Attachment #1

Clinton Power Station

Station Blackout Evaluation Report

July 3, 1985

Illinois Power Company

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EXECUTIVE SUMMARY

As part of the resolution of TMI Action Plan (NUREG-0694) Item I.G.1, entitled "Operator Training During Low Power Testing", the NRC required all BWR applicants for an operating license to commit to performing an in-plant Station Blackout (SBO) Test. The NRC Staff objectives for such testing would include evaluation of plant response and enhanced operator training under SBO conditions. The Staff modified this requirement via NRC Generic Letter 83-24, dated June 29, 1983, which allows compliance through participation in the BWR Owner's Group Program for TMI Action Plan Item I.G.1, if the applicant can demonstrate that temperature and/or other SBO test conditions would adversely impact and pose a hazard to plant equipment.

This report provides the results of a Clinton Power Station (CPS)-specific analytical evaluation of the plant response to SBO. The results conclusively demonstrate that such testing, if performed at CPS, would only partially fulfill the NRC Staff SBO test objectives and could potentially damage plant equipment.

Clinton Power Station can safely withstand an SBO event for a duration of at least 2.4 hours, given the Reactor Core Isolation Cooling (RCIC) system remains available. Given the conservatism in the calculations provided, this coping capability is likely to exceed 4 hours if less conservative methods of calculation are used.

Due to the potential risks from performing an SBO Test, such full-scale tests should not be performed at CPS. Tests are identified that will provide information and training about the plant response under SBO conditions.

1.0 INTRODUCTION

This report presents the results of an evaluation of the current capabilities of Clinton Power Station (CPS) to withstand a Station Blackout (SBO) and provides justification for not performing an SBO test. The SBO test requirement was proposed by the NRC as part of TMI Action Plan Item I.G.1, "Training During Low Power Testing" (References 1 and 4). This report describes design features at Clinton Power Station to prevent and mitigate the consequences of an SBO event and demonstrates the ability of CPS to withstand an SBO based upon analytically predicted plant responses. The report also describes the risks and deficiencies associated with performing an SBO test, including the adverse impact that the SBO test would have on plant equipment. This approach is in accordance with NRC Generic Letter 83-24, which allows for eliminating the Station Blackout test requirement if adverse impact from the test can be demonstrated.

This section provides a brief history of the SBO issue, a description of the NRC objectives for the SBO test and a description of how the NRC objectives will be achieved at CPS.

Section 2.0 describes those CPS design features that are available for decay heat removal. This section provides information about the electrical power supplies required for each heat removal system to show the independence of supply and the ability to maintain core cooling despite many electrical failures.

Section 3.0 presents the analytically predicted CPS response to an SBO event. This section includes a description of the modeling assumptions used and discusses the time required to exceed defined plant safety limits.

Section 4.0 provides justification for not performing in-plant SBO testing. The justification is based upon deficiencies and risks identified with SBO testing. CPS preoperational and startup tests that will accomplish objectives of the SBO test are also discussed.

Section 5.0 presents Illinois Power's conclusions with regard to this report.

1.1 <u>HISTORY OF THE STATION BLACKOUT ISSUE</u>

Station Blackout is defined as the loss of offsite power (LOOP) to a generating unit coincident with the failure of onsite emergency diesel generators to deliver power to their respective safety-related buses. Such an event would deprive the plant of all AC power except that provided from battery-inverter arrangements. It would eliminate the availability of many systems normally utilized to achieve safe shutdown conditions and to mitigate design basis events.

NRC concerns over risk to the public from an SBO arise from uncertainty in the reliability of offsite and emergency onsite

power supplies. Previous history on failures of diesel generators to start or provide power on demand and loss of offsite power events have contributed to this concern. Because of this, the NRC has identified Station Blackout as an Unresolved Safety Issue (A-44) (Reference 2). This unresolved safety issue is under review by the NRC.

The requirement for performing an SBO test arose from TMI Action Plan Item I.G.1, "Training During Low Power Testing". This Action Plan item, as presented in NUREG 0694 (Reference 3), requested that applicants, "...define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training."

In an October 27, 1981, letter from the NRC (Reference 4), Illinois Power was requested to commit to performing an SBO test as part of TMI Item I.G.1.

In the CPS-FSAR Appendix D, TMI Item I.G.1 response, Illinois Power committed to performing a Low-Power Test Training Program to be developed using the guidelines provided in the report "BWR Owners' Group Program for Compliance with NUREG 0737, Item I.G.1 Training During Low Power Testing" (Reference 5). Illinois Power also committed to performing a Station Blackout test if the

results of similar tests that were to be performed at the LaSalle and Grand Gulf stations indicated that the SBO test could be performed safely and would provide useful information and training for CPS.

In NRC Generic Letter 83-24, dated June 29, 1983, the NRC modified their position on SBO testing by stating "...if it can be demonstrated that temperature and/or other SBO test conditions would adversely impact and pose a hazard to plant equipment, the BWR Owners' Group recommendations by themselves would constitute compliance with Item I.G.1...". Section 4.0 of this report provides justification for not performing the Station Blackout test, in accordance with this generic letter.

A more complete chronology of the Station Blackout / I.G.1 issue is provided in Table 1-1.

1.2 NRC OBJECTIVES FOR THE SBO TEST

TMI Action Plan Item I.G.1 requires applicants for a new operating license to define and commit to a special low power test program in order to (1) provide meaningful technical information beyond that obtained in the normal startup test program and (2) provide supplemental training. The training component of this requirement can be satisfied by the BWR Owners' Group generic response to Item I.G.1. The general objectives and criteria for the testing requirement are as follows:

1. to provide meaningful technical information or data relative to plant response during off normal conditions, specifically information not provided by any of the tests described in Regulatory Guide 1.68, "Initial Test Programs" (Reference 6);
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.

When the NRC indicated that a simulated loss of all AC power or "Station Blackout" test would be required, several additional objectives were added as follows:

1. to determine the limitations and capabilities of BWRs to maintain safe reactor and containment conditions in the event of a station blackout.
2. to provide data on the response of reactor vessel and containment parameters during a Station Blackout (the 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); and
3. to familiarize operators with plant response to Station Blackout;

1.3 ACHIEVEMENT OF NRC OBJECTIVES AT CLINTON POWER STATION

The following subsections describe how the NRC objectives for the SBO test listed in Subsection 1.2 are being met at Clinton Power Station.

1.3.1 LIMITATIONS AND CAPABILITIES

Illinois Power has evaluated the capabilities and limitations of CPS to maintain safe reactor and containment conditions during an SBO event. The plant responses were determined analytically and are discussed in Section 3.

1.3.2 VERIFICATION OF ANALYTICALLY PREDICTED 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. Evaluation has shown that limitations necessary to protect plant equipment would preclude the determination of reactor vessel and primary containment response through testing. However, Preoperational and Startup tests have been identified that will verify the predicted performance of certain components needed for Station Blackout response. Section 4 presents an

evaluation of SBO testing and describes tests which can be used to confirm aspects of the analytical models used.

1.3.3 OPERATOR FAMILIARIZATION AND TRAINING

Because of restrictions necessary to protect the plant and 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 more effectively through a combination of classroom instruction (including procedure reviews), equipment walkdowns and reviews of system test results.

1.3.4 SUMMARY

The capabilities and limitations of CPS under SBO conditions have been determined through analytical modeling.

Although an SBO test is not desirable because of certain test deficiencies and risks, Preoperational and Startup tests will be conducted that can be used to confirm aspects of the analytically predicted response. Operator familiarization will be provided through methods other than actual Station Blackout testing. Therefore, a Station Blackout test at Clinton is not being planned. These conclusions are in accordance with NRC Generic Letter 83-24 requirements.

TABLE 1-1
Chronology of Events Related to
Low Power Testing and Training / Station Blackout Testing

DATE	Event Description	Discussion
5/80	NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident" Issued.	Item I.G.1 proposed that operating license applicants conduct a special "set of low power tests" to improve training of plant staff for off-normal events.
6/80	NUREG-0694, "TMI-Related Requirements for New Operating Licenses" Issued.	Item I.G.1 clarified to require development of a special low-power test program which will provide meaningful technical information beyond that obtained in the normal startup program and to provide supplemental training.
7/80	BWR Owners' Group for TMI Activities Authorized Generic Program to Address I.G.1.	
7/80	NRC identifies the Station Blackout Issue as an unresolved safety issue. (Task Action Plan A-44; NUREG-0606)	
11/80	NUREG-0737, "Clarification of TMI Action Plan Requirements" Issued.	Numerous TMI Action Plan requirements including I.G.1, approved by Commission for implementation.
1/16/81	NRC Letter to Commonwealth Edison and Pennsylvania Power and Light	NRC proposed that the LaSalle and Susquehanna programs for complying with I.G.1 include a simulated loss of onsite and offsite A/C power test and be conducted during the first cycle, subject to performing an acceptable safety analysis.

TABLE 1-1 (continued)
Chronology of Events Related to
Low Power Testing and Training / Station Blackout Testing

DATE	Event Description	Discussion
2/4/81	BWROG Generic Report on I.G.1 Issued to NRC .	Areas were identified where increased emphasis on operator training could be beneficial. In addition, several new tests were identified as follows: 1. Startup of RCIC system after loss of AC power to system. 2. Operation of RCIC system with a sustained loss of AC power. 3. RCIC operation to prove DC separation. 4. Integrated RPV level measurement function test. 5. Integrated containment pressure measurement test.
2/9/81	Commonwealth Edison Company Commits to Perform Loss of AC Power Test at LaSalle	
3/81	NRC Safety Evaluation Report for LaSalle Issued	License Condition No. 18g required the performance of the loss of AC power test as committed to by CECo in their 2/9/81 letter.
4/81	NRC Safety Evaluation Report for Susquehanna Issued	Outstanding Issue No. 79 states that PP&L has not committed to perform the loss of AC power test as part of the I.G.1 program and must do so to close this issue.

TABLE 1-1 (continued)
Chronology of Events Related to
Low Power Testing and Training / Station Blackout Testing

DATE	Event Description	Discussion
5/15/81	PP&L Commits to Perform Loss of AC Power Test at Susquehanna	
8/17/81	MP&L Commits to Perform Loss of AC Power Test at Grand Gulf.	
9/81	Grand Gulf SER Issue	TMI requirement I.G.1 classified as Confirmatory Issue No. 28.
9/81	CPS-FSAR Appendix D Issued	Appendix D, addressing TMI requirements, stated that IP would implement the BWROG program for I.G.1.
10/27/81	NRC Letter to IP	NRC requested that IP commit to perform the loss of AC power test at Clinton.
12/1/81	IP Meetings with NRC on SER Preparation	NRC informed IP that issue I.G.1 would be classified as an outstanding issue unless a commitment to perform the loss of AC power test is made.
1/4/82	CPS-FSAR Appendix D Revised	Item I.G.1 of Appendix D revised to commit IP to perform the loss of AC power test if such test is deemed safe and worthwhile results are achieved in the LaSalle and Grand Gulf tests.
2/82	CPS-SER Issued by NRC	TMI requirement I.G.1 classified as Confirmatory Issue No. 42. Issue to be resolved by performance of the loss of AC power test.
4/17/82	Operating License for LaSalle Issued	License Condition 2.C.30.e requires the loss of AC power test to be performed.

