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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards safety evaluation of station blackout.Plant can survive station blackout of limited duration w/appropriate operator actions.Hot functional tests proposed in lieu of station blackout test.

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JUN 15 1982

Mr. A. Schwencer, Chief
Licensing Branch No. 2
U.S..Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
STATION BLACKOUT SAFETY ANALYSIS AND TEST PLAN
ER 100450 FILE 841-12 Docket Nos. 50-387
PLA-1136 50-388

Dear Mr. Schwencer:

The attached report provides the results of a detailed evaluation of a station blackout at the Susquehanna plant. We believe that we now have a thorough understanding of the consequences of this event and a plan to mitigate them. This analysis clearly demonstrates that the Susquehanna plant can survive a station blackout of limited duration with appropriate operator actions. This analysis also clearly demonstrates that a station blackout test unnecessarily jeopardizes the plant and the public. A full scale test will definitely create unnecessary costs for PP&L since equipment will be exposed to environmental conditions close to qualification limits. Reanalysis and/or replacement will be required.

We propose to collect data from several startup tests and to perform additional hot functional tests to provide information to support our evaluation. These tests would be in lieu of the station blackout test. The attached report provides a discussion of the proposed testing, however this will not be implemented until NRC approval has been received.

Very truly yours,

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

DPM/mks

B001

Attachment

cc: R. Perch

SAFETY EVALUATION REPORT
STATION BLACKOUT
TESTING AT SSES

Prepared by the Plant Systems Analysis Section, based upon input and review provided by the Technical Section, SSES Plant Staff, and by the Nuclear Design Discipline Groups, Nuclear Plant Engineering.

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JUNE, 1982

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SAFETY EVALUATION REPORT
STATION BLACKOUT
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SUMMARY

This report presents the results of a PP&L evaluation of the feasibility of performing tests which simulate the loss of AC power at Susquehanna, with the objective of determining plant response under these conditions. PP&L has committed to perform a "Station Blackout" test to comply with an NRC requirement specified in the Safety Evaluation Report. This requirement arose from NRC's interpretation of additional testing needs under TMI-2 Action Plan Item I.G.1, "Training During Low Power Testing". This report is submitted to fulfill PP&L's obligations relative to Item I.G.1 as provided in SER Supplement #1. In addition, this report presents information generated in response to NRC Generic Letter 81-04, "Emergency Procedures and Training for Station Blackout Events".

PP&L believes that the regulatory intent for station blackout testing was to provide operator familiarization, to generate data for model verification, and to determine the capabilities and limitations of BWRs under blackout conditions. The major conclusion from this evaluation is that no single test can be formulated which adequately simulates station blackout conditions without undue risk to plant equipment and public health and safety.

Nuclear power plants are provided with multiple, highly reliable sources of AC power in order to prevent a station blackout condition. As a consequence, the probability of a blackout occurring is so low that this event has been justifiably excluded from the design basis for nuclear power plants. Nevertheless, PP&L is committed to achieving the objectives underlying the regulatory requirement for station blackout testing. Major efforts have been underway to determine the capabilities and limitations of the Susquehanna Plant if a station blackout were to occur. Detailed evaluations of the logic, instrumentation, and electrical design of the plant have been performed, together with extensive analysis of the transient response of the plant to a variety of station blackout scenarios. We have found that adequate core cooling can be provided for an extended period of time with appropriate operator action, even if an additional single failure is postulated. These results are summarized in this report. Emergency procedures and training programs are being developed based upon these evaluations to familiarize operators with plant response and necessary mitigating actions. The plant-specific simulator is being modified to simulate station blackout conditions. Finally, additional hot functional tests will be conducted which will yield data to support analytical predictions of plant response to station blackout.

PP&L believes that the above approach to the station blackout issue has resulted in a comprehensive understanding of plant response, from which

contingency plans which consider a broad range of event scenarios have been derived. These provide assurance of an adequate level of protection against the potential consequences of a station blackout event at SSES.

1.0 INTRODUCTION

This report presents an assessment of in-plant testing as a vehicle for determining the capabilities and limitations of SSES during station blackout. This section provides a brief history of the station blackout issue leading to the commitment of a test at Susquehanna, presents the objectives of that test, and briefly describes actions taken by PP&L to accomplish those objectives.

Section 2.0 presents the expected response of Susquehanna to a station blackout event, and discusses the capabilities and limitations of the plant to maintain a safe condition during a blackout.

Section 3.0 describes a test which was proposed within PP&L to study the response of various systems under simulated station blackout conditions to fulfill the test commitment. This section also presents the results of a later, detailed safety evaluation of the proposed test, and describes new planned testing which will yield data applicable to the station blackout condition.

Section 4.0 presents PP&L's conclusions and alternative actions relative to in-plant testing of station blackout.

1.1 History of the Station Blackout Issue

Station Blackout is defined as the loss of offsite power (LOOP) to a generating unit coincident with the failure of onsite emergency generators to deliver power to their respective Engineered Safeguard System buses. Such an event would deprive the unit of the power normally utilized to maintain safe, stable shutdown conditions and to mitigate design basis accidents.

The importance of power supply to a nuclear unit is recognized, and provisions are taken to assure adequate supplies in the design of nuclear sites. The Susquehanna Station is supplied with two electrically and physically independent sources of offsite power. Onsite AC power is supplied by four emergency diesel generators, with the design basis that any three provide enough power to bring one unit to a cold shutdown and to accommodate a LOCA in the other. In addition, electrical power is available from station storage batteries. When the plant is pressurized, steam generated by decay heat is available to drive the High Pressure Coolant Injection and Reactor Core Isolation Cooling pump turbines, providing water injection for reactor level control. However the plant was not designed to maintain safe conditions indefinitely with only battery power and decay heat steam available.

Concerns over the risk to the public from station blackout arise from concerns over the reliability of offsite and emergency onsite power

supplies. A number of nuclear plants have experienced LOOP events (LOOP is an expected operational occurrence and is considered in plant design), and failures of individual diesel generators to supply power upon demand have been reported. The NRC is currently assessing station blackout as a generic unresolved safety issue (A-44); completion of the review is projected for October 1982. In July, 1980, an Atomic Safety and Licensing Appeal Board, considering the reliability of the offsite power to the St. Lucie II plant, ruled that station blackout should be a "design basis event" (ALAB-603, 7/30/80). Although this assessment was later set aside by the Commission (CLI-81-12, 6/15/81), it focussed further attention by the NRC and the industry on station blackout.

In February, 1981, the Division of Licensing, Office of Nuclear Reactor Regulation, requested all licensees (except St. Lucie I & II) to assess their capability to mitigate a station blackout event and to generate emergency procedures and training programs for station blackout (Reference 1). In response, PP&L embarked on a three-phase station blackout evaluation program (Reference 2). Phase I would be a preliminary assessment resulting in the generation of Emergency Operating Procedures, initial modifications to the training simulator, and a training program prior to initial fuel load. Phase II would be a thorough evaluation of the entire issue as it relates to SSES. Portions of this evaluation are utilized in this report. In Phase III, Phase II results would be incorporated into Emergency Operating Procedures and training by the end of the first cycle. The incorporated Phase I assessment has been completed, and operating procedures and training programs based upon this assessment will be in place prior to initial fuel load. The evaluation of the station blackout event, including the development of models to predict plant response and to test various response strategies, is nearing completion. As a consequence of both these efforts, we have a comprehensive understanding of Susquehanna's response to station blackout events. In the process we have determined the capabilities of Susquehanna to maintain safe conditions for extended periods under station blackout conditions.

At approximately the same time as the generic request, NRC requested that a simulated loss of AC power test be performed at Susquehanna to satisfy the requirements of TMI Action Item I.G.1, "Training During Low Power Testing" (Reference 3). In the FSAR for Susquehanna, PP&L had previously submitted the BWR Owner's Group generic response for this item. In its review (Reference 4), NRC indicated that PP&L's response to I.G.1 was adequate with respect to supplemental training, but was inadequate with respect to additional testing, and again requested that PP&L commit to a simulated loss of all AC power test. PP&L subsequently committed (Reference 5) to performing a test which would simulate a loss of AC power condition for the reactor and containment systems, contingent upon (1) a continuing NRC

requirement, (2) a favorable safety evaluation, and (3) NRC approval. NRC subsequently agreed (Reference 6) that performance of the above "Station Blackout" test would satisfy their interpretation of the testing requirements of Item I.G.1. PP&L was requested to submit a detailed test procedure and safety evaluation for review four weeks prior to licensing.

1.2 NRC Objectives for the Station Blackout Test

TMI Task Action Plan Item I.G.1, "Training During Low Power Testing", requires applicants for a new operating license to define and to commit to a special low power test program (1) to provide meaningful technical information beyond that obtained in the normal startup test program and (2) to provide supplemental training. The training component of this requirement has been satisfied by the BWR Owner's Group generic response to Item I.G.1 (Reference 4). The general objectives and criteria for the testing requirement are as follows (References 3, 4, 6):

1. to provide meaningful technical information or data relative to plant responses during off-normal conditions, specifically information not provided by any of the tests described in Regulatory Guide 1.68, "Initial Test Programs";

2. to be equivalent in scope to the PWR Special Low Power Tests;

3. to pose no undue risk to public health and safety; and

4. to pose no undue risk to the plant.

The NRC has indicated that the testing requirements can be satisfied by a simulated loss of all AC power, or "station blackout" test (reference 3, 4). In addition to the above, the specific objectives of this test are as follows:

1. to familiarize operators with plant response to station blackout (Reference 3);

2. to provide data on the response of reactor vessel and containment parameters during a station blackout; NRC would use this data to evaluate the accuracy of analytical predictions and to determine whether simulator training is a satisfactory substitute for operator training during the test (Reference 4); and

3. to determine the limitations and capabilities of BWRs to maintain safe reactor and containment conditions in the event of a station blackout (References 3, 6).

1.3 Achievement of NRC Objectives at SSES

1.3.1 Limitations and Capabilities

As directed by the NRC in Reference 1, PP&L has been conducting studies to understand plant response to station blackout in order to generate procedures and operator training. As a result of these efforts, we have determined the capabilities and limitations of the Susquehanna BWR to maintain safe reactor and primary containment conditions in the event of a station blackout. Section 2.0 presents the results of our evaluations of plant response, including strategies developed to mitigate and to recover from a blackout, and discusses the plant's capability to accommodate additional component and system failures.

1.3.2 Verification of Predicted Plant Response

Support of analytical predictions with in-plant test data is desirable to assure that response strategies are based on a realistic conception of plant response. However, testing to achieve this objective must be consistent with the need to protect plant equipment and to minimize risk to public health and safety. At the outset both NRC and PP&L recognized that a full station blackout test should not be performed at an operating plant, and the test was limited to reactor and containment system response under simulated blackout conditions.

Further evaluation showed that limitations necessary to protect plant equipment would preclude the determination of reactor vessel and primary containment response through testing. However, special tests of particular components whose response can be safely tested have been defined and will be conducted during hot functional testing to verify the predicted performance of these components. Section 3.0 presents the results of our safety evaluations of various aspects of station blackout testing and describes startup tests and additional hot functional tests whose output will be used to support the analytical models used.

