FOUND DURING HPCI SUCTION HYDRO	VLVFUN	PENG
52	2-89-150	4		10/05/89	X	HYDROLASING OF FH F032A/B DRAINED INTO HPCI INJECTION LINE		MNT
	OTHER AFFECTED SYS: 45							
52	2-89-166	1		10/16/89	X	HPCI AUXILIARY OIL PUMP DID NOT START ON SEVERAL OCCASIONS	B/A/1/1/5	CMPL
52	2-89-201	1		11/08/89	B	HPCI INBOARD BYPASS, HV-255-F100 HAD DUAL INDICATION DURING SURV	VLVIND	CMPL
52	2-90-028	6	LRR 90-001-00	2/16/90	X	HPCI FLOW CONTROLLER FAILED TO PROVIDE STABLE FLOW CONTROL	INSTFC	PNSS
52	88-295	1	LER88022-0	11/04/88	A	INCORRECT FUSES WERE REMOVED, RESULTING IN UNIT 1 HPCI INOP	HUMAN	OPS

Example of Preinitiator Operator Error

TOTAL: 57

RA-B-NA-033
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ATTACHMENT B TO PLA-4631
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UNIT	SYS	PS	OOS DATE	OOS TIME	RETURN DATE	RETURN TIME	CRIT TIME	SD TIME	CRIT SDTIME	FAULT EXPOSURE TIME	PM	CM	SURV	MOD	REF FORM1	REF FORM2	REF FORM3	COMPONENT ID	SYS	DIV	CONAM	CNP	VALID	FAIL	DESCRIPTION
152	1		06/05/87	1125	06/05/87	1250	1.42	0.00	1.42	0.00	Y				A55369			1N012A/C	Y				N	N	HPCI O/S TO REPLACE BANANA JACKS FOR HPCI TURBINE EXHAUST PRESSURE SENSORS (PSH-1N012A&C).
152	1		06/15/87	0500	06/16/87	0700	26.00	0.00	26.00	0.00	Y				NONE			N/A	Y				N	N	HPCI TAKEN O/S FOR SCHEDULED EQ INSPECTION.
152	1		07/07/87	1310	07/07/87	1445	1.58	0.00	1.58	0.00		Y			SI-180-305			1N024B/D	Y				N	N	HPCI O/S FOR 1N024B&D SURVEILLANCE (54" TRIP).
152	1		08/05/87	0850	08/05/87	1115	2.42	0.00	2.42	0.00		Y			SO-152-004			N/A	Y				Y	N	HPCI OUT OF SERVICE FOR SURVEILLANCE (VALVE EXERCISING).
152	1		08/06/87	1428	08/06/87	1456	0.47	0.00	0.47	0.00		Y			SI-180-305			1N024B/D	Y				N	N	HPCI HIGH LEVEL CALIBRATION SI.
152	1		09/03/87	1406	09/03/87	1445	0.65	0.00	0.65	0.00		Y			SI-180-205			1N024B/D	Y				N	N	HPCI O/S FOR SI.
152	5		10/14/87	1400	11/05/87	1710	0.00	531.17	531.17	0.00		Y			A61216.			DBA-102-1A	Y				N	N	HPCI STEAM LINE WELD DBA-102-1-1A FAILED IN-SERVICE INSPECTION (ISI).
152	1		11/21/87	1700	11/21/87	1906	2.10	0.00	2.10	0.00		Y			SO-152-005			1F006	Y				N	N	HPCI O/S FOR SI WITH UNIT SHUTDOWN AND DEPRESSURIZED.
152	1		12/08/87	0845	12/08/87	0925	0.67	0.00	0.67	0.00		Y			SI-180-005			1N024B/D	Y				N	N	HPCI O/S FOR SI ON HIGH LEVEL TRIP.
152	1		12/10/87	0445	12/10/87	1015	5.50	0.00	5.50	0.00	Y				A52142			N/A	Y				N	N	PACKING ADJUSTMENT ON 1F001 AND 1F007 VALVES.
152	1		12/31/87	1010	12/31/87	1700	6.83	0.00	6.83	0.00	Y				S72417	A52147		1F007	Y				N	N	PACKING ADJUSTMENT ON 1F007 VALVE.
152	1		01/08/88	1030	01/08/88	1100	0.50	0.00	0.50	0.00		Y			SI-180-205			1N024B/D	Y				N	N	HPCI HIGH LEVEL TRIP DISABLED FOR SI.
152	1		01/20/88	1030	01/20/88	1305	2.58	0.00	2.58	0.00	Y				A68971			N/A			Y		N	N	HPCI TEMP CHANNEL O/S FOR MAINTENANCE.
152	1		02/07/88	0835	02/07/88	1000	1.42	0.00	1.42	0.00		Y			SI-180-305			1N024B/D	Y				N	N	HPCI HIGH LEVEL TRIP OUT OF SERVICE FOR SI.
152	1		02/07/88	1100	02/07/88	1247	1.78	0.00	1.78	0.00		Y			SI-180-305			1N024B/D	Y				N	N	HPCI HIGH LEVEL TRIP OUT OF SERVICE FOR SI.