TABLE 1-1 (continued)
Chronology of Events Related to
Low Power Testing and Training / Station Blackout Testing

DATE	Event Description	Discussion
6/15/82	PP&L Issued Safety Evaluation on Loss of AC Power Test to NRC.	A detailed safety evaluation of the loss of AC power test was submitted to the NRC. This evaluation concluded the following. 1. PP&L has a thorough understanding of the event and its consequences and a plan to mitigate them. 2. Susquehanna plant can survive a station blackout of limited duration (6-8 hours) with appropriate operator actions. 3. A station blackout test will unnecessarily jeopardize the plant and the public. Of particular concern will be the exposing of plant equipment to conditions close to qualification limits. Reanalysis or replacement of equipment would be necessary, if this test were performed.
6/16/82	Operating License for Grand Gulf Issued	License Condition 2.C.44.b requires the loss of AC power test to be performed.
7/17/82	Operating License for Susquehanna Issued	License Condition 2.C.28.b requires the loss of AC power test to be performed

TABLE 1-1 (continued)
Chronology of Events Related to
Low Power Testing and Training / Station Blackout Testing

DATE	Event Description	Discussion
10/15/82	PP&L Meeting with NRC to discuss their 6/15/82 Safety Evaluation.	NRC stated tentative agreement with PP&L's safety evaluation on why performing the loss of AC power test is not recommended.
5/83	NUREG/CR-3226 "Station Blackout Accident Analysis (Part of NRC Task Action Plan A-44)", was issued.	Provided a methodology for estimating core damage probabilities due to Station Blackout.
6/29/83	NRC Issues Generic Letter 83-24 on Performing Loss of AC Power Test at BWRs.	Generic Letter 83-24 acknowledges the conclusion of the Susquehanna evaluation that performance of the test could have the potential for damaging equipment. All operating license applicants and recently licensed BWRS (LaSalle and Grand Gulf) were required to provide justification for not performing the station blackout test. If the plant-specific evaluations reach this conclusion, then performance of loss of AC power tests would no longer be required. All BWR utilities must, as a minimum, commit to the BWROG I.G.1 program.
7/83	NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants", was issued.	Provided estimates of the probability of loss of onsite AC power for certain plant configurations.
8/2/83	CECo Issues Safety Evaluation on Performance of Loss of AC Power Test at LaSalle	An evaluation report, similar to the PP&L report on Susquehanna, was issued by CECo on LaSalle in response to Generic Letter 83-24.

TABLE 1-1 (continued)
Chronology of Events Related to
Low Power Testing and Training / Station Blackout Testing

DATE	Event Description	Discussion
8/11/83	Station Blackout issue first presented to the BWROG as a 10CFR rulemaking issue.	SBO introduced as part of TMI Action Plan Item I.G.1.
11/83	SSER No. 6 on LaSalle Issued	This supplement accepted the CECo evaluation of the loss of AC power test at LaSalle. The License Condition requiring performance of the test was deleted.
7/84	NUREG/CR-3840 "Cost Analysis for Potential Modifications to Enhance the Ability of a Nuclear Plant to Endure Station Blackout" was published.	Provided cost estimates for plant modifications intended to increase the coping ability of nuclear plant to SBO. (e.g. increasing battery capacity, increasing compressed air supply for instrument air.)
7/84	NSAC/80 "Losses of Off-Site Power at US Nuclear Power Plants All Years Through 1983" was issued.	Provided statistical data and brief event descriptions for loss of off-site power events that have occurred at US Nuclear Power Plants.
1/85	NUREG 1032 "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44" released in draft form.	Provided data concerning the probability of Station Blackout events, and the contribution to core damage risk due to SBO.
2/85	NUREG/CR-3992 "Collection and Evaluation of Complete and Partial Losses of Off-Site Power at Nuclear Power Plants".	Provided statistical data concerning loss of off-site power events, and evaluated factors that contribute to these events, to determine which factors are most significant.
3/12/85	Letter sent from D. A. Ward (ACRS) to W. J. Dircks (NRC) "ACRS Comments on the NRC Staff Proposal for the Resolution of USI A-44 Station Blackout.	This letter reaffirms that the proposed recommendations of the NRC Staff are appropriate for resolving USI A-44.

2.0 CPS DESIGN FEATURES FOR DECAY HEAT REMOVAL

This section of the report describes design features Clinton Power Station has available for decay heat removal. The emphasis of this section is upon the electrical power supplies required for each decay heat removal system. The ability to remove decay heat is the primary concern associated with Station Blackout. This section puts Station Blackout into proper perspective as an event involving multiple failures of both onsite and offsite power sources.

2.1 ELECTRICAL POWER SYSTEMS

An SBO involves a loss of all offsite and emergency onsite AC power supplies. For CPS these supplies are designed to ensure continuous operability of the 4.16 kv Engineered Safety Feature (ESF) buses and orderly control of the reactor following a Loss of Coolant Accident (LOCA). The following subsections describe the AC power systems showing qualitatively their reliability and independence of supply. In addition, a description of the emergency DC power supplies is provided since they would be available to power certain instruments and controls during SBO conditions.

2.1.1 OFFSITE AC POWER SYSTEM DESIGN

Clinton Power Station is supplied from four offsite AC power sources. Three separate 345 kv transmission lines supply the Reserve Auxiliary Transformer (RAT) which in turn supplies the plant ESF buses. A separate 138 kv transmission line supplies the Emergency Reserve Auxiliary Transformer (ERAT) which serves as a backup offsite power supply for the ESF buses. For a single line drawing of the AC power supply system serving CPS see figure 2-1.

The three 345 kv transmission lines for CPS are the Clinton-Brokaw line (approximate transmission line length of 22.4 miles), the Clinton-Latham line (approximate transmission line length of 29.5 miles), and the Clinton-Rising-Oreana line (approximate transmission line length of 51.3 miles). Approximately 7.0 pole-miles of the 345 kv lines running from Clinton to Latham and from Clinton to Rising are on common double-circuit single-pole steel structures. The remainder of both lines consists of single-circuit, wood H-frame structures, with the exception of 2.7 miles of laminated wood "Y" structures in the Rising line. The 345 kv line from Clinton to Brokaw uses single-circuit wood H-frame construction.

The three 345 kv lines are connected in the CPS switchyard and in turn are used to supply the RAT. The RAT supplies

the three divisions of ESF buses which have an operating voltage of 4.16 kv. The RAT also supplies 6.1 kv buses and other 4.16 kv buses which serve non-class 1E loads.

The 138 kv line is 8.4 miles long and originates from a 138 kv line running between the North Decatur and South Bloomington Substations.

The 138 kv line supplies the ERAT, which in turn can be used to supply the three divisions of 4.16 kv ESF buses.

The RAT is the normal power supply for these buses, but in the event it should become unavailable, the ERAT serves as the backup offsite power supply. Switching between these two offsite sources, and the emergency diesel generators (the diesel generators are described in the next section) can be accomplished in the following ways:

1) Automatic Fast Offsite Source Transfer:

   This transfer occurs if the following conditions are fulfilled:

   a. All source breakers become open as a result of the supplying source breaker opening on its feeder bus lockout protection actuation.

   b. The other source is "available" to the bus at the instant all source breakers become open.

c. The synchronism checking relay permits closure.

Shedding of the loads on the 4.16 kv buses does not occur during fast transfer. If both reserve and emergency reserve transformer source breakers are available to the bus, the reserve transformer source will have closing priority. Automatic fast closing of the diesel generator source breaker is not employed.

2) <u>Automatic Slow Source Transfer</u>:

This transfer occurs if the following conditions are fulfilled:

a. All source breakers are open for any reason.

b. Fast source transfer has not occurred. Fast transfer will not occur when there is no other transfer source "available" at the instant all source breakers become open, or when fast transfer is not initiated by instruments on the source that was lost.

c. A source becomes "available" to the bus after the bus undervoltage relays have tripped all bus breakers feeding motor services (e.g. a reserve auxiliary transformer is reenergized).

After the slow transfer has occurred, loads on the 4.16 kv buses are reconnected in sequence.

3) Manual Source Transfer:

A manual source transfer can be made by the control room operator provided the voltage and frequency of the new intended source is verified to match that of the existing source for the bus.

2.1.2 ONSITE AC POWER SYSTEM DESIGN - DIESEL GENERATORS

In the event that all offsite power sources become unavailable, each of the three divisions of safety related power has its own emergency diesel generator that will automatically start and assume the loads of that division. The starting circuitry and control power is provided by a 125 volts DC battery for each division load group (the emergency DC power supplies are described in the next section). The diesel generator automatic starting and loading will proceed as follows:

a. Each diesel generator is automatically started by one of the following events:
    1. associated bus voltage below preset value for a predetermined period of time (indicative of a zero voltage condition) or
    2. reactor low water level; or
    3. high drywell pressure; or
    4. associated bus voltage below preset value for a predetermined period of time (indicative of a degraded voltage condition).

b. Upon loss of voltage at the 4.16 kv division buses, all 4 kv motor loads on the Division I and Division II buses will be shed. Division III loads are not shed following a loss of bus voltage.

c. After each diesel generator set has accelerated to approximately rated frequency and voltage, its breaker will close if normal AC power has not been restored to either of the other sources.

d. Loads are started in a predetermined sequence.

e. If normal AC power is still present and the diesel generator was started by signals indicated in Items a.2 or a.3, the diesel-generator breaker will not close, and the set will remain unconnected at rated frequency and voltage until manually shut down.

f. If normal AC power is lost and signals indicated in Items a.2 or a.3 are not present, only the loads needed for safe shutdown will be connected automatically or manually by the operator's action as station conditions require.

g. If, while operating as per Item f, a LOCA signal is received, any nonemergency load that is running will be automatically tripped, and the required Class 1E loads will be started automatically as in Items c and d.

h. During sustained low grid voltage conditions which cause Class 1E equipment to operate at voltages outside their recommended continuous operating limits, an additional level of relays is provided which will automatically disconnect offsite power sources and start diesel generators whenever the voltage setpoint and time delay limits have been exceeded.

Interlocks prevent automatic closure of the diesel-generator breaker (after an automatic start) unless the normal and reserve source breakers are all open.

The diesel generators are started via separate air driven starting systems supplied for each diesel generator set. The Division I and II starting systems each consist of six air powered starting motors, three for each of the two diesels in a division. The Division III diesel has four air start motors for the single diesel in that division. Each of the sets of three or four air start motors is supplied from an air start receiver which stores sufficient air to start the diesel five times. The receivers for each division are supplied from redundant air compressors through a single air dryer.