1.3.3 Operator Familiarization and Training

Because of the restrictions necessary to protect the plant and the public, a station blackout test will not be of benefit for familiarizing operators with plant response or for providing training in implementing mitigating procedures. Operator familiarization will be provided through a combination of classroom instruction, equipment walkdowns as appropriate and simulator training.

The SSES simulator is being modified to simulate station blackout conditions. Computer programs for the simulated plant have already been modified to include LOOP and the loss of all four emergency diesel generators. Those front panel indicators and recorders which would lose power directly under station blackout have been identified, and the simulator has been checked and modified as necessary to reflect these losses. Instrument loop studies for indicators required to achieve cold shutdown have been performed in conjunction with the preparation of our response to IE Bulletin 79-27, Loss of Non-Class 1E Instrumentation and Control Power during Operation. These will be utilized to identify indicators which fail under station blackout as a result of loss of power to instruments within a signal loop, and the results will be incorporated into the simulator programs. Similarly, transient analyses (such as those described in Section 2.0) and dynamic response studies of station blackout will be incorporated to the extent possible within the scope of simulation. With these changes, PP&L's plant specific operator training simulator will enable operators to experience the instrumentation blackout condition without placing the plant in jeopardy.

1.3.4 Summary

The capabilities and limitations of Susquehanna under station blackout are being determined through detailed operational and engineering reviews. The information generated by these reviews, which considered a range of event sequences, is presented in Section 2.0. Operator familiarization and training will be provided by a Training Program based upon Emergency Operating Procedures generated from these reviews. The training program, which will be in place by fuel load, will include simulator exercises, and the simulator is being modified to simulate station blackout conditions.

The primary application for a station blackout test would be to support analytical predictions. However, station blackout was not a design basis event for Susquehanna, and models predict that certain operating limits will be exceeded very quickly. Necessary restrictions on in-plant testing to protect plant equipment will limit the degree to which station blackout conditions can be duplicated. Rather than one station blackout test, Section 3.0 will describe testing from which data will be generated to support analytical predictions.

2.0 EVALUATIONS OF SSES RESPONSE TO STATION BLACKOUT

2.1 Discussion of Assumptions

2.1.1 Initial Conditions

The impact of a station blackout (SBO) on the plant is dependent on the operating state at the time of occurrence. In order to limit the magnitude of the analysis required, four initial conditions were defined which would bound all relevant modes of plant operation. These are discussed in the following.

1. Full Power Operation

The plant was assumed to be operating under normal conditions at 100% power with an equilibrium core. The initiating event was taken to be a LOOP, followed by a failure of all four of the emergency diesel AC generators to supply power. The immediate plant response would be similar to a LOOP transient. The reactor would scram and recirculation pumps would trip, Main Steam Isolation Valves (MSIVs) would close, and all systems and equipment not powered off station batteries would be lost. The primary containment would isolate as a consequence of (1) isolation signals generated by the LOOP event and (2) closure of air-operated, fail closed containment isolation valves due to loss of the instrument air and containment instrument gas compressors. Decay heat would be dissipated to the suppression pool through the Safety/Relief Valves (SRVs). Water injection to the reactor would be available from the steam-driven High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The following functions would be unavailable:

- drywell cooling;
- suppression pool cooling;
- cooling to the HPCI and RCIC equipment areas.

Under these conditions, the reactor could be maintained in a safe condition provided coolant injection remained available. However, decay heat would accumulate in the containment, causing the suppression pool and drywell temperatures to rise and pressurizing the containment. Temperatures would also rise in the HPCI and RCIC equipment areas due to the passage of reactor steam through the turbines. Suppression pool level would rise due to thermal expansion and due to the accumulation of reactor injection

water from the Condensate Storage Tank (CST) in the pool as condensed steam. If reactor injection water were lost (a potential consequence of degrading environmental conditions or loss of support equipment), the core could be uncovered with potential fuel damage.

Power operation was considered to be the limiting initial condition for the station blackout event. The critical systems for plant safety were identified as the reactor coolant system, primary containment, and the HPCI and RCIC systems. The responses of these systems (hereafter referred to as "the plant") were analyzed intensively to develop strategies to mitigate the event and to determine the effects of postulated equipment failures. The analytical approach utilized was as follows:

- The initial plant response (first twenty seconds) was assumed to be identical to the LOOP transient presented in Reference (9).

- The transient response after the initial 20 second period was determined using a model which simulated plant conditions with the reactor shutdown and isolated, with an option to depressurize the reactor by holding open an SRV. This model permitted the simulation of various equipment failure events or operator response actions to determine the effect on the overall plant response. The model was constructed to provide a best estimate of transient response for the input parameters provided.

- The decay heat curve utilized in the model was computed using the ANS 5.1 Decay Power Standard (1979) for an equilibrium core.

- The fuel relaxation energy and the sensible heat released from the reactor vessel and internals during depressurization were considered in addition to decay heat in determining depressurization rate, reactor makeup requirements and suppression pool heat loads.

These assumptions resulted in conservative best estimates of plant response to a station blackout occurring during power operation.

2. <u>Shutdown Cooling</u>

The plant was assumed to be shutdown with decay heat being

removed by the Residual Heat Removal (RHR) system operating in the shutdown cooling mode. This mode of operation is possible only when reactor pressure is below 113 psia. As a consequence of station blackout, core cooling would be lost and decay heat could cause the RPV to re-pressurize. If no corrective actions were taken, the RPV water inventory would boil-off through the SRVs to the suppression pool, eventually leading to uncovering of the core.

Station blackout was assumed to occur four hours after shutdown, because four hours is the shortest possible interval between shutdown and the initiation of the shutdown cooling mode. The same decay heat curve as described above for the 100% power case was used. The RPV water inventory was assumed to be at normal water level. These assumptions again provided an upper bound estimate of reactor vessel heat-up, repressurization and boil-off rates. As will be shown later, this condition is much less demanding than a station blackout at power because the initial high decay heat rates and the thermal energy releases from fuel relaxation and from RPV depressurization have already been removed from the containment prior to SBO.

3. Refueling

a. Head Removed: Reactor Well Not Flooded.

The plant was assumed to be shutdown, cooled down, and depressurized with decay heat being removed by the RHR system operating in the shutdown cooling mode. At the initiation of a station blackout, the reactor vessel head had been removed but the reactor well had not yet been flooded. As a consequence, core cooling would be lost and decay heat would cause the water in the RPV to heat-up and boil-off, eventually leading to uncovering of the core if not mitigated.

Station blackout was assumed to occur ten hours after the plant had been shutdown by a scram from 100% power. This is the minimum time at which RPV head removal is possible. Reference (7) indicates that the average time from shutdown for head removal is forty-eight hours. The reactor vessel was assumed to be flooded to a level above the steam dryers. Heatup of the reactor water inventory to the boiling point would require a minimum of thirty minutes. Boil-off to the

top of active fuel would require an additional five hours with no make-up.

Because of the greatly reduced decay heat rate, this condition is much less demanding than the shutdown cooling condition. Ample time is available for recovery of AC power or for provision of make-up water to replace boil-off. As described later, make-up water can be provided to the RPV from the Fire Protection Water Supply System if necessary. The plant could be maintained in a safe condition for an extremely long time under these conditions.

This condition was not analyzed any further.

b. Reactor Well Flooded

Station blackout was assumed to occur during the refueling process with the reactor well flooded. Any fuel bundles in transit or up on hoists would be locked in position by the loss of power to the refueling bridge and trolley, and by brakes on the cable drums which automatically engage on loss of power. These provisions would preclude fuel handling accidents as a consequence of station blackout.

During refueling, decay heat removal would be provided by the Fuel Pool Cooling System, perhaps supplemented by the RHR System. As a consequence of station blackout, pool cooling is lost, and decay heat will cause pool temperatures to increase.

This condition is much less severe than that prior to flooding, because of the much larger inventory of water available to absorb decay heat. Heat up of the water inventory in the vessel and reactor well and eventual boil off would require a very long period of time. Make up, if needed, could be provided by the Fire Protection Water Supply System.

This is not a limiting case relative to ability to maintain the plant in a safe condition, and was not analyzed any further.

4. Spent Fuel Pool Cooling

During a station blackout, cooling to the spent fuel pools would be lost. Decay heat from the spent fuel inventory

could raise the pools to boiling and eventually result in loss of pool inventory.

The consequences of the loss of Spent Fuel Pool cooling have already been examined relative to the potential loss of the cooling systems in seismic events (Reference 8). The conservative results showed that the pools would not boil until at least 25 hours after the loss of cooling. If cooling cannot be restored before the pool boils, then make-up water from the Fire Protection Water Supply System can be added to replace boil-off.

This condition was not considered limiting in terms of ability to maintain safe conditions and therefore was not considered further.

2.1.2 Equipment Availability, Station Blackout During Power Operation

Only equipment which does not require or depend upon normal or emergency AC power can be utilized to maintain the plant in a safe condition after occurrence of SBO. Evaluation of the plant electrical, instrumentation, and logic systems has demonstrated that equipment adequate to maintain the plant in a safe condition for an extended period of time will be available. This equipment includes:

1. Reactor vessel level instrumentation.

   Narrow range indication would be available in the control room at the Standby Information Panel; narrow range, wide range, and fuel zone indications would be available at local reactor building instrument racks.

2. Reactor pressure instrumentation.

   Indication would be available at the Emergency Core Cooling Control Panel in the control room and at local reactor building instrument racks.

3. Suppression pool temperature instrumentation.

   Although the transmitters and indicators for the SPOTMOS (Suppression Pool Temperature Monitoring System) lose power, detector output can be monitored with a battery-powered meter and temperature determined.

4. Suppression pool level instrumentation.

Indication would be available at local reactor building instrument racks.

5. HPCI, injection mode from CST or Suppression Pool as well as test mode, at reactor vessel pressures in excess of the HPCI turbine low pressure trip setpoint of 115 psia.

6. RCIC, same operating modes as HPCI, at reactor vessel pressures greater than 65 psia. For analytical purposes, RCIC was conservatively assumed to shutdown at 75 psia.

   Both HPCI and RCIC would be monitored and controlled from the Emergency Core Cooling Control Panel in the control room.

7. Diesel Drive Fire Pump and Fire Protection Water Supply System.

8. SRVs in the pressure relief mode (until accumulator depletion) and in the safety relief mode.

   Containment instrument gas supply to the relief mode accumulators is lost due to shutdown of the compressor on LOOP. It was conservatively assumed that the residual pressure in the accumulators would provide only one lift at the relief mode set pressure, and that subsequent automatic lifts would be self-actuated (safety mode) at the higher safety set pressures.

   Analysis indicated that four of the five SRV groups would have opened at their relief mode setpoints within three seconds of the LOOP. To simplify the station blackout response analysis, it was assumed that SRVs would lift automatically only in the safety mode after twenty seconds. In actuality, the remaining Group 5 SRVs would lift at least once in relief mode, and sufficient accumulator pressure could exist to lift all the SRVs once more in relief mode, prior to any of the SRVs lifting in safety mode. The effect of this on the analysis results was not considered significant.