152	1		02/19/88	0510	02/19/88	1500	9.83	0.00	9.83	0.00	Y				P21088	P80222	P71167	1P204	Y				N	N	HPCI INOP FOR MAINTENANCE PM'S - CALIBRATION CHECK ON PUMP TRIPS (INSPECTED TRIPS ONE AT A TIME).
152	1		02/19/88	0940	02/19/88	1245	3.08	0.00	3.08	0.00	Y				A71653			1F003	Y				N	N	PACKING ADJUSTMENT ON HPCI 1F003 VALVE - NO BLOCKING USED.
152	1		03/23/88	1460	03/30/88	0630	159.50	0.00	159.50	0.00			Y		S80409			1F100			Y		N	N	HPCI WARMUP VALVE (1F100) HAD DUEL INDICATION.
152	1		03/28/88	1250	03/28/88	1330	0.67	0.00	0.67	0.00		Y			SI-183-208			1N024B/D	Y				N	N	54" TRIP DISABLED FOR SURVEILLANCE.
152	1		04/06/88	1410	04/06/88	1445	0.58	0.00	0.58	0.00		Y			SI-180-205			1N024B/D	Y				N	N	HIGH LEVEL TRIP FOR HPCI O/S ONE AT A TIME FOR I&C SURVEILLANCE.
152	1		05/03/88	0830	05/03/88	1015	1.75	0.00	1.75	0.00		Y			SI-180-305			1N024B/D	Y				N	N	HPCI RX HIGH WATER LEVEL TRIP INOP FOR I&C SURV. - INOPS HPCI.
152	1		05/09/88	0110	05/09/88	0210	1.00	0.00	1.00	0.00		Y			SO-152-004			N/A	Y				Y	N	HPCI O/S FOR VALVE STROKE TIMING.
152	1		05/12/88	0850	05/12/88	1600	7.17	0.00	7.17	0.00		Y			SI-152-211			1N012D			Y		N	N	HPCI EXHAUST DIAPHRAGM PRESS SWITCH PSH-E41-N012D FAILED SURVEILLANCE ACCEPTANCE CRITERIA.
152	1		05/20/88	0125	05/20/88	0200	0.58	0.00	0.58	0.00	Y				NONE			1R203	Y				Y	Y	HPCI INVERTER POWER FAILURE - INVERTER OUT OF SERVICE.
152	1		06/01/88	1121	06/01/88	1155	0.57	0.00	0.57	0.00		Y			SI-180-205			1N024B/D	Y				N	N	HPCI SYSTEM LEVEL 8 ACTUATION INST. O/S WHILE PERFORMING SI.
152	1		06/30/88	1250	06/30/88	1320	0.50	0.00	0.50	0.00		Y			SI-180-205			1N024B/D	Y				N	N	HPCI HIGH LEVEL TRIP CHANNELS O/S ONE AT A TIME FOR SURV.
152	1		07/28/88	1125	07/28/88	1245	1.33	0.00	1.33	0.00		Y			SI-180-305			1N024B/D	Y				N	N	QUARTERLY CALIBRATION OF RX LEVEL CHANNELS.
152	1		08/13/88	0849	08/13/88	0925	0.60	0.00	0.60	0.00		Y			SI-180-205			1N024B/D	Y				N	N	O/S FOR SURVEILLANCE ON LIS-B21-1N024B&D (RX LEVEL 8).
152	1		08/19/88	0800	08/19/88	1515	7.25	0.00	7.25	0.00		Y			A82584	SO-152-002	P81947	N/A	Y				Y	N	SPEED CONTROLLER INOP FOR I&C AND CALIBRATION.
152	1		09/10/88	0915	09/10/88	0950	0.58	0.00	0.58	0.00		Y			SI-180-205			1N024B/D	Y				N	N	O/S FOR SURVEILLANCE ON LIS-B21-1N024B&D (RX LEVEL 8).
152	1		09/28/88	1645	09/28/88	2030	3.75	0.00	3.75	0.00	Y				SI-152-313	S87530		1N604D			Y		N	Y	TOSH-ES1-1N604D FAILED SURVEILLANCE (PIPE ROUTING AREA DELTA T).
152	1		10/08/88	0815	10/08/88	0940	1.42	0.00	1.42	0.00		Y			SI-180-205			1N024B/D	Y				N	N	HPCI O/S FOR 1N024B/D SURVEILLANCE (54" TURBINE TRIP).
152	1		11/01/88	1250	11/01/88	1430	1.67	0.00	1.67	0.00		Y			SI-180-305			1N024B/D	Y				N	N	HPCI O/S FOR 92 DAY CALIBRATION OF 1N024B/D.
152	1		11/04/88	0545	11/04/88	1015	4.50	0.00	4.50	0.00	Y				1-88-295			1N024B/D	Y				N	N	HPCI HIGH LEVEL TRIP (54") O/S DUE TO TRIP LOGIC FUSES REMOVED FOR A PERMIT.
152	1		11/08/88	1130	11/08/88	1345	2.25	0.00	2.25	0.00	Y				S84740	SO-152-004		1F001			Y		N	N	STEAM LEAK AT VALVE PACKING OF 1F001 (WA S74357 WILL BE USED TO REPACK VALVE THE WEEK OF 11/14).