Each diesel generator set is supplied with diesel fuel oil from a day tank which has enough capacity to supply the diesel for two hours at maximum load. The day tank can be

refilled from diesel storage tanks (one for each division) which hold enough fuel to supply the engine for seven days at maximum load. The fuel is transferred from the storage tank to the day tank by an electric fuel transfer pump which is supplied from divisionally associated Class 1E motor control centers.

The diesel engines are cooled by the diesel generator cooling water system that removes heat from the diesel engine and rejects heat to the associated division of shutdown service water via a heat exchanger. The pumps for the cooling water system are gear driven by the associated diesel.

The lube oil for the diesels is circulated by pumps driven by the diesel, or driven by motors supplied from divisionally associated Class 1E AC and DC power supplies. (The motor driven pumps are used for lubrication when the diesel is not running.)

Failure of any component within the diesel generator set for one division will not impair the function of the diesel generators in the other divisions.

2.1.3 EMERGENCY DC POWER SYSTEM DESIGN

The Class 1E DC power system is divided into four electrically and physically independent divisions. Each division operates at a nominal voltage of 125 volts DC and is supplied from a battery or a battery charger. The battery and battery charger operate in a "float charge" configuration. Loss of either source does not interrupt power flow to the bus. Each battery system operates ungrounded. Because of this it is impossible to render the system inoperable by having one wire inadvertently make connection with ground.

The battery chargers are fed from divisionally associated 480 VAC motor control centers, except for the Division IV battery charger which is fed from a Division II 480 VAC motor control center. Each battery charger is adequately sized to meet the demand of associated steady state DC loads during any mode of station operation while recharging a fully discharged battery.

Batteries are the secondary sources of power to the emergency DC system. The ampere-hour capacity of each battery is adequate to supply expected essential loads, assuming lighting loads are shed after 1 hour, for a period of four hours following station trip and loss of all AC power coincident with a design basis accident without

battery terminal voltage falling below 84% of rated voltage. Service to the equipment needed to mitigate the consequences of a Station Blackout can be maintained for longer than four hours by further reducing the loads on these DC buses (see section 3.2.11).

The DC buses supply DC loads directly through DC distribution panels. In addition, some AC loads such as the Nuclear System Protection System (NSPS) are supplied from the DC buses via inverters. Typical loads connected to the four divisions of emergency DC power are as follows:

1. Division I
   a. DC Motor Operated Valves in the RCIC System.
   b. The RCIC Gland Seal Compressor
   c. Auxiliary Building and Fuel Building Emergency Lighting
   d. Control Power for Division I AC Switchgear
   e. Feeds for NSPS Division I (includes instrumentation and logic for RCIC, LPCS, LPCI Loop A, and Division I SRVs)
2. Divison II
   a. Radwaste Building and Control Building Emergency Lighting
   b. Control Power for Division II AC Switchgear
   c. Feeds for NSPS Division II (includes instrumentation and logic for RCIC, LPCI Loops B & C and Division II SRVs)

3. Division III

   a. Control Power for Division III AC Switchgear

   b. Feeds for NSPS Division III (includes instrumentation and logic for HPCS)

4. Division IV

   a. Containment Building Emergency Lighting

   b. Feeds for NSPS Division IV (includes instrumentation for HPCS)

2.2 REACTOR/CONTAINMENT HEAT REMOVAL SYSTEMS

The following subsections describe the mechanical equipment and list the associated power supplies available for reactor and containment heat removal.

2.2.1 HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM

The HPCS system consists of a motor driven pump along with associated system piping, valves, controls and instrumentation for providing high pressure makeup water to the Reactor Pressure Vessel (RPV). Along with its divisionally associated electrical supply (Division III) the HPCS takes the place of the turbine driven High Pressure Coolant Injection System existing in some earlier BWR Plant designs. The HPCS with its support equipment is capable of functioning after a loss of all offsite power, and the Division I and II diesel generators.

A Reactor Pressure Vessel low water level or high drywell pressure signal will initiate HPCS and its support equipment. The system can also be placed in operation manually. Water for the HPCS pump suction can be provided from the RCIC storage tank (125,000 gal.) or alternately from the suppression pool. The RCIC storage tank is the normal source. When sensors detect low water level in the RCIC tank or high water level in the suppression pool automatic suction transfer to the suppression pool occurs.

2.2.2 REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

The RCIC system consists of a turbine driven pump along with associated system piping, valves, controls and instrumentation for providing high pressure makeup water to the RPV. Because the RCIC system does not require primary AC power for system operation, RCIC is capable of functioning during station blackout conditions as long as DC power is available for control and reactor pressure is sufficient to drive the RCIC turbine.

A Reactor Pressure Vessel low water level signal automatically initiates the RCIC system. The system can also be placed in operation manually.

Water for the RCIC pump suction can be provided from the RCIC storage tank or from the suppression pool. The RCIC

storage tank is the normal source. When sensors detect low water level in the RCIC tank or high water level in the suppression pool, automatic suction transfer to the suppression pool occurs.

2.2.3 AUTOMATIC DEPRESSURIZATION SYSTEM (ADS)

If the RCIC and HPCS cannot maintain reactor water level, the ADS reduces the reactor pressure so that flow from the low pressure emergency core cooling systems can be provided to the reactor (see the following two subsections). ADS utilizes 7 of the 16 Safety Relief Valves (SRVs) in performing its function. The 7 valves have accumulators that store sufficient air for two actuations of the valves against 70% of the design drywell pressure. (Design drywell pressure is 30 psig.) The accumulators are normally supplied by instrument air, but can be supplied by air bottles which have a 7 day supply of air. The control function for all sixteen safety relief valves is provided from the emergency DC supply.

2.2.4 LOW PRESSURE CORE SPRAY (LPCS) SYSTEM

The LPCS system consists of a motor driven pump along with associated system piping, valves, controls and instrumentation for providing low pressure makeup water to the Reactor Pressure Vessel. In the event the high

pressure RPV water makeup systems cannot maintain water level, the LPCS, operating in conjunction with ADS, can provide additional water to the RPV. The LPCS is supplied from Division I engineered safety feature buses so it is capable of functioning after losses of offsite power and Division II and III diesel generators.

Water for the LPCS suction is provided from the suppression pool.

2.2.5 LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM

LPCI is an operating mode of the Residual Heat Removal System (RHR). All three loops of RHR are capable of performing the LPCI function. In this mode, a motor driven RHR pump takes suction from the suppression pool and injects into the RPV through piping separate from the LPCI piping for the other loops of RHR. In the event that the high pressure RPV water makeup systems cannot maintain water level, LPCI, operating in conjunction with ADS, can provide additional water to the RPV. Loop A of LPCI is supplied from Division I buses. Loops B and C of LPCI are supplied from Division II buses. Each loop of LPCI is capable of functioning with no other primary AC power source available except its divisionally associated diesel generator.

2.2.6 SUPPRESSION POOL COOLING MODE OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM

The A and B loops of RHR are capable of operating in the Suppression Pool Cooling Mode. In this mode, suppression pool water is cooled by the RHR heat exchangers. Heat is rejected into water provided by the Shutdown Service Water System (SX). (The SX system is described in subsection 2.2.8.) Loop A is supplied with cooling water and electrical power from Division I. Loop B is supplied with cooling water and electrical power from Division II.

The RHR system, when operating in this mode, cools the suppression pool and thereby helps maintain the pool's pressure suppression function for the containment.

2.2.7 CONTAINMENT SPRAY MODE OF THE RHR SYSTEM

The A and B loops of RHR are capable of operating in the Containment Spray Mode. In this mode, suppression pool water is cooled by the RHR heat exchangers and is sprayed into the containment air space from spargers located at the top of the containment. The heat exchangers reject heat into water provided by the SX system. Loop A is supplied with cooling water and electrical power from Division I. Loop B is supplied with cooling water and electrical power from Division II.

The RHR system, when operating in this mode, reduces containment temperature and pressure so containment integrity can be maintained during the course of an accident.

2.2.8 SHUTDOWN SERVICE WATER (SX) SYSTEM

The SX system consists of motor driven pumps, piping, valves, controls and instrumentation for providing cooling water to plant emergency equipment. The system is divided into divisions that correspond to the three divisions of emergency AC power. Each SX division serves and is powered by equipment within that division. Typical cooling loads supplied by each division include the following:

1. Division I
   a. Division I Diesel Generator Heat Exchangers.
   b. RHR Loop A Heat Exchanger.
   c. Area Coolers for the LPCS Pump Room, the RHR Loop A Pump Room, the RHR Loop A Heat Exchanger Room and the RCIC Pump Room.
   d. Division I Switchgear Heat Removal.
   e. RHR Loop A Pump Seal Cooler.
2. Division II
   a. Division II Diesel Generator Heat Exchanger.
   b. RHR Loop B Heat Exchanger.
   c. Area Coolers for the RHR Loop B and Loop C Pump Room and the RHR B Heat Exchanger Room.

d. Division II Switchgear Heat Removal.

e. RHR Loop B and C Pump Seal Coolers.

3. Division III

a. Division III Diesel Generator Heat Exchanger.

b. Area Cooler for the HPCS Pump Room.

c. Division III Switchgear Heat Removal.

2.3 INSTRUMENTATION FOR PLANT PROTECTIVE ACTIONS

In addition to the instruments used to control or monitor the proper functioning of the equipment listed in subsections 2.1 and 2.2, the following instrument indications are particularly useful to the plant operator for responding to transients where adequate core cooling or containment integrity could be in jeopardy:

-Control Rod Position Indication -
This indication displays the position of each control rod and can be used to confirm that a reactor scram has properly occurred.

-Reactor Water Level -
This indication provides conclusive evidence that adequate core cooling is being maintained. Adequate

core cooling is assured at decay heat power levels, as long as water level remains above the top of active fuel.

-Reactor Pressure -

This indication, in conjunction with suppression pool temperature, is used to determine when actions should be taken to prevent damage to containment structures from safety relief valve blowdowns. It, along with drywell or containment temperature, is used to determine if flashing errors can occur in the RPV water level indication.

-Containment Pressure -

This indication is used to determine when action steps such as initiation of containment sprays, reactor vessel depressurization/flooding or containment venting should be performed to prevent loss of containment integrity.

-Containment Temperature -

This indication, in conjunction with reactor pressure, is used to determine if flashing errors can occur in the RPV water level indication. This instrumentation also provides a measure of containment integrity through its use in the Emergency Operating Procedures.

-Drywell Pressure -

This indication can be used to detect leakage of reactor coolant into the drywell.

-Drywell Temperature -

This indication, in conjunction with reactor pressure, is used to determine if flashing errors can occur in the RPV water level indication. This instrumentation also provides a measure of Drywell integrity through its use in the Emergency Operating Procedures.

-Suppression Pool Water Level -

This indication, in conjunction with reactor pressure, is used to determine if containment structural loads are approaching design limits under abnormal conditions (e.g. containment flooding).

-Suppression Pool Water Temperature -

This indication provides a measure of the steam quenching capability of the suppression pool and, along with reactor pressure, is used to determine when actions should be taken to prevent damage to containment structures from safety relief valve blowdowns.