9. SRVs individually actuated in the Automatic Depressurization System (ADS) mode via keylock switches from panels in the upper and lower relay rooms.

   SRV actuation in the ADS mode remains available since the external gas supply to the pneumatic actuators for the six SRVs assigned to ADS is not isolated. Upon loss of the containment instrument gas compressor, the external supply

is provided by a bank of high pressure nitrogen cylinders which may be replaced as needed. In addition, these SRVs are equipped with large capacity accumulators inside containment which, in the remote event of loss of the external supply, provide sufficient capacity for SRV actuation to mitigate station blackout.

10. 250 VDC batteries (2) which provide power to DC-operated valves, pumps, and logic in the HPCI and RCIC systems.

    The 250 V batteries are expected to last a minimum of 24 hours under station blackout conditions provided unnecessary loads are stripped per response procedures. They are not considered limiting for station blackout mitigation.

11. 125 VDC batteries (4) which provide power for emergency lighting and SRV actuation in relief and ADS modes among other functions.

    The 125 V batteries are expected to last a minimum of 6 hours under station blackout conditions, provided unnecessary loads are stripped per response procedures and substantial reductions are made in emergency lighting loads (the major load on these batteries). Alternatively, battery life could be extended by transferring some of the emergency lighting loads to the 250V batteries, which would require center tapping of the batteries and temporary cables.

In order to demonstrate the capability of the plant to withstand the SBO event failure of some of this equipment was also considered. The failures considered included:

1. HPCI failure
2. RCIC failure
3. Stuck Open Relief Valve (SORV)
4. SORV and HPCI failure
5. SORV and RCIC failure
6. HPCI and RCIC failure

The capability of the plant in terms of how long a safe condition could be maintained is discussed in Section 2.4.

Loss of compressed gas for ADS was not considered due to the capacity, redundancy, and provision for external resupply provided in the design of this system. Loss of DC power (prior to the expected depletion of the batteries) was not considered on the basis that cross connection of the DC distribution centers could be quickly accomplished. The expected long time to depletion of these systems in combination with the expectation that back up could be quickly achieved should eliminate loss of a DC system as a significant contributor to severe consequences resulting from SBO.

2.2 Plant Limits for Safety, Station Blackout during Power Operation

While the plant may be maintained in a safe condition for some period of time after occurrence of SBO, the inability to remove decay heat from the containment does not permit a stable condition of the plant to be achieved as long as SBO conditions exist. After a finite period of time some plant parameter will exceed its limiting value. The parameters which must be considered in this regard are:

1. Reactor water level
2. Suppression pool level
3. Reactor pressure
4. Containment pressure
5. Suppression pool temperature
6. Drywell temperature
7. HPCI/RCIC room temperature
8. Control room temperature

The limits on each are presented individually.

2.2.1 Reactor Water Level

The success criterion for reactor water level is that it be maintained above the top of active fuel (TAF), but no higher than level 8 (L8). This criterion is somewhat conservative in that a level transient that drops below TAF by only a limited distance and for a short time would not actually indicate absence of adequate two phase cooling of the fuel rods.

Exceeding L8 is considered unacceptable on the basis of the potential for introducing liquid phase into the steam lines. This would adversely affect the operation of the HPCI/RCIC turbines and possibly of the SRVs. This limit is much less likely to be violated than TAF.

The criterion that reactor vessel level be maintained above TAF is the primary criterion for the judging safety of the plant. It assures a high level of fuel integrity and therefore a

relatively low level of fission product release from the fuel. These low release levels would not represent a significant threat to public safety even if released to the environment.

2.2.2 Suppression Pool Level

During a blackout, suppression pool level will rise due to (1) expansion from heatup, and (2) addition of condensed steam from the reactor, less water returned to the reactor by HPCI or RCIC when pumping from the pool. The upper limits were set at the elevation of the bottom of the HPCI and RCIC turbine steam exhaust lines where they pass through the wet well wall, 25'6" and 25'11" respectively. These limits were selected to avoid flooding of these lines to the check valves (located just outside containment), which would occur if the turbines were shutdown. Failure of the turbines upon restart due to water in the casings could occur if the check valve were to leak or if water were to flood the line upon initial opening of the check, prior to the development of sufficient steam velocity to clear the line. As long as the turbines are operating, steam velocity will be sufficient to keep the lines clear even if these limits are exceeded. However, even though higher pool levels may be acceptable (assuming continuous turbine operation), response strategies have been developed to maintain pool level below these limits.

2.2.3 Reactor Pressure

Overpressure protection is provided by the SRVs operating in first the pressure relief mode and later in the safety mode if necessary.

Following reactor depressurization, pressure should be maintained between 115 and 200 psia in order to avoid shutdown of the HPCI turbine on low steam pressure. In the event that RCIC only is operating, the low pressure limit could be reduced to 65 psia.

Reducing reactor pressure below 200 psia is not essential to maintaining the plant in a safe condition. Depressurization will reduce the probability of SORV (because fewer valve lifts would be required for decay heat rejection) and will slow the heatup of the drywell. This would extend the time over which equipment would not be exposed to excessive drywell temperatures. With a lower risk of equipment failure due to thermal degradation, there is greater likelihood that safe conditions will be maintained.

Failure to maintain reactor pressure above the minimum needed for RCIC or HPCI operation is not an essential requirement (except during the first hour after SBO) in that the Diesel Driven Fire Pump would be available to supply makeup water to the reactor vessel, if necessary. This source would be utilized only if HPCI, RCIC, or AC power had not been restored by the time the vessel level dropped to TAF. Nevertheless, maintaining RCIC or HPCI available does increase the likelihood of maintaining safe conditions throughout an SBO event.

2.2.4 Containment Pressure

The design value for containment pressure, 53 psig, has been selected as the limiting condition for station blackout. Containment pressure remained below this limit for most of the transient cases considered. However, this limit could be reached if (1) the reactor is not depressurized and suppression pool water temperature reaches its design limit of 220°F (Section 2.2.5), or (2) if suppression pool temperature greatly exceeds its design limit.

2.2.5 Suppression Pool Temperature

A bulk average suppression pool temperature limit of 220°F was selected for SBO. The following phenomena were considered in selecting this value.

1. SRV discharge air clearing loads.
2. SRV condensing loads.
3. Structural design basis for suppression pool structures and components.

The first of these is not expected to be limiting in that the reactor pressure will have been reduced to less than 200 psia prior to the suppression pool bulk average temperature exceeding 150°F. The second is not expected to be limiting in that the amount of sub-cooling will be about 60 F° at 220°F and the test data reported in the SSES Design Assessment Report shows that smooth condensation is expected for this condition.

Subsequent to reactor depressurization localized high temperatures in the vicinity of the quenchers are not expected, since the steam discharge and heating of the pool occurs over a several hour period. Temperature non-uniformities are expected to be in the form of thermal stratification, resulting from the rise of water heated by steam condensation to the surface of the

pool. Replenishment of this water with cooler pool water should result in a smooth condensation process.

The structural design of the suppression pool structures and components has considered pool temperatures up to 220°F.

On this basis we expect no adverse consequences from pool temperatures up to 220°F for the event sequences considered for SBO.

2.2.6 Drywell Temperature

The drywell temperature limit has been selected based upon environmental qualification limits for the most critical safety-related item in the drywell for SBO, the SRV actuators and associated components. Accident environment limits have been used to determine acceptability using a time at temperature approach. This results in the following limits.

1. Below 150°F - indefinite
2. Between 150°F and 200°F - 2376 hours (99 days)
3. Between 200°F and 250°F - 18 hours
4. Between 250°F and 320°F - 3 hours
5. Between 320°F and 340°F - 3 hours

It is believed unlikely that these time at temperature limits will be exceeded by the mean drywell temperature. It is likely, however, that the temperatures reached in the drywell will cause degradation to some non-essential components which would as a result require immediate or premature replacement. There is also concern that local temperatures could exceed the mean temperature.

2.2.7 HPCI/RCIC Room Temperatures

The design limits for HPCI and RCIC turbine and pump room equipment is 104°F normal maximum and 148°F for up to twelve hours under accident conditions. Room temperatures are believed to rise only slowly and the equipment should be capable of extended operation under the conditions anticipated. Further evaluation to define these limits will be required.

Violation of the HPCI/RCIC room temperature limits does not represent failure to maintain the plant in a safe condition. First, violation of the temperature limits does not necessarily result in immediate loss of the equipment. Second, the existing water inventory in the RPV can maintain fuel integrity for a significant period of time. Finally, a backup source of makeup water is available from the Fire Protection Water Supply System.

2.2.8 Control Room Temperatures

No hard limit for control room temperature has been established for evaluation of the SBO event. The SBO event will result in loss of power to all equipment in the control room except that supplied from DC or uninterruptible power sources and much of this will be switched out very early in the transient to conserve DC power supplies. The major heat sources will therefore be the sensible heat stored in control room equipment at equilibrium operating temperatures and the biological heating by the room occupants.

Control room temperature and humidity will be moderated by establishing natural circulation pathways to the control structure stairwells. This will be done by opening the access doors to the control room, providing an opening to the Technical Support Center at the top of the control room, and opening the access doors to the Technical Support Center.

2.3 Operational Response Strategy for SBO

In reviewing the preceding discussion on limits for the important plant parameters in the SBO event, it becomes clear that consideration should be given to the consequences of additional equipment failure during the period when AC power is not available. As one example, a SORV would eventually depressurize the vessel and cause loss of both HPCI and RCIC which in turn could cause eventual core uncovery due to lack of makeup water. As another example, avoidance of overfilling the vessel eventually requires intermittent operation of HPCI and RCIC in the injection mode, but repeated trips of these systems could result in failure to restart when needed after some number of cycles.

The SBO event is considered to be extremely improbable; the concurrent loss of reliable equipment represents an even more improbable condition of the plant. Nevertheless we have selected response strategies which reduce the probability of such failures and which provide mitigating actions to avoid severe consequences from such failures. The response strategies have been developed from the following guidelines.

1. When makeup flow is available to the reactor vessel, the reactor vessel is depressurized and maintained at low pressure to limit drywell temperatures.

2. The number of SRV lifts is minimized to reduce the probability of SORV.

3. Both HPCI and RCIC systems are initiated and maintained in operation until the steam generation rate cannot sustain their continuous operation.

4. Reactor vessel water level is maintained at normal level whenever possible or is recovered to normal level as quickly as possible.

5. During the early part of this transient HPCI and RCIC suctions are switched to the suppression pool until depressurization is complete or until the suppression pool temperature exceeds 135°F This is done to reduce pool level rise early in the scenario due to the accumulation of condensed steam from the reactor vessel. This in turn extends the time during which pool level can be kept below the limits given in Section 2.2.2, and preserves cooler CST water for use later in the scenario.

   The temperature of 135°F was selected to assure adequate cooling of the HPCI lube oil (which is cooled by pumped fluid). However, temperatures up to 170°F may be allowed if necessary to assure suppression pool level control.