The instrument loops providing the above indications are powered by different combinations of AC and DC power. A list of important instruments available under SBO conditions is provided in Subsection 3.1.2.

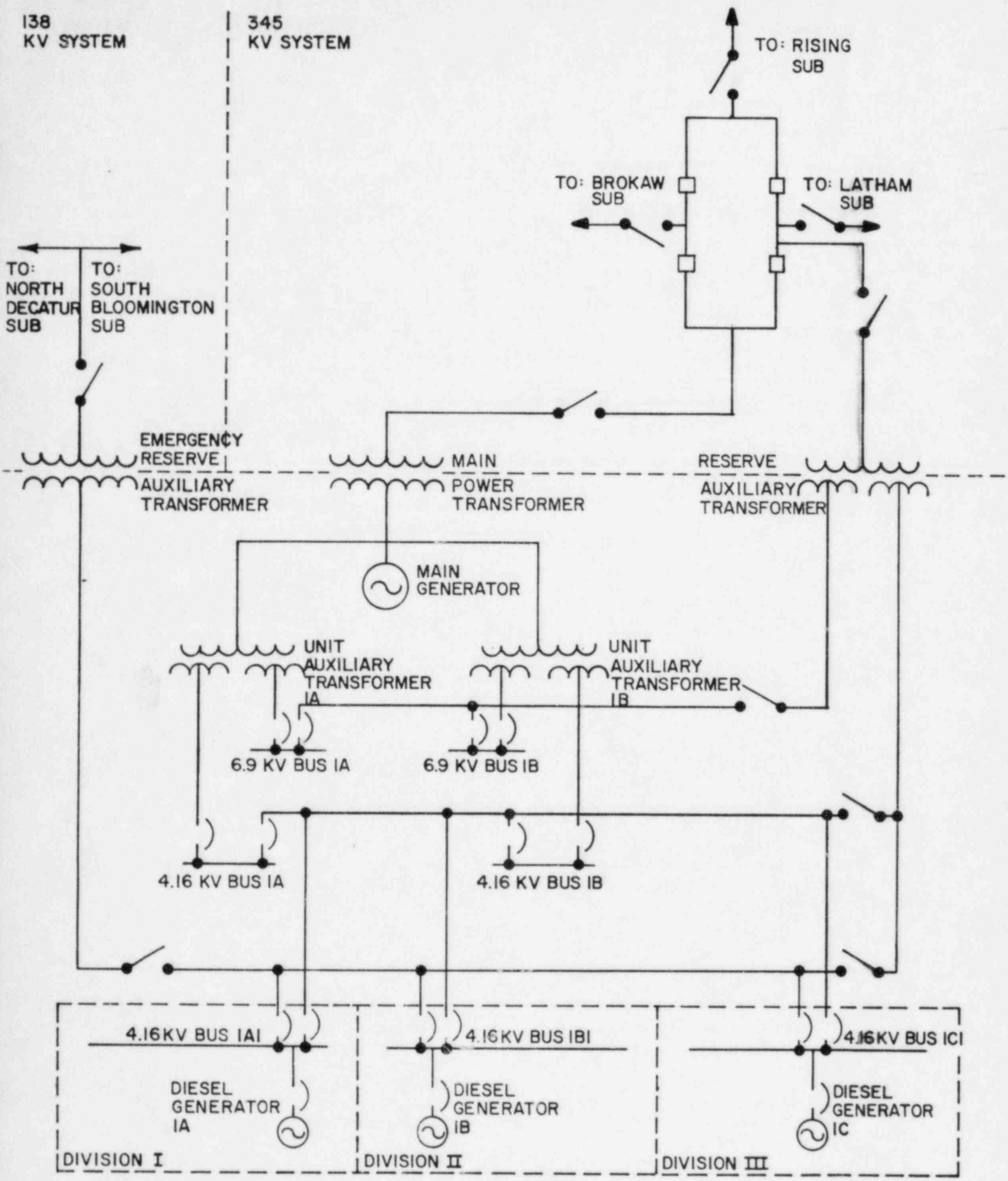

FIGURE 2-1 SINGLE LINE DRAWING OF THE AC POWER SUPPLY FOR CLINTON POWER STATION

3.0 EVALUATION OF CPS RESPONSE TO A STATION BLACKOUT

The following subsections describe the inputs to, and evaluate the results of, the analysis performed to determine the response of Clinton Power Station to a Station Blackout Event.

3.1 DISCUSSION OF ASSUMPTIONS

The initial conditions and equipment availability assumed for purposes of the SBO analysis are described in the next two subsections. The operational response strategy proposed for SBO, which is incorporated in the analytical models for the analysis computer codes, is described in subsection 3.3.

3.1.1 INITIAL CONDITIONS

The impact of an SBO on the plant is dependent in part on the operating state at the time of occurrence. Three SBO scenarios were analyzed:

1. Station Blackout from 100% power with no additional failures,
2. Station Blackout from 100% power with RCIC failure,
3. Station Blackout from 100% power with a stuck open relief valve (SORV).

All of these scenarios involve the occurrence of Station Blackout from 100% power with an equilibrium core. This is the only mode of plant operation that will be considered since decay heat levels are greatest under this condition. (Even the case of Station Blackout occurring with the vessel head removed, in which case RCIC would not be available, is less restrictive than these cases. If the vessel head were removed as early as 10 hours after plant scram and the Station Blackout were to occur at this time, it would require in excess of 5 hours for the vessel water inventory to heat up and boil off to the level of the top of active fuel. This number is based upon the assumption of previous full power operation with an equilibrium core, and vessel inventory being at the top of the steam dryer.)

Plant conditions assumed for these analyses, in general, bound those expected to be encountered under normal plant operation. In many cases the limiting values allowed by the CPS Technical Specifications were used.

3.1.2 EQUIPMENT AVAILABILITY

Only equipment which does not depend upon AC power (with the exception of those items supplied from battery-inverter arrangements) can be used to maintain the plant in a safe condition after occurrence of an SBO.

The following is a list of major systems/equipment that would be unavailable under SBO conditions:

- Feedwater
- High Pressure Core Spray

  Because HPCS is supplied from an independent AC power supply (the Division III diesel) it has a similar level of independence as HPCI in earlier BWR designs. Because these plants typically assume the availability of HPCI under SBO conditions, assuming loss of HPCS at CPS is a conservatism in the CPS analysis.
- Low Pressure Core Spray
- Low Pressure Coolant Injection
- Drywell Cooling
- Suppression Pool cooling mode of RHR
- Cooling to the RCIC Equipment Areas
- Containment Spray Mode of RHR
- Shutdown Service Water

The following is a list of equipment/functions that would be available under SBO conditions. Restrictions on their use are also described:

- <u>Reactor Core Isolation Cooling</u>

  RCIC injection flow will be available to the reactor pressure vessel as long as there is sufficient reactor pressure to drive the RCIC turbine, battery power is available for control, and injection water temperature remains low enough to provide adequate

cooling for RCIC lube oil. The RCIC system is capable of providing water flow to the reactor at a rate of 600 gpm at reactor pressures greater than 178 psig. An RCIC flow indicator located in the main control room remains available after a Station Blackout. RCIC flow can be maintained at reduced rates down to reactor pressures as low as 50 psig. The maximum RCIC injection water temperature allowed for this analysis is 140°F. This restriction is listed in the vendor manual for RCIC and is intended to provide adequate cooling for the RCIC lube oil. Actual operating experience with turbines of this type indicates that operability can be maintained to lube oil temperatures in excess of 180°F. Transfer of RCIC suction from the RCIC storage tank to the suppression pool both manually and automatically is possible after a Station Blackout.

- Safety Relief Valves

  SRVs will be available for reducing reactor pressure during a Station Blackout Event. The valves can be opened in the relief mode, the manual relief modes, the safety mode, the Automatic Depressurization System (ADS) mode, or in the manual ADS mode. All these modes, with the exception of the safety mode, require accumulator air pressure for valve actuation and DC power for valve actuator control. The safety mode does not require accumu-

lator air or DC power because reactor pressure provides the motive force for opening of the valves in this mode. The safety mode serves as a backup to the relief mode in providing the reactor with over-pressure protection.

Accumulators for the SRVs are normally supplied with air from the instrument air system. Instrument air would be unavailable during Station Blackout because the service air compressors which supply the instrument air system are powered from AC sources.

Accumulators for the SRVs are capable of providing at least 1 actuation of these valves against normal drywell pressure when in the relief mode. An ADS accumulator low pressure alarm located in the Main Control Room remains available after a Station Blackout.

Backup air bottles exist to supply the accumulators for ADS valves. The motor operated valves that need to be opened to make this supply available are powered by AC buses, but can be manually opened.

Although direct SRV position indication is not available during an SBO, it is possible to

infer if an SRV is open for an extended period of time by increased temperatures in the quadrant of the suppression pool associated with the open SRV.

- <u>Control Rod Position Indication</u>

  Direct readout of control rod position is not possible after Station Blackout. However, control rod position can be determined on a rod by rod basis by attaching portable resistance meters to jacks on a backrow panel in the main control room. Average Power Range Monitor (APRM) offscale low readings would provide the operator with initial indication that a reactor scram had occurred.

- <u>RPV Water Level Indication</u>

  This indication is available in the Main Control Room after a Station Blackout.

- <u>Reactor Pressure Indication</u>

  This indication is available at the analog trip system display on the NSPS panel. It is also available at the remote shutdown panel after a Station Blackout.

- Containment Pressure Indication

  This indication is available at the analog trip system display on the NSPS panel after a Station Blackout.

- Containment Temperature Indication

  Direct readout of containment temperature is not possible after Station Blackout. However, containment temperature can be determined using a portable resistance meter attached to connections within the control room.

- Drywell Pressure Indication

  This indication is available at the analog trip system display on the NSPS panel after a Station Blackout.

- Drywell Temperature Indication

  Direct readout of drywell temperature is not possible after Station Blackout. However drywell temperature can be determined using a portable resistance meter attached to connections within the control room.

- Suppression Pool Water Level Indication

  Direct readout of suppression pool water level is not possible after Station Blackout. However,

suppression pool level can be determined using a 24 volt DC supply and a voltmeter attached to connections within the control room.

- Suppression Pool Water Temperature Indication
  Direct readout of suppression pool water temperature is not possible after Station Blackout. However, suppression pool water temperature can be determined using a portable resistance meter attached to connections within the control room.

- Battery Depletion Status Indication
  Battery terminal voltage, which is indicative of the battery depletion level is provided in the main control room for all four divisions of emergency DC power.

3.2 PLANT LIMITS FOR SAFETY

The following subsections describe plant limits for safety. These parameters, when exceeded, indicate a significant reduction in the ability of the plant to protect fuel or containment integrity. The limits are used to determine the amount of time Clinton Power Station can safely withstand a Station Blackout.