6. Immediate action is taken to make a connection from the Fire Protection System to the Residual Heat Removal (RHR) Service Water System at the service water pumphouse. This allows injection of fire water into the RHR system via a cross-connect with the RHR Service Water System. This provides a path for use of Fire Protection Water for reactor vessel makeup or drywell spray if needed.

7. All DC loads not required to mitigate the station blackout event are stripped from the DC supplies to extend the life of the DC power supplies.

8. Immediate action is taken to provide for cross connection of DC busses to accommodate potential premature depletion of one of the DC systems.

9. Immediate action is taken to strip the AC busses to assure acceptability of the initial loading when AC power is re-established.

The overall objective of the response procedures is to maximize the period of time the plant can be maintained in a safe condition. Maintenance of core coverage is the top priority consideration in this regard. As long as adequate core coverage is maintained the plant is considered to be in a safe condition.

2.4 Evaluation of Plant Response

Plant response has been determined analytically for the initial conditions of full power operation and shutdown cooling, utilizing the strategy described in Section 2.3 to maximize the time before violating the plant limits for safety given in Section 2.2. Various combinations of subsequent equipment failures as described in Section 2.1.2 were postulated. The cases considered are discussed individually below. Each discussion presents the response strategy appropriate for the plant condition and equipment availability, describes the plant response, and summarizes the analytical results.

2.4.1 SBO At Full Power With No Additional Failures

The first twenty seconds (approximately) of plant response would be the same as for a LOOP transient. Reactor scram and MSIV closure signals would be generated immediately due to loss of the Reactor Protective System (RPS) power. The recirculation pump drive motor breakers would open on LOOP, resulting in shutdown of recirculation. Within two seconds the MSIVs would begin to close. Within three seconds the first four groups of SRVs would have opened at their relief mode set pressures. All of these would have reclosed by about nineteen seconds into the transient. Because of the modeling assumption that all relief mode accumulators would have been depleted by this time, the Group 1 SRVs (lowest set pressure group) would not reopen until about ninety seconds, at the higher safety settings.

Reactor water level during this time would drop rapidly due primarily to void collapse in the core and steam separator risers following the scram and partly due to steam inventory loss which is not completely replaced by feedwater flow in the first seconds of the transient. Reactor water level would reach Level 2 at about two minutes into the transient, initiating HPCI and RCIC. Reactor water level would be restored to normal levels at about four minutes by HPCI and RCIC with water from the CST.

The first operator mitigating action would be to adjust HPCI and RCIC to minimum flow to maintain a stable water level at about 4 minutes into the transient. This action is not essential since these systems would automatically trip at Level 8 to prevent overfill. It would be preferred to avoid trip of either turbine.

Within the first ten minutes of the transient the operators would (1) shift HPCI and RCIC suction from the CST to the suppression pool, and (2) initiate depressurization. As discussed in Section 2.3, this suction shift would be made to

minimize pool level rise during depressurization. The operators would depressurize by opening one of the SRVs in ADS mode (presuming all relief mode accumulators had been depleted) and by adjusting HPCI and RCIC as necessary to maintain normal water level. The reactor would be depressurized into the minimum pressure control range for HPCI or RCIC as appropriate.

Operation at low reactor pressure, with pressure controlled by using HPCI and RCIC in test mode and SRV lifts as necessary, and reactor vessel level controlled with HPCI (early) and RCIC (later), would continue until restoration of power. HPCI and RCIC pump suctions would be shifted back to the CST when the Suppression Pool bulk average temperature reached 135°F in order to provide adequate cooling for the turbine lube oil. In the event that steam pressure dropped below the minimum necessary for RCIC operation, injection water could be supplied from the Fire Protection Water Supply System if necessary.

Figure 2-1 shows the decay heat generation rate and also the rate of release of energy to the Suppression Pool as a function of time from shutdown, assuming that an SRV was opened at ten minutes to initiate depressurization. Figure 2-2 shows the response of reactor pressure. The minimum pressure control range (200-115 psia) would be reached at about 40 minutes after the start of depressurization.

Figures 2-3, 2-4, 2-5 and 2-6 show the response of drywell temperature, suppression pool temperature, containment pressure, and HPCI/RCIC turbine room temperatures respectively during the event. The first plant safety limit violated would be the suppression pool temperature limit of 220°F. This would occur at just over eight hours.

It is evident that core coverage could be maintained for several additional hours, although suppression pool temperature and later containment pressure would exceed design values.

2.4.2 SBO at Full Power With HPCI Failure

The strategy and plant response for this case would be essentially the same as that described above. The unavailability of HPCI would cause a reactor water level transient that reaches a minimum slightly above top of active fuel at about 40 minutes, with full recovery to normal level at about 2.5 hours. Except for concern over the period during which water level is low, loss of HPCI would be preferred to loss of RCIC (Section 2.4.3) in an extended blackout because level control later in the transient, when makeup requirements are reduced, is much easier with RCIC.

2.4.3 SBO at Full Power With RCIC Failure

The strategy and plant response would again be similar to the base case above. Frequent switching of HPCI between injection and test mode would be required to provide reactor vessel level and pressure control. Eventually, shutdown of the HPCI turbine would be required at about four hours due to low reactor pressure. Subsequent makeup would require restart of HPCI after the reactor vessel had repressurized sufficiently, or injection from the Fire Protection Water Supply System.

2.4.4 SBO at Full Power With SORV

With a SORV, a depressurization similar to that which would be performed manually would occur. With either HPCI or RCIC (or both) available, the strategy during the early stages of the event would be the same as without SORV. As pressure approached the minima for HPCI and/or RCIC operation, the reactor vessel level would be brought to Level 8 so as to maximize vessel inventory.

Following loss of the turbines, reactor level would drop due to decay heat boil-off through the open SRV. In the event that reactor vessel level reached Top of Active Fuel with no other sources of injection available, the operator would inject water from the Fire Protection Water Supply System to re-establish or to control level. Depressurization to about 70 psia will permit the Diesel Driven Fire Pump to inject enough water to replace boil-off at a decay heat rate of 1% power, which is reached at about two hours.

In the analysis, SORV was assumed to occur on the first SRV lift, resulting in a slightly earlier depressurization. Reactor water level would be maintained until reactor pressure drops below 75 psia at about 6 hours. Injection of water from the fire main would be possible when the reactor pressure drops below 100 psia, which occurs at about 5.5 hours. Reactor water level could then be maintained indefinitely, but the suppression pool temperature limit would be reached at about 7 hours, or about an hour earlier than in the base case, Section 2.4.1.

2.4.5 SBO at Full Power With SORV and HPCI Failure

Plant response would be identical to that without SORV except that when the reactor pressure falls below 75 psia, the RCIC system is assumed to shut down due to low steam pressure. In the event that no other injection sources become available, make-up would be provided from the Fire Protection Water Supply System in order to keep water above the Top of Active Fuel.

Makeup from this source is possible when reactor pressure drops below 100 psia at about three hours, but does not become necessary until level reaches TAF at about seven hours.

2.4.6 SBO at Full Power With SORV and RCIC Failure

Plant response would be similar to SORV with HPCI failure except that injection from the Fire Protection Water Supply System must begin sooner. Injection would be possible at about 1.5 hours and would be required at about 2.5 to 3 hours.

2.4.7 SBO at Full Power With HPCI and RCIC Failure

Loss of HPCI and RCIC would leave no means of high pressure injection to the RPV. Since conservation of the RPV water inventory to keep the core covered as long as possible is considered of primary importance, the reactor would not be depressurized. The operator would take manual control of pressure relief, lifting individual SRVs in ADS mode, in order to minimize the number of SRV lifts in safety mode.

In the event that reactor vessel level drops to Top of Active Fuel with no other sources of injection available, the operator would depressurize rapidly using all six of the ADS mode SRVs and attempt to reflood the core with Fire Protection Water.

Analysis indicated that the reactor vessel water level would drop to the Top of Active Fuel within 45 minutes of the initiation of the blackout. Adequate core cooling might be expected until the core is less than half covered.

Use of injection from the Fire Protection Water Supply System cannot be accomplished for this event without uncovering of the core for a significant period of time. In order to use this low pressure injection water source, prompt depressurization of the vessel is required to much less than 100 psia. Given the decay heat rate and the thermal energy stored in the vessel and internals early in the transient, the vessel would have to be essentially emptied of water to accomplish this depressurization, resulting in core uncovery. The maximum possible injection rate at essentially complete reactor depressurization (35 psia) is 500 gpm. At this injection rate, the core is expected to be deprived of adequate cooling for at least 90 minutes; slower injection rates will result in proportionally longer core exposure times. As a consequence, severe core damage would be anticipated but core melt and consequential vessel failure would be avoided.

Imposition of an SORV aggravates the above transient, since the operator cannot prevent depressurization. Top of Active Fuel would be reached within 15 to 20 minutes. Core damage as described above is probable unless AC power is restored within this time frame.

2.4.8 SBO During Shutdown Cooling

Operator strategy in this case would be to let the vessel repressurize to a level where RCIC could be utilized to maintain inventory. Should RCIC be unavailable, HPCI (preferred) or injection from the Fire Protection Water Supply System would be available and adequate to replace boil-off. Intermittent operation of RCIC and an SRV in ADS mode would be required to control reactor pressure and water level. This strategy is similar to that at the latter stages (4-5 hours) of a blackout at full power.

In the event of a SORV, the reactor would not repressurize. The operator would allow water level to drop to the top of active fuel while efforts were made to restore power. If no other sources were then available, the reactor would be flooded with Fire Protection Water. Because of the low decay heat rate expected at this time, it is expected that flooding could be accomplished without core damage.

Since this event begins with the reactor depressurized, the suppression pool at normal temperature and the reactor core at a reduced level of decay heat, it is expected that the plant may be maintained within the safe parameter limits for a much longer time than the case for operation at power. The first limit reached will be suppression pool temperature limit which will occur at a time well beyond 8 hours.

2.5 Summary

The evaluations described in 2.1.1 and 2.4 indicate that considerable flexibility exists to maintain the plant in a safe condition should a station blackout occur. It is expected that no unsafe condition will exist prior to 6 to 8 hours after occurrence of SBO. Beyond eight hours, certain of the limits specified in Section 2.2 are likely to be violated. However, these violations are not expected to immediately place the plant in an unsafe condition. It is anticipated that adequate core cooling would be maintained, using methods based upon circumstances in existence at that time, for a greatly extended period of time.

The only equipment failure combination which will not allow the plant to be maintained in a safe condition is early failure of both HPCI

and RCIC. If either of these systems are available for as long as one hour after SBO occurrence it is possible to reduce reactor pressure rapidly via ADS and inject Fire Protection Water for makeup without significant fuel damage.

On this basis, the plant has the ability to withstand the SBO event and be maintained in a safe condition until AC power is restored. The probability of fuel damage is very low on the basis of the long time period available for reacquisition of AC power and the low probability for the loss of both HPCI and RCIC systems on occurrence of SBO.