3.2.1 REACTOR WATER LEVEL

Reactor water level provides an indication of the state of fuel cooling. Inadequate core cooling and possible fuel damage can not occur after reactor shutdown as long as reactor water level remains above the Top of Active Fuel (TAF) which is about 200 inches below normal water level. Adequate core cooling can be maintained at lower reactor water levels, but depends upon steam cooling of the fuel. Therefore, unless adequate steam cooling can be confirmed, TAF is used as the reactor low water level plant safety limit.

Entry of reactor water into the main steam lines could ultimately result in damage to the RCIC turbine, steam line and SRVs. Therefore, Level 8, which is about 20 inches above normal water level, is used as the reactor high water level plant safety limit.

3.2.2 REACTOR PRESSURE

A reactor pressure of 1375 psig, which is 110% of design pressure, is used as the reactor high pressure plant safety limit. This is the maximum pressure allowed under upset conditions by the ASME Boiler and Pressure Vessel Code.

3.2.3 SUPPRESSION POOL TEMPERATURE

The limiting suppression pool temperatures would be defined by the Heat Capacity Temperature Limit Curves (HCTLC) for the suppression pool. The HCTLC for Clinton are shown in figure 3-1. The area below the HCTLC defines reactor pressure and suppression pool temperature conditions during which SRV discharge can occur without exceeding SRV quencher or containment load limits. The HCTLC are used to define the suppression pool high temperature plant safety limit. This is a conservative limit and plant damage is not realistically expected if the limit is exceeded at low RPV pressures.

3.2.4 SUPPRESSION POOL WATER LEVEL

The Suppression Pool Load Limit Curve (SPLLC) defines reactor pressure and suppression pool level conditions during which SRV discharge can occur without causing damage to containment structural components. The SPLLC for Clinton is shown in figure 3-2. This curve is used to define the suppression pool high water level plant safety limit.

3.2.5 CONTAINMENT PRESSURE

The design pressure for the CPS containment is 15 psig.

This value will be used as the containment high pressure plant safety limit, although structural failure of the containment would occur at a much higher pressure.

3.2.6 CONTAINMENT TEMPERATURE

The environmental qualification envelope used to qualify containment equipment includes the assumption of 185 °F temperatures for a period of one day to account for Loss of Coolant Accident conditions. This limit, if not exceeded, will provide assurance that safety related equipment will be functional after the reestablishment of AC power systems. Therefore, it is used as the high containment temperature plant safety limit.

3.2.7 DRYWELL PRESSURE

The drywell design pressure is 30 psig. This value is high enough to prevent drywell structural damage after large high energy linebreaks occur in the drywell. This value is used as the high drywell pressure plant safety limit.

3.2.8 DRYWELL TEMPERATURE

The environmental qualification envelope used to qualify drywell equipment includes the assumption of 250 °F or higher temperatures for a period of one day to account for

Loss of Coolant Accident Conditions. This limit, if not exceeded, will provide assurance that safety related equipment will be functional after the reestablishment of AC power systems. Therefore, 250°F is used as the high drywell temperature plant safety limit.

3.2.9 RCIC ROOM TEMPERATURE

The environmental qualification envelope used to qualify equipment in the RCIC pump room and the RCIC instrument panel room includes the assumption of 153 °F or higher temperatures for a period of 6 hours to account for high energy line break conditions. It is expected that this temperature would be acceptable for a period in excess of eight hours, because the actual qualification envelope includes temperatures in excess of 200 °F for over an hour. Therefore, 153 °F is used as the RCIC room high temperature plant safety limit.

3.2.10 MAIN CONTROL ROOM TEMPERATURE

No specific limit for control room temperature has been established for evaluations of the SBO event. The SBO event results in loss of power to all equipment in the control room except that supplied from DC or uninterruptible power sources and much of this would be switched off very early in the transient to conserve

battery life. The major heat sources, therefore, are the sensible heat from control room equipment and the biological heating from the room occupants.

Control room temperature and humidity can be moderated via natural circulation pathways by opening the access doors to the control room.

3.2.11 PLANT BATTERY DEPLETION

The plant safety limit used to define when emergency batteries are depleted is battery terminal voltage falling to 105 volts DC (84% of the design value of 125 volts DC). This voltage is sufficient to operate all the required components attached to the associated DC busses.

3.3 OPERATIONAL RESPONSE STRATEGY FOR STATION BLACKOUT (SBO)

The following are proposed action steps to be taken by the plant operators under Station Blackout conditions. These are the operational responses assumed for purposes of analysis. Actual responses adopted in CPS procedures may be slightly different than these.

1. If the plant is at power operation when the Station Blackout occurs, the operator would first confirm that the reactor had scramed. Preliminary indication of this would be given by

an Average Power Range Monitor offscale low reading. Confirmation that each control rod is fully inserted could be made by connecting a resistance meter to each individual control rod position indication wire located in the control room.

2. Attempts would be initiated to reestablish offsite power and/or start a diesel and connect it to its associated bus.
3. RCIC flow would be confirmed to exist. (RCIC initiates upon Reactor Low Water level, Level 2) If the RCIC system is not running, it would be manually initiated. (RCIC isolation signals, such as RCIC area high temperature isolation, may need to be overriden.)
4. The valves that need to be opened to make the backup air supply available to the ADS-SRVs will be manually opened.
5. If the RCIC system is confirmed to be operating properly, an ADS-SRV would be manually opened to depressurize the reactor to a pressure somewhat above the low pressure constant flow operating limit for RCIC. Several advantages exist with this operational approach, as follows:
   A. The number of challenges to SRVs can be reduced, because depressurization of the reactor will prevent multiple openings of

SRVs in the relief or safety mode. A reduction in the number of SRV challenges will reduce the probability of a stuck open relief valve (SORV).

B. Reactor depressurization will reduce the temperature of the reactor, and thus the expected heat loads in the drywell. This will limit the peak temperatures experienced by the drywell.

C. Further depressurization of the reactor from this reduced pressure will be faster than depressurization from rated reactor pressure and will result in smaller containment loadings. (Depressurization may be required to use low pressure water makeup systems if they become available.)

6. RCIC pump suction will be switched to the suppression pool early in the event to use the suppression pool water before it heats up excessively. This will preserve the cooler RCIC storage tank water for use later in the event. In this way lube oil temperature problems are delayed and RCIC operability is maintained for the greatest length of time.

RCIC pump suction would be switched back to the RCIC tank when suppression pool temperature exceeds 140°F. (Note: This is the temperature

switchover used for purposes of analysis. Plant procedures may be different than this because of higher permissible lube oil temperatures. See Subsection 3.1.2) Pump suction will automatically be returned to the suppression pool when RCIC tank water is exhausted.

7. If AC power can not be reestablished within approximately 1 hour, loads on the batteries should be shed to maximize the length of time DC power will be available for essential control functions.

The actions taken if additional failures occur may be different than those listed above. The other two specific scenarios evaluated are discussed in the following paragraphs.

If the RCIC system is unavailable, manual depressurization of the reactor would not be initiated. This would preserve water inventory in the vessel for the greatest period of time. If AC power or RCIC flow cannot be reestablished quickly, attempts would be made to align fire protection water (provided by the diesel driven fire pump) to the reactor vessel. When fire protection water is available, the reactor would be depressurized via ADS to allow maximum flow from this low pressure injection source. If reactor water level falls below top of active fuel

before fire protection water becomes available, core cooling can be maintained for a short period of time using the Steam Cooling Contingency of the CPS Emergency Procedures.

If an SORV occurs during an SBO, the operator would not initiate manual reactor depressurization because depressurization would occur through the SORV. If AC power cannot be reestablished attempts would be made to align fire protection water. Before reactor pressure falls below the minimum required for RCIC operation the operator would raise the vessel water level to slightly below level 8 to maximize the vessel inventory and the time core cooling can be maintained without injection flow. When water level falls to top of active fuel, fire protection water would be provided to the vessel if available.

3.4 EVALUATION OF PLANT RESPONSE

Plant responses have been determined analytically for Station Blackouts occurring at full power operation with an equilibrium core. The following scenarios were analyzed.

1. Station Blackout with no additional failures
2. Station Blackout with RCIC failure
3. Station Blackout with a Stuck Open Relief Valve

Plant responses for these scenarios were determined analytically using the assumptions from section 3.1 and the operational response strategy from section 3.3. Reactor responses were determined using the computer code RETRAN, with a simple thermal hydraulic model of the CPS primary system. RETRAN took into account SRV flow, RCIC flow and core heat inputs in determining reactor responses. Suppression pool, containment and drywell responses were determined using an enhanced version of the computer code COMPARE. COMPARE took into account heat transfer from drywell and containment heat structures as well as steam flows through the SRVs in determining suppression pool, containment and drywell responses. The plant responses are described in the following subsections and are compared to the plant limits for safety from section 3.2.

3.4.1 SBO AT FULL POWER WITH NO ADDITIONAL FAILURES

Station Blackout would result in reactor scram, reactor recirculation pump coastdown and main steam isolation valve (MSIV) closure. Due to MSIV closure, reactor pressure would increase rapidly causing SRVs to lift. SRV low-low setpoint logic would be initiated and the low-low setpoint valve would cycle open and close until its accumulator air is depleted, or until the operator manually opens one SRV to depressurize the vessel. If accumulator air for SRV

operation is depleted, some SRVs will open in the safety mode to protect the reactor vessel from over-pressurization.

Reactor water level would drop rapidly due to void collapse after the reactor scram and MSIV closure, loss of the feedwater pumps and steam inventory loss through SRVs. The RCIC system would be actuated when reactor water level falls to Level 2.

Once RCIC is confirmed to be running, the reactor would be depressurized using an ADS SRV. The backup air supply to the ADS accumulators would be lined up to ensure that this is possible. When reactor pressure falls close to the minimum value assumed for RCIC operation (178 psia), the operator would maintain pressure somewhat above this value by manually closing and opening SRVs.

Reactor water level would continue to drop until RCIC flow exceeds boil-off losses. Water level will then gradually increase up to the normal water level. The operator would then control RCIC to maintain water level within the normal range.

Suppression Pool temperature would be rising due to SRV discharges. RCIC pump suction would be switched from the RCIC storage tank to the suppression pool early in the

event, back to the storage tank when suppression pool water temperature exceeds 140°F; and back again to the pool when the RCIC storage tank is depleted.

Drywell, Containment and RCIC room air temperature will rise due to loss of cooling in these areas.

An event sequence for a Station Blackout with no additional failures is shown in Table 3-1. Response curves showing reactor water level, reactor pressure, drywell temperature and suppression pool temperature as a function of time are contained in figures 3-3 through 3-6 respectively.

RCIC pump room and instrument room temperatures as a function of time are shown in figure 3-7. These curves were determined using the RATT computer code and modeling assumptions that bound all three event scenarios. RATT modeled the heat structures and heat loads in these rooms in determining their temperatures. Heat inputs in the rooms include the RCIC pump, the RCIC turbine and DC electrical components. Cooling for the rooms is not available because of loss of AC power.