FIGURE 2-1 - DECAY POWER AND ENERGY TRANSFERRED TO SUPPRESSION POOL DURING BASE CASE STATION BLACKOUT EVENT

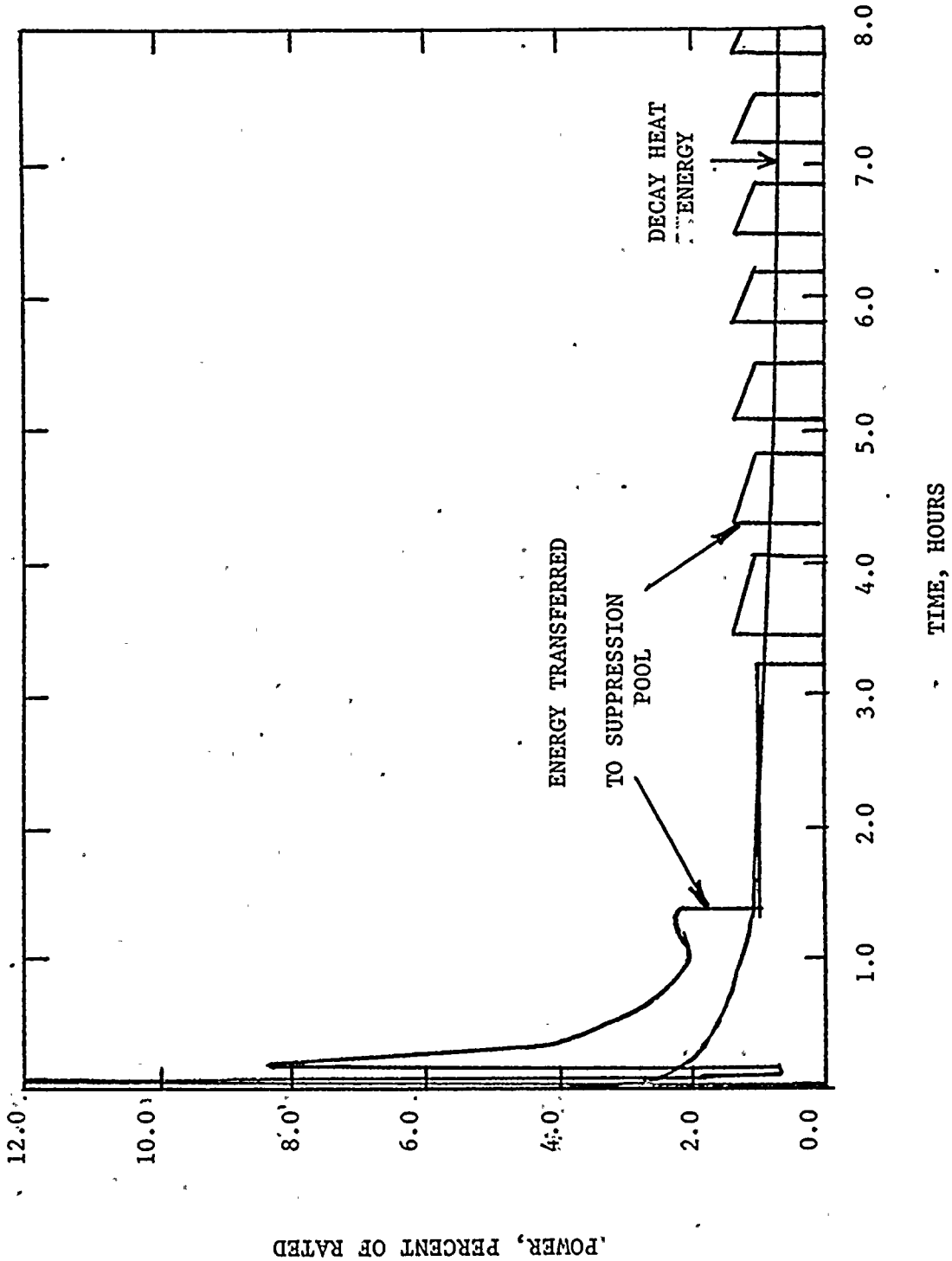
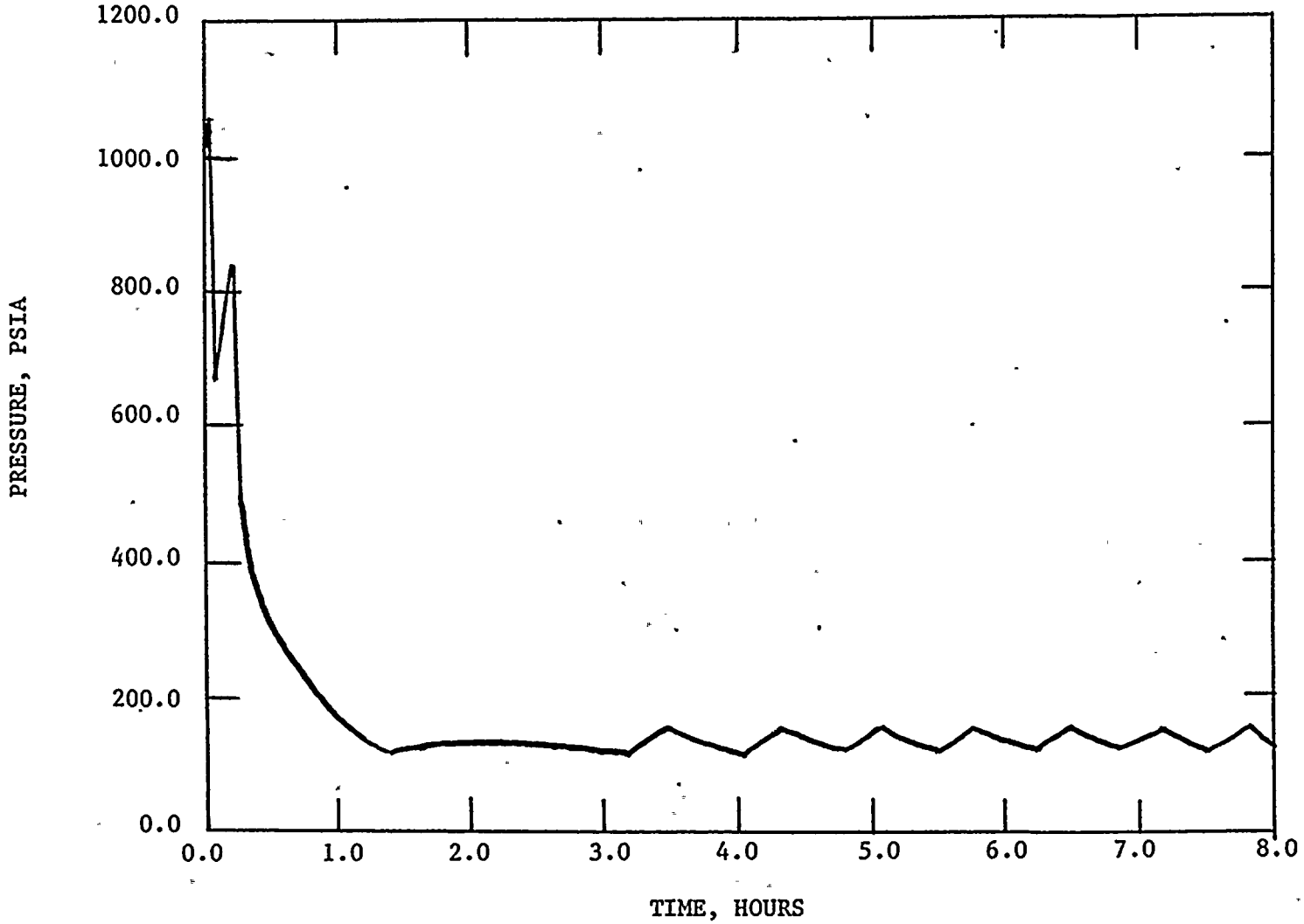


FIGURE 2-2 - REACTOR VESSEL PRESSURE VERSUS TIME

CURVE FOR THE BASE CASE



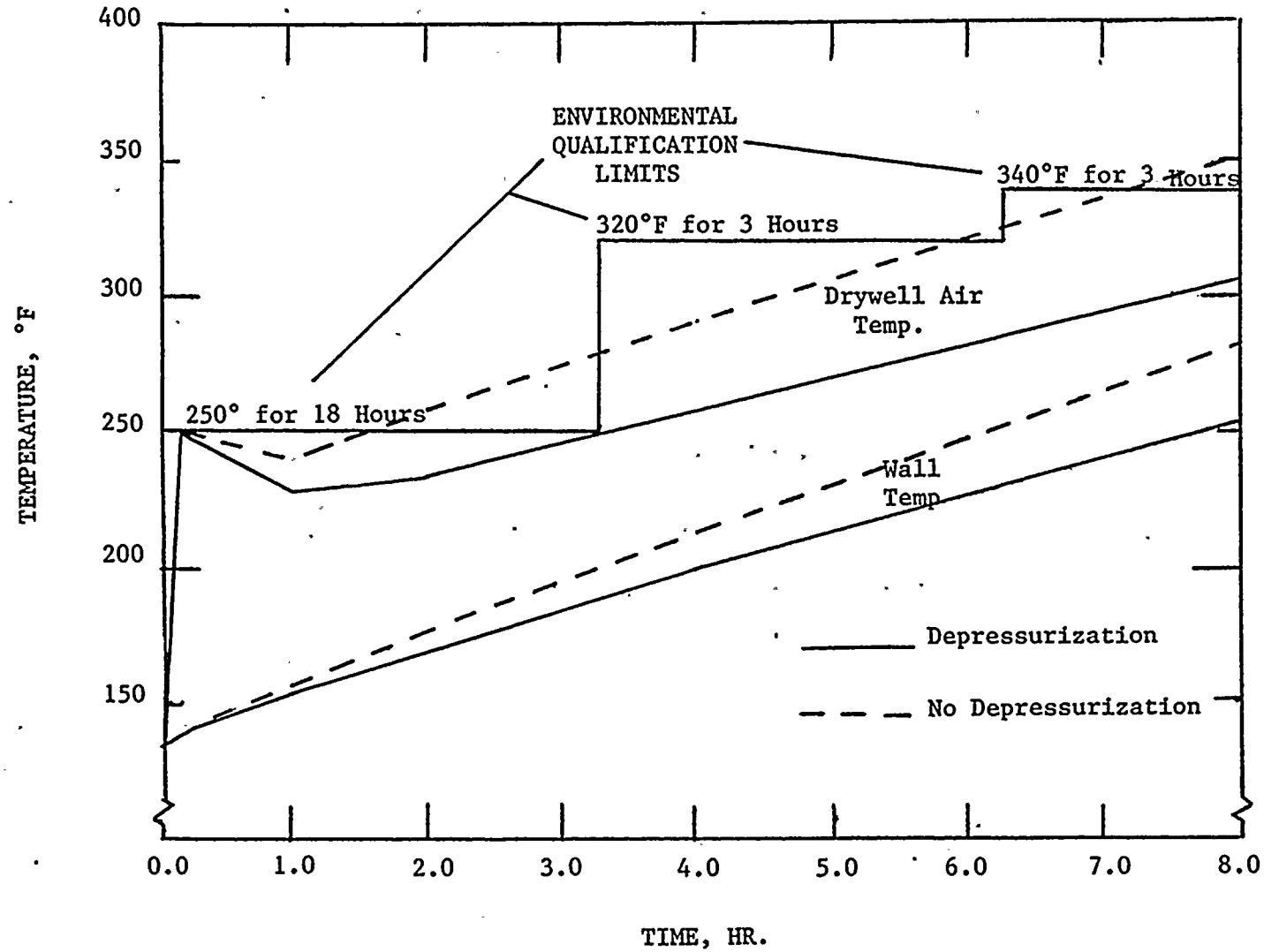


FIGURE 2-3 - DRYWELL AIR AND STRUCTURAL TEMPERATURE IN STATION BLACKOUT

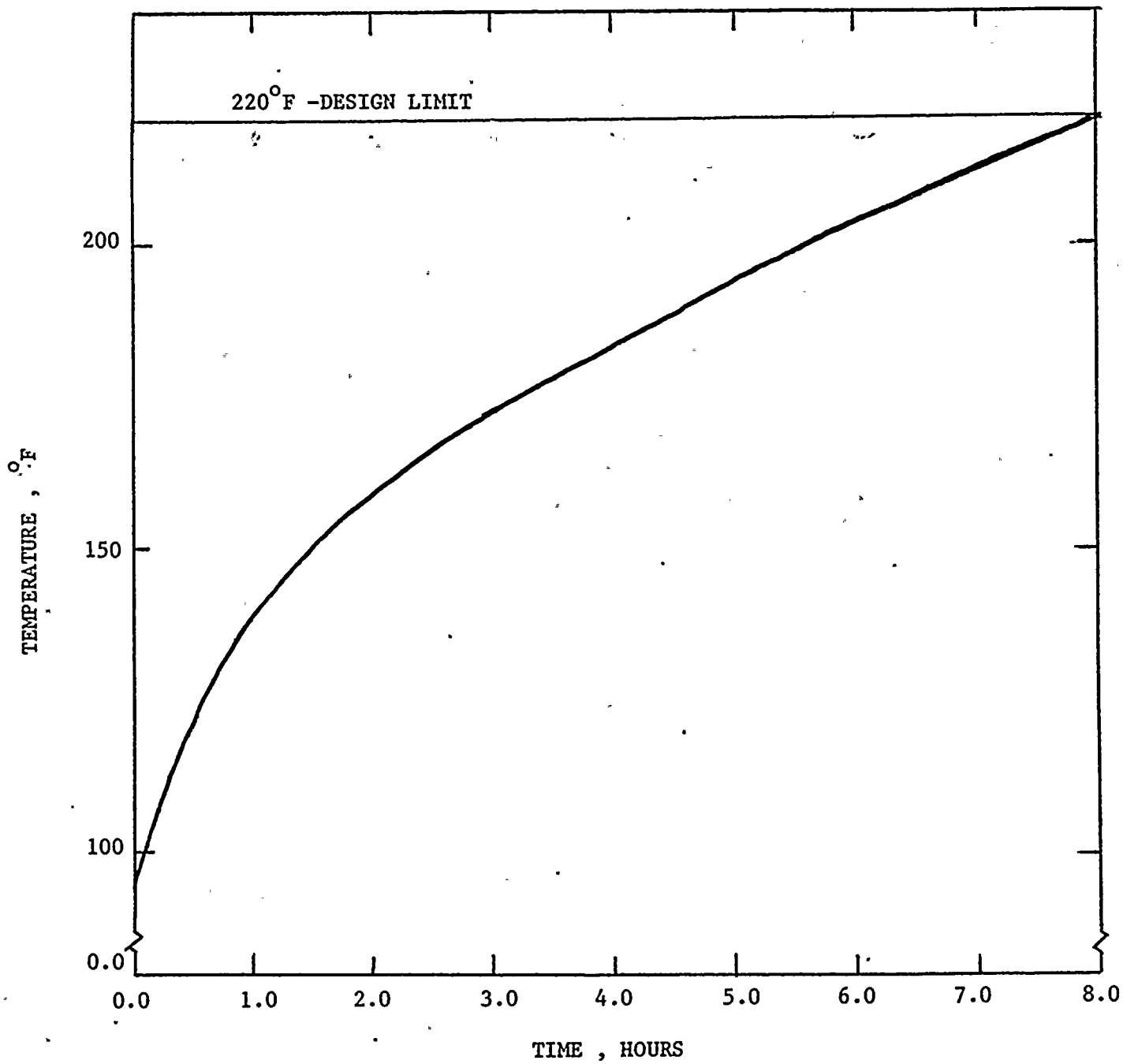


FIGURE 2-4 - SUPPRESSION POOL

TEMPERATURE IN STATION BLACKOUT

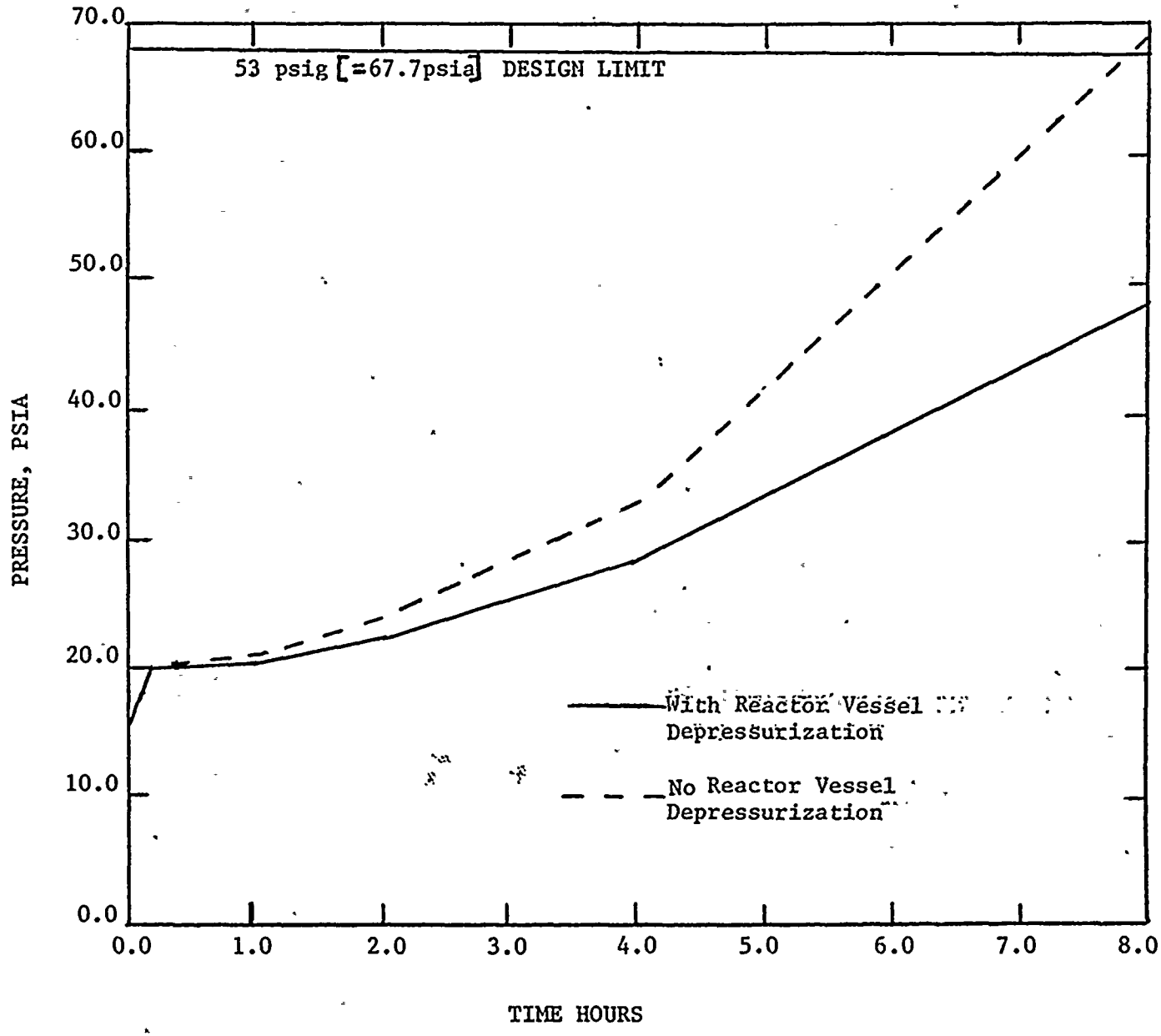


FIGURE 2-5 - DRYWELL PRESSURE DURING STATION BLACKOUT

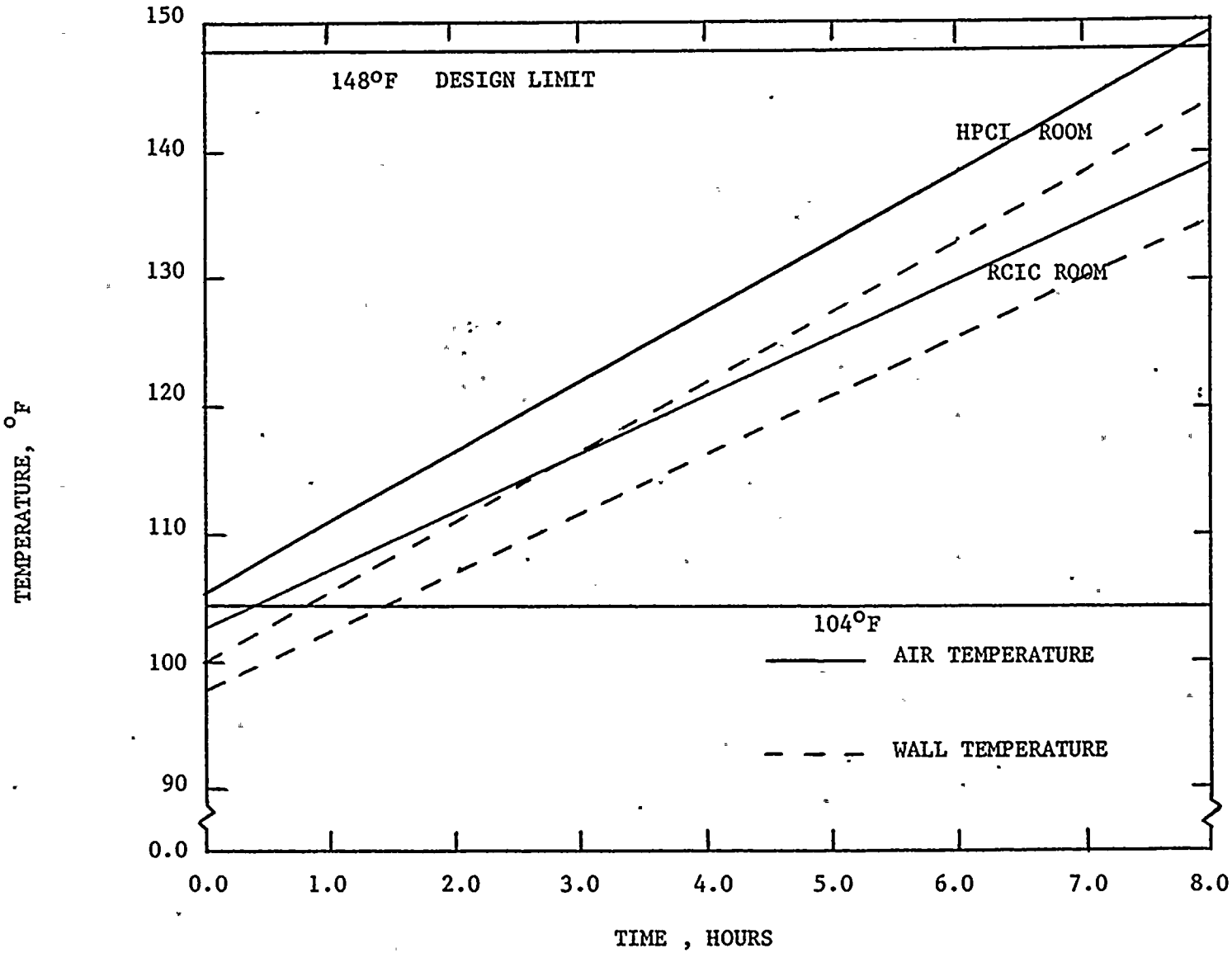


FIGURE 2-6: HPCI AND RCIC ROOM TEMPERATURES IN STATION BLACKOUT

3.0 EVALUATION OF STATION BLACKOUT TESTING

Section 3.1 presents the commitment made by PP&L in Reference (5) to perform a station blackout test. Section 3.2 describes a preliminary test plan prepared in October 1981 to satisfy that commitment. Section 3.3 presents the results of a detailed evaluation of that test plan and other testing options, and discusses the deficiencies and risks of such tests. Finally, Section 3.4 identifies additional testing which can be performed safely and would contribute to our understanding of station blackout.

3.1 Summary of Test Commitment

In Reference 5, PP&L committed to a test which would simulate a loss of AC power condition for the reactor and containment systems, in order to satisfy NRC requirements as stated in NUREG-0776 (Reference 6). The purpose of the test was to obtain data relative to the performance of these systems under the imposed condition of no AC power available for mitigation of transient effects. As described in Section 1.3.4, the primary use of the data would be for support of analytical predictions of the response of these systems. In order to protect plant equipment and to assure public safety during the test, the following restrictions were imposed.

1. Plant AC buses would not be de-energized, safety equipment would not be separated from power sources, and emergency diesel generators would be available or running.

2. Low pressure ECCS functions would not be blocked; the test would be planned to avoid initiation of these systems, but the systems would be permitted to function as designed in the event initiation points were reached.

3. Plant instrumentation would remain energized in order to assure control of the test and to provide for data collection.

4. Limiting conditions for operations would not be violated.

The test would be conducted at a convenient point during the first cycle, with the constraint that adequate decay heat existed to provide a valid test. Performance of the test was made contingent upon (1) a favorable safety evaluation, (2) NRC approval and (3) a continuing requirement for the test.

3.2 Station Blackout Test Plan

This section presents a description of a station blackout test formulated to satisfy the preceding commitment.

3.2.1 Objectives

The major objectives were as follows:

- To demonstrate the capability of the available equipment and planned operator actions in the Emergency Operating Procedures to maintain reactor water level and to conduct a controlled depressurization of the primary system;
- to collect data on temperature buildup in the drywell, suppression pool, and HPCI/RCIC equipment areas;
- to collect data on battery depletion rates.

3.2.2 Test Description

3.2.2.1 Prerequisites And Preconditions

Prior to initiating the test, the plant would have been operated at $\geq$ 85% power for $\geq$ 7 days to assure sufficient fission product inventory for significant decay heat.

Plant power would be rapidly reduced, following normal operating procedures, to the point of separating the main generator from the grid (∿20% power).

Normal plant systems not involved in the test would be shut down or placed on standby. The Control Rod Drive pump would be kept on to cool the CRDs and cooling would be provided to the recirculation pump seals. Safety related systems would be energized and placed in configurations appropriate for the test.

3.2.2.2 Test Initiation

In quick succession, (1) the reactor would be manually scrammed, (2) the turbine would be tripped, (3) the recirculation pumps would be tripped by opening the drive motor breakers, (4) the feedwater pumps would be tripped, and (5) the feedwater supply valves would be closed. The main steam isolation valves would be manually closed from the Control Room a few seconds later. At this point the reactor would be shut down and isolated. The remainder of the plant would shut down normally.

Cooling to the drywell, HPCI and RCIC equipment areas, would be manually shut off as follows:

a. Control switches for the HPCI and RCIC equipment area cooling fans would be placed to the "stop" position prior to initiating the test (these fans would not normally be

running at this time, since HPCI and RCIC would not have been started). In addition, manual valves in the Emergency Service Water supply to the area coolers (emergency cooling water supply) would have been closed.

b. All drywell cooling fans would be shut off, and Reactor Building Chilled Water isolation valves to the drywell coolers would be closed immediately following test initiation.

3.2.2.3 Control of System Parameters

HPCI and RCIC would be manually initiated prior to reaching their automatic initiation point, RPV Level 2 (L2). This would be done to avoid an unnecessary isolation of certain non-essential penetrations on L2. HPCI (primarily) and RCIC flow would be modulated to maintain normal water levels after the transient.

If the test were to last long enough that RCIC alone could maintain reactor level, HPCI would be placed in the full-flow test mode. In this configuration HPCI would be dissipating decay heat energy as work (to the CST) and as exhaust steam to the suppression pool.