Calculations were made assuming a variety of load shedding schemes to determine battery life under Station Blackout conditions. The calculations, performed using the methodology in IEEE 485, are intended to be bounding for

all three scenarios. One set of these calculations shows that for the most limiting division of emergency DC power, battery life can be extended to approximately 7.5 hours, assuming that those DC loads not essential for responding to the Station Blackout event are shed after 1 hour. Essential SBO loads left on after 1 hour include those required to maintain RCIC operable, control SRVs, retain needed instrumentation, and retain switchgear control. The last load is intended to ensure proper control of AC systems when AC power is reestablished.

The first plant safety limit exceeded during this scenario is the suppression pool heat capacity temperature limit. This value was conservatively estimated to be exceeded at 2.4 hours after the occurrence of the SBO event. Because of the conservatism associated with the extrapolation used to determine the time the suppression pool HCTL is exceeded, and because plant damage is not expected when the limit is exceeded at low reactor pressures, it is anticipated that the plant can in reality be maintained in a safe condition for at least a 4 hour period. Table 3-2 shows the estimated times when plant safety limits are exceeded.

3.4.2 SBO AT FULL POWER WITH RCIC FAILURE

Loss of RCIC would leave no means of high pressure

injection to the reactor. As noted in the operational response strategy to SBO in subsection 3.4 the operator would not manually depressurize the reactor as he would in the case of SBO occurrence with no additional failures. The objective in this case is to maintain core coverage for the greatest period of time. SRVs would lift in the relief mode to prevent reactor overpressurization.

Analysis indicates that reactor water level outside of the shroud would reach top of active fuel at approximately 15 minutes into the Station Blackout. The reactor water level response is shown in figure 3-8. Studies indicate that adequate core cooling can be maintained down to a vessel water level of approximately 3.5 feet above bottom of active fuel as long as steam flow through the core (such as would result from intermittent SRV lifts) occurs (Reference 7). For this case it is estimated that water coverage will reach this point approximately 28 minutes into the event. Core cooling can be maintained for a longer time using the Steam Cooling Contingency of the CPS Emergency Procedures.

If makeup water flow to the reactor cannot be established within 30 minutes into the event, core damage may occur.

3.4.3 <u>SBO AT FULL POWER WITH A STUCK OPEN RELIEF VALVE (SORV)</u>

With an SORV, a depressurization similar to that which would be performed manually would occur. With RCIC available, the plant behavior during the early stages of the event would be virtually identical to the case of SBO occurring at full power with no additional failures.

At approximately 83 minutes into the event reactor pressure would fall to 178 psia. For purposes of this analysis 178 psia was assumed to be the minimum pressure at which RCIC is capable of functioning. For a time reactor pressure fluctuated around 178 psia, during which RCIC was assumed to operate intermittently. In reality it could be operated at reduced flows continuously to much lower reactor pressures.

Reactor water level would decrease due to continued inventory loss through the SORV and incomplete makeup from the RCIC system. Water level would fall to top of active fuel at approximately 2.9 hours after the occurrence of SBO. Reactor water level for this case is shown in figure 3-9.

Similar to the case of the SBO occurrence with no additional failures the heat capacity temperature limit for the suppression pool is estimated to be exceeded at 2.5

hours into the event. This estimate is conservative and the actual time until the suppression pool HCTLC is exceeded is expected to be longer.

3.5 SUMMARY OF CPS STATION BLACKOUT COPING CAPABILITY

As is shown in subsections 3.4.1 and 3.4.3, in situations where SBO occurs during full power operation and RCIC remains available, the plant can be maintained in a safe condition (where safety is defined by the plant safety limits) for a period in excess of 2.4 hours. The suppression pool heat capacity temperature limit is the limit exceeded at this time.

If RCIC is not available for reactor coolant makeup as is discussed in subsection 3.4.2, the first plant safety limit that would be exceeded is reactor water level. Level would reach top of active fuel approximately 15 minutes into the event. Core cooling can be maintained for a longer period because of steam cooling of the fuel assemblies. If makeup flow to the reactor can not be reestablished at approximately 30 minutes into the event, core damage may be expected.

TABLE 3-1

Event Sequence for Station Blackout at

Full Power with No Additional Failures

	Calculated Time (sec)
Start of Transient and MSIV closure, scram, FW Isolation	0
MSIV closure complete	3.5
SRV operation on high reactor pressure	3 - 55
Vessel level falls to Level 2	85
RCIC system begins operation	115
One SRV manually opened	600
RPV Pressure reaches 178 psia	5081
RPV Pressure maintained around 178 psia due to equilibrium between RCIC inflow, SRV outflow and decay heat generation	5082 - 9590
Suppression Pool Temperature Reaches HCTL	8645

TABLE 3-2

Estimated Times At Which Plant Safety Limits Are Exceeded For Station Blackout At Full Power With No Additional Failures

Plant Safety Limit	Time When Limit Exceeded (hr)
Reactor Water Level	>8
Reactor Pressure	>8
Suppression Pool Temperature	2.4
Suppression Pool Water Level	Never Exceeded
Containment Pressure	>4
Containment Temperature	>4
Drywell Pressure	>4
Drywell Temperature	>4
RCIC Room Temperature	>8
Plant Battery Depletion	7.5

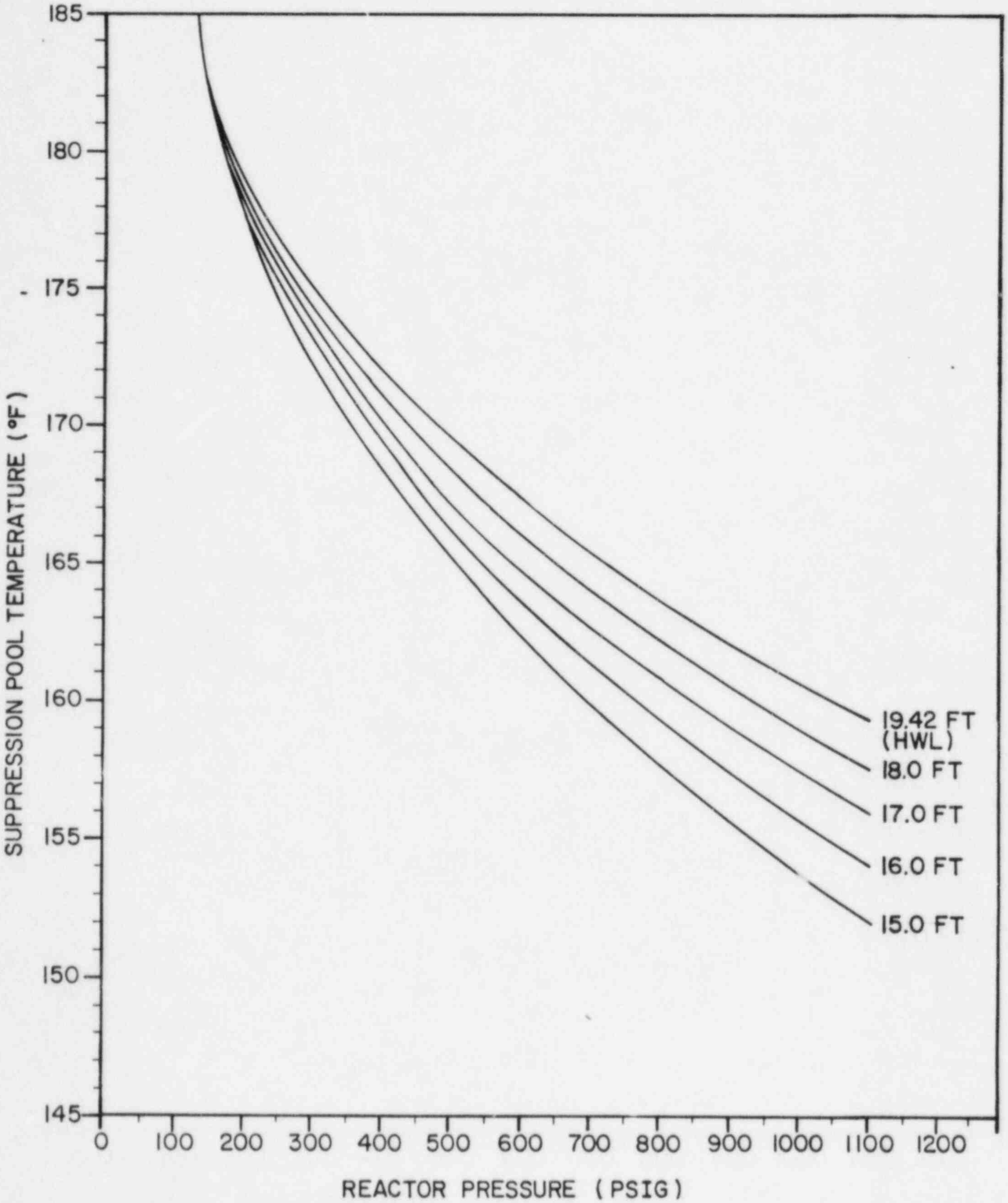

FIGURE 3-1: SUPPRESSION POOL HEAT CAPACITY TEMPERATURE LIMIT CURVES (HCTLC)

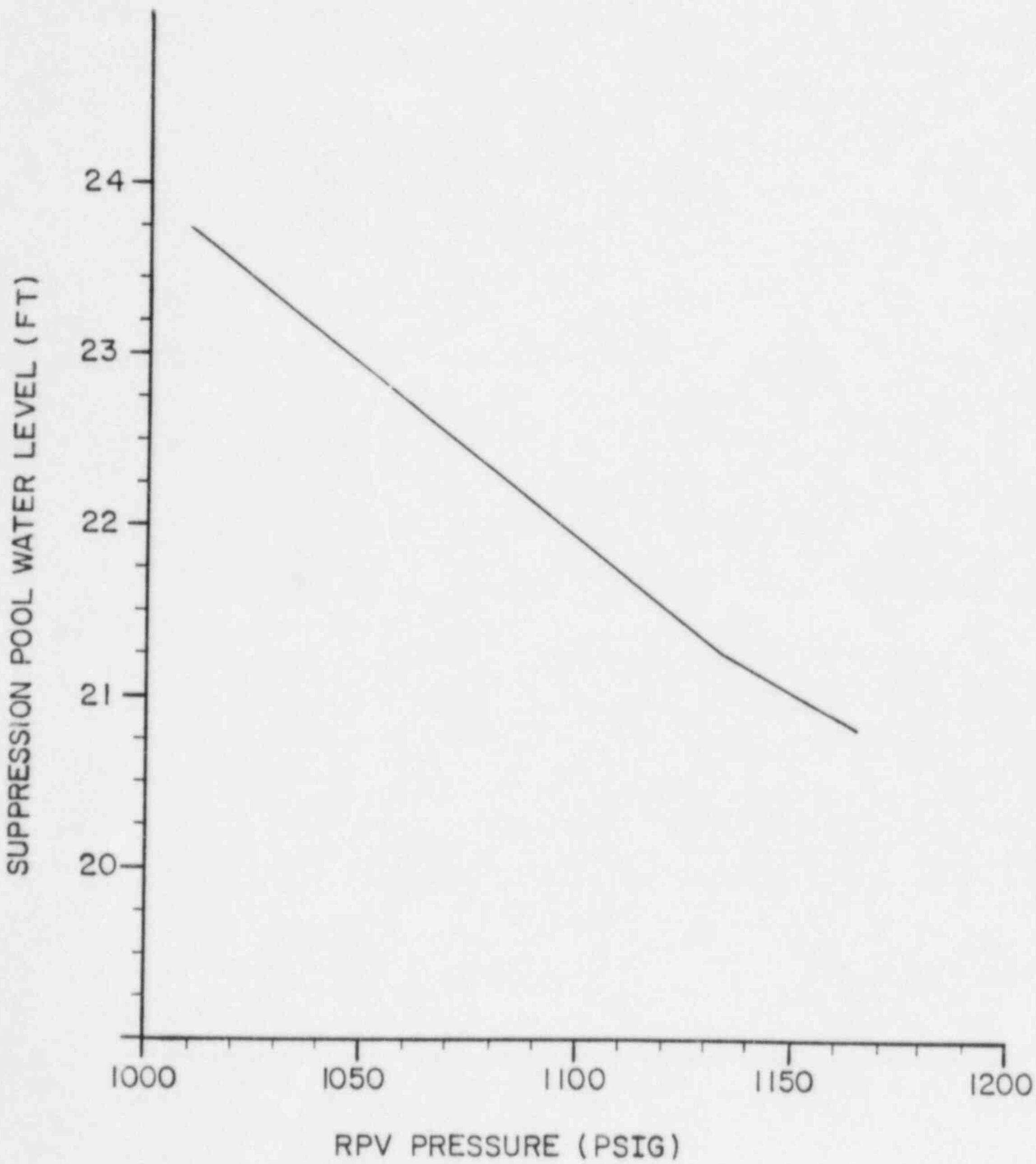

FIGURE 3-2: SUPPRESSION POOL LOAD LIMIT CURVE

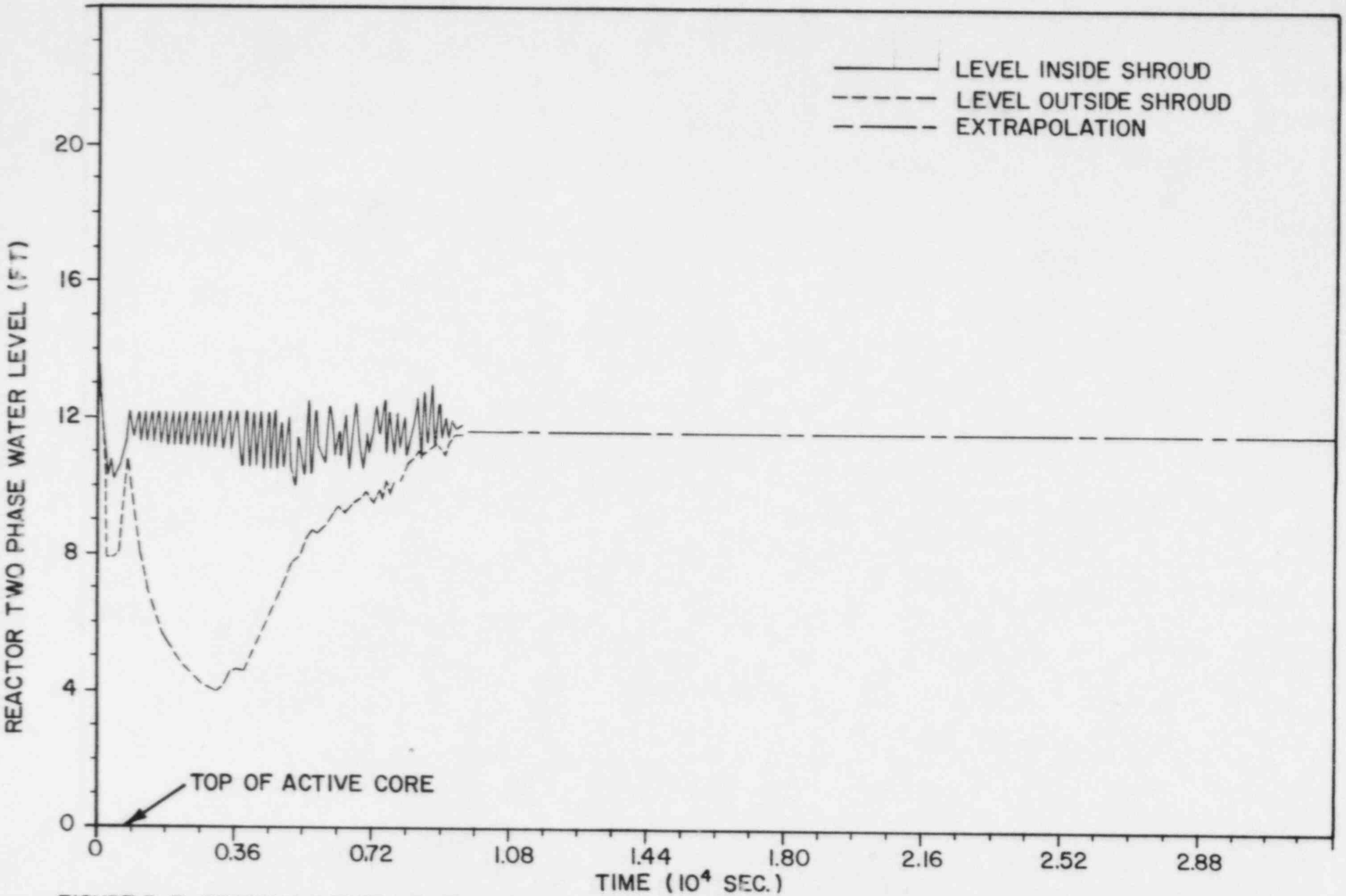

FIGURE 3-3: REACTOR WATER LEVEL RESPONSE FOR AN SBO AT FULL POWER WITH NO ADDITIONAL FAILURES

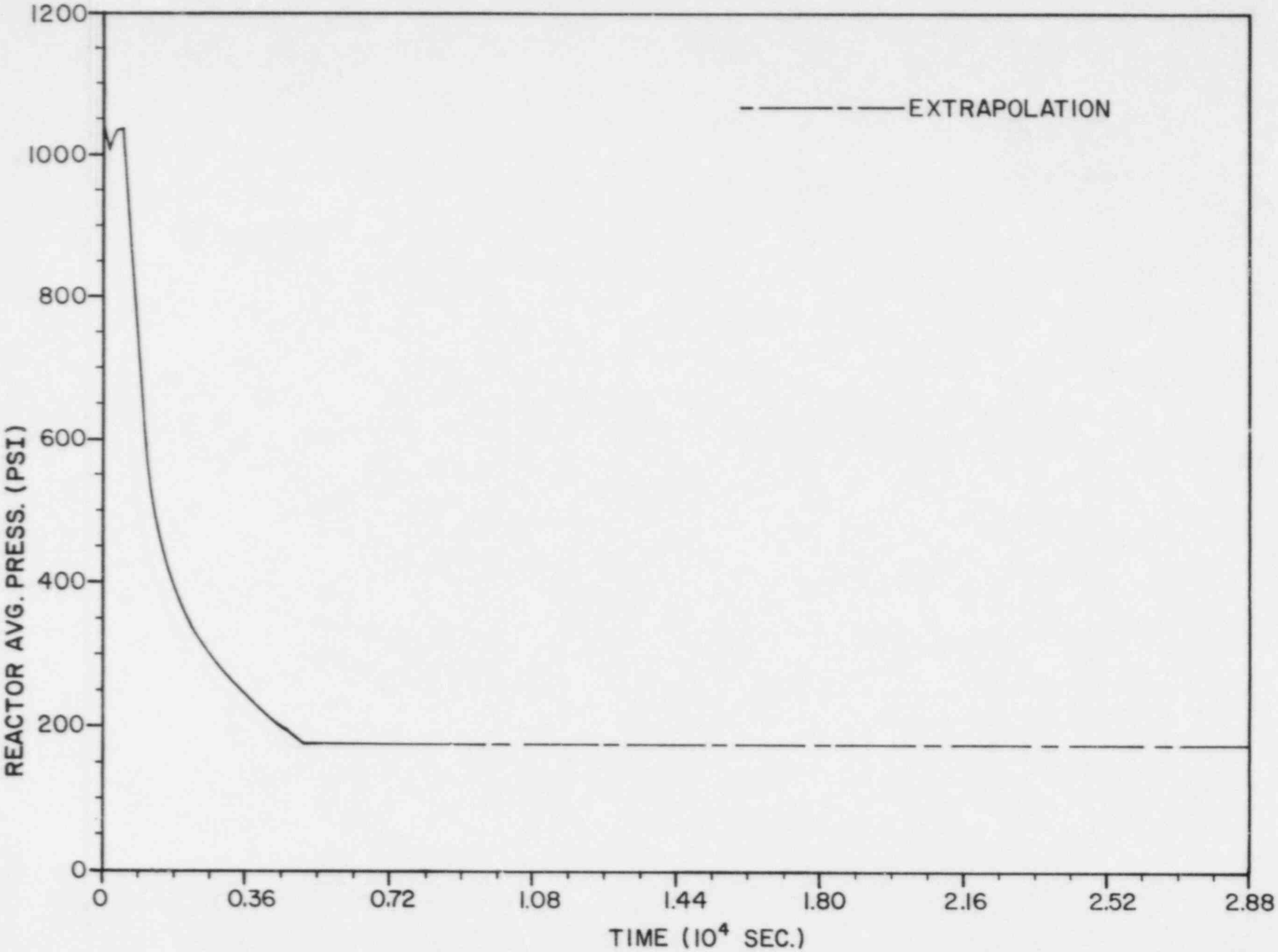

FIGURE 3-4: REACTOR PRESSURE RESPONSE FOR AN SBO AT FULL POWER WITH NO ADDITIONAL FAILURES

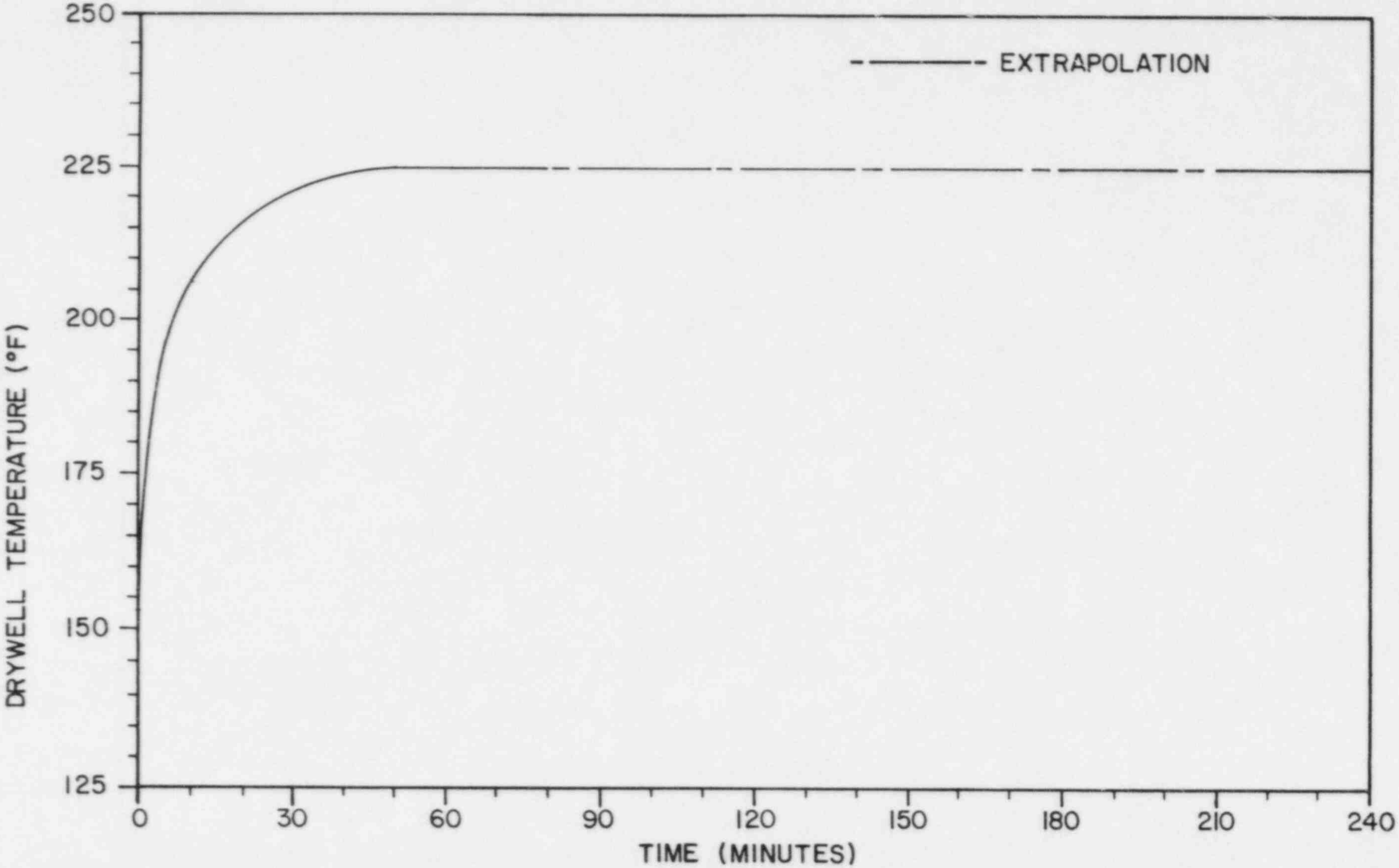