A controlled reactor pressure reduction (cool down <100°F/hour) would be initiated 60 seconds after test initiation, utilizing HPCI in injection and test modes and safety relief valves (in alphabetic order) as necessary. The purpose of the pressure reduction would have been to simulate depressurization to reduce drywell heat load. The rate would have been limited to <100°F/hour to reduce the severity of the fatigue cycle on the reactor vessel. SRVs would have been lifted alphabetically to distribute decay heat evenly around the suppression pool.

Reactor Pressure would be reduced to 200-300 psig, a range consistent with the operation of HPCI or RCIC at full design output flow.

Data collected during the test would have consisted of the following:

- Reactor Vessel Level
- Reactor Vessel Pressure
- Reactor Vessel Temperature
- Drywell Temperatures
- Drywell Pressure
- Suppression Pool Level
- Suppression Pool Pressure

- Suppression Pool Temperatures
- HPCI Space Temperatures
- RCIC Space Temperatures
- Battery Parameters (current, voltage vs time)

3.2.2.4 Test Termination

The test would be terminated under either of the following conditions:

o Upon determination by the test director that sufficient data had been collected;

o Upon reaching a limiting value for a system parameter.

The plant would subsequently be brought to a normal hot standby condition by restoring all isolated systems and using normal operating procedures.

3.2.2.5 Limiting Values For System Parameters

In most cases, environmental limits for normal operation were chosen as the limiting values for system parameters. This was consistent with the restriction that station blackout testing pose no undue risk to plant equipment. Table 3-1 lists the parameters, limiting values, and bases.

TABLE 3-1

STATION BLACKOUT TEST
LIMITING PARAMETERS

Parameter	Limiting Value(s)
Reactor Vessel Level	
Initial Transient, Minimum	-38" (L2)
Selected to avoid isolation signal at L2	
Control Range	30"(L4) to 39" (L7)
Normal vessel levels	
Drywell Pressure	<1.5 psig
Selected to avoid high drywell pressure isolation signal at 1.68 psig	
Drywell Temperatures	
Normal Environmental Conditions	
Zones 1, 2, 5 - Maximum	150°F
Zone 3 - Maximum	185°F
Zone 4 - Maximum	135°F
Suppression Pool	
Level, Maximum Tech. Spec.	24'
Temperature, Maximum	120°F
Tech. Spec. MSIVs closed after scram	
HPCI and RCIC Room Temperature	<104°F
Normal Environmental Conditions	
Steam Relief Valve	Stuck Open
Recirculating Pump	
Seal Temperature, Alarm Set Point	<200°F
Batteries	
Tech. Spec., 80% Discharge	
Cell Voltage, Minimum	1.75 V/cell
Specific Gravity, Minimum	1.195

3.3 Test Deficiencies and Risks

3.3.1 Deficiencies in the Simulation of Station Blackout Conditions

In this section, plant response and operator actions under station blackout conditions (Section 2.0) are compared to those under test conditions to show deficiencies in the simulation.

3.3.1.1 Reactor Level and Suppression Pool Level Control

Under blackout conditions, HPCI and RCIC suction would be switched from the CST to the suppression pool (SP) early in order to minimize SP level rise during reactor depressurization. This would also conserve cooler CST water for use later in the transient, which would be helpful in assuring adequate cooling of HPCI and RCIC lube oil. In the proposed test, water for injection is drawn only from the CST. Thus, operator actions to control levels during the test would not duplicate actual blackout conditions. CST water after passing through the reactor vessel will accumulate in the SP as condensed steam. SP level will rise fairly rapidly during RPV depressurization in contrast to the expected slow rise (due to thermal expansion) corresponding to the station blackout response strategy.

The restriction of using only CST water during a test is necessary in order to avoid injecting suspended solids and impurities from the SP into the reactor in a non-emergency situation. Such impurities if injected could result in corrosion and fouling problems in the reactor. If activated, these impurities could produce higher plant radiation fields and thus increased man-REM exposure. The time required to clean up the reactor coolant prior to the resumption of power operation may impose a significant economic penalty in terms of lost production.

3.3.1.2 Drywell Temperature Response

If a station blackout were to occur, the operator would take the following actions in order to reduce the expected increase in drywell air temperature:

- a rapid depressurization of the reactor vessel would be initiated within the first ten minutes by holding open one of the steam relief valves;
- pressure would be reduced to the minimum required for HPCI and/or RCIC to provide reactor level control.

- if available, water from the Fire Protection Water Supply System would be provided to the drywell spray header via the RHR service water to RHR system cross connect.

The proposed test differs from actual conditions in the following ways.

- The cool down rate from depressurization ($\leq$100°F/Hr) is slower and thus would be less effective in controlling the peak drywell air temperature. This restriction is necessary in order to avoid a thermal cycle on the RPV which may reduce its fatigue life.

- The low pressure control range (for HPCI and RCIC operation) is about 100 psi higher in the test than it would be during a real blackout. This would produce higher drywell temperatures during the test, presuming that a test could be extended to these later stages of the event. The restriction is desirable for test conditions because it provides margin to keep HPCI and RCIC within their design operating envelopes.

- Drywell spray will not be used to cool the drywell during a station blackout test. Although such a test may be desirable to demonstrate the effectiveness of the connection with Fire Protection Water Supply System, it would have serious economic impact through damage to drywell equipment not required for safety.

3.3.1.3 Suppression Pool Temperature Response

Suppression pool temperature response during a test will not simulate blackout conditions in the following areas.

- Heatup Rate

  Suppression pool heatup rate during the test will be lower than during a blackout because:

  - the reactor will be depressurized at a slower rate, and

  - pool water inventory will be increasing due to the addition of CST water.

The reasons for these differences are discussed in sections 3.3.1.2 and 3.3.1.1, respectively.

o Condensation Load Distribution

For the test, the reactor is depressurized using a number of relief valves in order to equalize condensation loads around the suppression pool. In an actual blackout, one SRV would be held open for the entire depressurization. This action would minimize the number of valve lifts and thus the probability of a stuck open relief valve (SORV). RPV depressurization would be completed before suppression pool temperatures rose high enough to adversely affect the condensation of SRV discharge steam.

3.3.1.4 Battery Depletion Rate

Under station blackout conditions, the batteries assume the loads normally carried by their respective chargers as well as additional loads such as emergency lighting off the 125 VDC buses. All non-essential loads would be stripped in order to extend battery life. It will be difficult to simulate battery emergency loading and non-essential load stripping under test conditions. Loads cannot be stripped because the rest of the plant is energized and on standby during this test, and DC power is required for protective and control logics, for instrumentation, and for valve and pump motors throughout the plant. Battery chargers would remain in service during the test to assure the availability of DC power. As a consequence, test observations would not be representative of station blackout.

There is no need to perform a battery depletion test as part of a station blackout test. During pre-operational testing all station batteries passed a capacity discharge test; this test will be repeated every five years. Every 18 months batteries will be subject to a Service Test in which a simulated LOCA load profile will be applied. The manufacturers battery discharge curves have been used in conjunction with anticipated loads under station blackout to derive battery lifetimes.

3.3.2 Risks in Station Blackout Testing

3.3.2.1 Excessive Drywell Temperature

Termination of drywell cooling will result in an extremely rapid increase in drywell temperature. Assuming an initial drywell temperature of 135°F at 15.4 psia (0.7 psig), the environmental design specification of 150°F for normal operation will be reached in less than one minute. Within three minutes, the air temperature will have increased to the point that a high drywell pressure LOCA signal (at 1.68 psig) will be generated (173°F). Temperature will continue to rise to a peak of ∿250°F at five to

six minutes into the test. These estimates are for average drywell air temperatures. Much higher local temperatures may be experienced in thermal plumes above individual heat sources (scram discharge piping, SRVs and SRV exhaust lines) and in the containment dome region. Even with prompt restoration of drywell cooling, temperatures are not expected to return to normal for five to ten minutes. Longer exposures to excessive drywell temperatures are possible if difficulties are experienced with the restoration of cooling.

Accelerated aging and thermal degradation of non-safety equipment are potential consequences of such excessive drywell temperatures. Additional testing may be required to assure that the equipment is functional prior to restart, and premature failures may be experienced during operation. Thus, plant availability may be reduced as a result of excessive drywell temperatures during a station blackout test.

Safety-related equipment is expected to remain functional throughout the temperature excursion described above. However, the qualification of this equipment to survive future excursions (e.g. LOCAs) may be compromised. Re-analysis, re-qualification or replacement of safety-related equipment may be required, which would be a substantial cost to PP&L.

Operator options to minimize drywell heatup in the event of inability to restore cooling are limited. Drywell spraydown would not be initiated because of the certainty of water damage to drywell equipment. The reactor could be depressurized rapidly to reduce drywell heat load (ADS), but this would impose a severe thermal cycle on the reactor pressure vessel, reducing its design allowable fatigue lifetime and exposing PP&L to the potential costs of re-analyzing fatigue life or reduced vessel lifetime.

In summary, any test in which drywell cooling is turned off (an expected consequence of SBO) with primary system at operating temperature will result in drywell air temperatures exceeding normal operating limits within the first minute. Drywell cooling must be maintained during any station blackout test in order to avoid undue risk to equipment in the drywell. This restriction will prevent the determination of drywell thermal response to a station blackout.

3.3.2.2 Stuck Open Relief Valve (SORV)

The station blackout test will result in numerous relief valve lifts as a consequence of isolating the reactor and conducting a slow, controlled depressurization. SRVs may fail by sticking

open, or failing to close when the actuator is de-energized. This would result in an uncontrolled depressurization of the reactor vessel and loss of HPCI and RCIC for coolant injection. The rapid depressurization of the vessel would consume a portion of the vessel fatigue life, which may expose PP&L to future costs to re-analyze fatigue capability.

3.4 Planned Testing to Support Analytical Predictions

As indicated above, a single test of the response of multiple systems and components to station blackout cannot be formulated without undue risk to non-essential plant equipment. However, tests can be performed on single systems or components to verify their performance under blackout conditions. The startup tests described in Chapter 14 of the SSES FSAR have been reviewed to determine tests which are applicable to model verification for station blackout verification. Two tests have been identified, as follows:

ST-31 Loss of Turbine Generator and Offsite Power

ST-32 Containment Atmosphere and Main Steam Tunnel Cooling

During the performance of these tests, additional data will be collected and analyzed. Additional hot functional tests will be written to collect supporting information on the following systems: HPCI, RCIC, RPV level instrumentation. Data would be analyzed upon completion of the tests. Results would be available prior to commercial operation.

3.4.1 Loss of Turbine Generator and Off-Site Power (ST-31)

The station blackout event progresses exactly like the loss of off-site power event until the diesel generators come on (about 10 seconds). Therefore, data from the early portion of the test can be of significant importance to an overall evaluation of station blackout. The further data that should be gathered in this test include:

1. a trace of reactor water level as a function of time after scram initiation;

2. a trace of reactor power as a function of time after scram initiation;

3. SRV openings and closings as a function of time after scram initiation;

4. suppression pool level, pressure and temperatures as a function of time and temperature detector position;

5. drywell pressure and temperatures as a function of time and temperature detector positions;

6. performance data for the drywell coolers (air temperature), reactor building chilled water (RBCW) system (Δ T's and flow rate), reactor building closed cooling water (RBCCW) system, and emergency service water (ESW) system, together with the timing of various system re-alignments during the test.

Even though the LOOP test will begin to show deviations from the anticipated station blackout scenarios after about 20 to 30 seconds, the LOOP transient will be tracked for an extended period. The data accumulation will be valid supporting evidence for the adequacy of our transient analysis methods.

3.4.2 Containment Atmosphere and Main Steam Tunnel Cooling (ST-32)

The effect of the station blackout event on the drywell depends largely on the initial conditions in the drywell and the drywell heat load at normal conditions. Therefore, during the containment atmosphere cooling test, enough data should be gathered to give a good temperature profile in the drywell. Performance data on the cooling units should be taken to provide an indication of the steady state drywell heat load.

In addition to the temperature distribution at steady state, the transient heat loads due to scram initiated by closure of the MSIVs will be determined by measuring the temperature rise as a result of such events during this test. These results would then be used as supporting data for the calculations and the initial conditions used in the station blackout evaluation.

3.4.3 Additional Tests

3.4.3.1 Reactor Water Level Measurement Test

The measured reactor water level is a controlling parameter in the response to the station blackout event, and the analysis completed thus far has assumed that the pressure compensation on the level reading is exact at all stages of the transient, that is, the level instrumentation is capable of determining the actual steam-water interface at all times. Information on the error, if any, due to the density effects of reducing pressure would be desirable.

Data will be collected to compare the wide range and narrow range readings over the range from full depressurization to rated reactor pressure. From this

comparison, the accuracy of the instruments over the pressure range may be compared with the predictions. Results of this test will be useful both for model verification as well as the development of operating procedures.

3.4.3.2 RCIC and HPCI Operation.

The HPCI and RCIC systems, being turbine driven by reactor steam are the two systems that are available for injection of makeup water into the pressurized reactor vessel during a station blackout event. These systems are expected to be functional under conditions that are outside those anticipated in normal operation. Tests of these off-normal conditions have been planned.

The operating map (or head-capacity curve) of these systems at steam pressures below the rated full flow minimum pressure of 150 psig will be determined. In the SSES response to station blackout, the reactor vessel is depressurized to as low a pressure as possible to reduce drywell heat load. Thus, this information will permit verification or modification of the minimum pressure limits specified for station blackout.

Full flow of the HPCI and RCIC systems is not needed at all times during a station blackout. Therefore, the current anticipated response to the event is to throttle back on the flows as required to maintain a constant level. The performance of the HPCI and RCIC systems under throttling conditions will be investigated as part of the testing done on the systems.

A test has been planned to measure room temperature rise when the HPCI and RCIC systems are operating at full flow with no room cooling. The average temperature rise in the HPCI and RCIC room, as well as the temperatures in the vicinity of the electronic control panel, will be determined. Determining temperatures around the turbine controller electronics is important because the electronics have been (or will be) qualified to function up to 150°F for twelve hours. Results from this test will be used to support the analytical predictions that the average room temperature does not reach 150°F until about eight hours after the event initiation. Additionally, since essentially the same modelling approach was used to predict the drywell heatup as was used to predict HPCI and RCIC equipment room heatup, test data supporting the equipment room predictions would increase confidence in the drywell calculations.

4.0 CONCLUSIONS

PP&L has evaluated plant response to station blackout and has assessed the extent to which that response can be simulated in testing. Restrictions on testing necessary to protect public health and plant equipment prohibit a plant-wide station blackout test and severely limit the degree to which expected station blackout conditions can be simulated in the reactor vessel and drywell. Station Blackout testing is essentially limited to performance and response testing of a limited number of specific systems and components where blackout conditions can be safely simulated.

As a consequence, the value of station blackout testing towards determining SSES capabilities or providing operator familiarization is minimal. Such testing would generate data which would provide increased confidence in the analytical methods used to predict plant response. PP&L will utilize data from the Start-up Tests and will conduct additional hot functional tests to generate data on the response of individual systems and components to station blackout conditions. Test data will be used to support the adequacy of analytical models.

The primary NRC objectives are being addressed through a study of Station Blackout which was undertaken by PP&L in response to NRC concerns expressed in Generic Letter 81-04 (Reference 1). The objective of this effort is to understand plant response to SBO, and to develop strategies, procedures, and training for an SBO event. PP&L has determined the initial plant reaction to SBO and has drafted preliminary procedures; operator training programs based on these procedures will be prepared prior to fuel load.

A thorough engineering evaluation of plant response to SBO has been performed. Plant and equipment design have been reviewed to assure operability under SBO conditions, analytical models have been constructed to allow prediction of the response of major plant parameters to various SBO scenarios, and strategies have been developed which maximize the time that safe conditions can be maintained while minimizing the risk of failure of key equipment. These efforts have demonstrated that SSES can tolerate SBO conditions for extended times, even if further equipment failures following SBO are postulated.

Output from the engineering evaluation and testing during the initial start-up program will be incorporated into revised SBO procedures and updated operator training during the first operating cycle.

Upon completion of these efforts PP&L will have developed an adequate understanding and predictive capability for plant response in the remote chance that a station blackout would occur. This knowledge will have been incorporated into contingency planning and operator training programs. PP&L believes that this approach satisfies the regulatory objectives underlying the requirement for a station blackout test, so that a special station blackout test is not necessary.

5.0 REFERENCES

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