FIGURE 3-5 DRYWELL TEMPERATURE RESPONSE FOR AN SBO AT FULL POWER WITH NO ADDITIONAL FAILURES

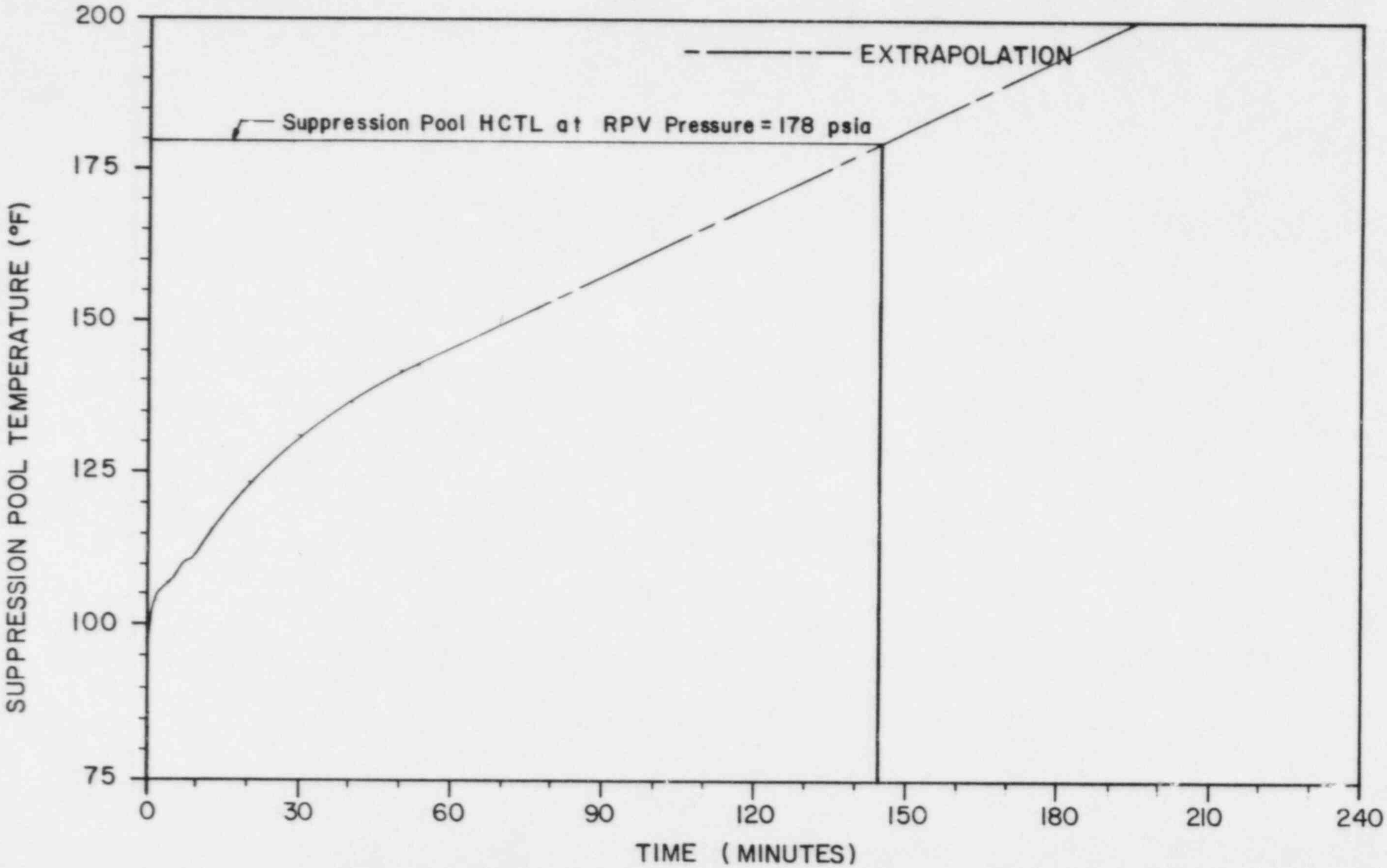

FIGURE 3-6: SUPPRESSION POOL TEMPERATURE RESPONSE FOR AN SBO AT FULL POWER WITH NO ADDITIONAL FAILURES

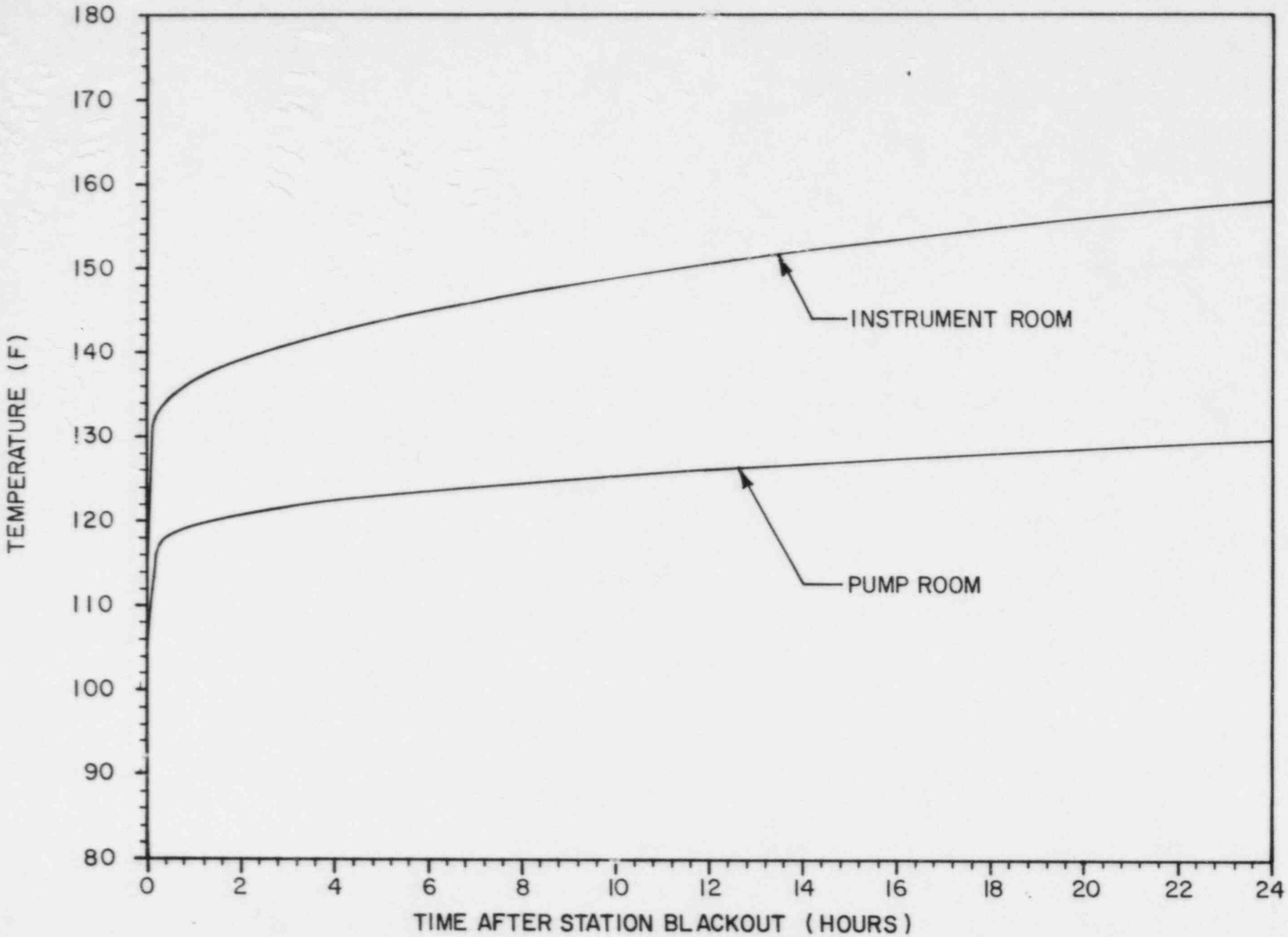

FIGURE 3-7: RCIC PUMP ROOM AND INSTRUMENT ROOM TEMPERATURES AS A FUNCTION OF TIME

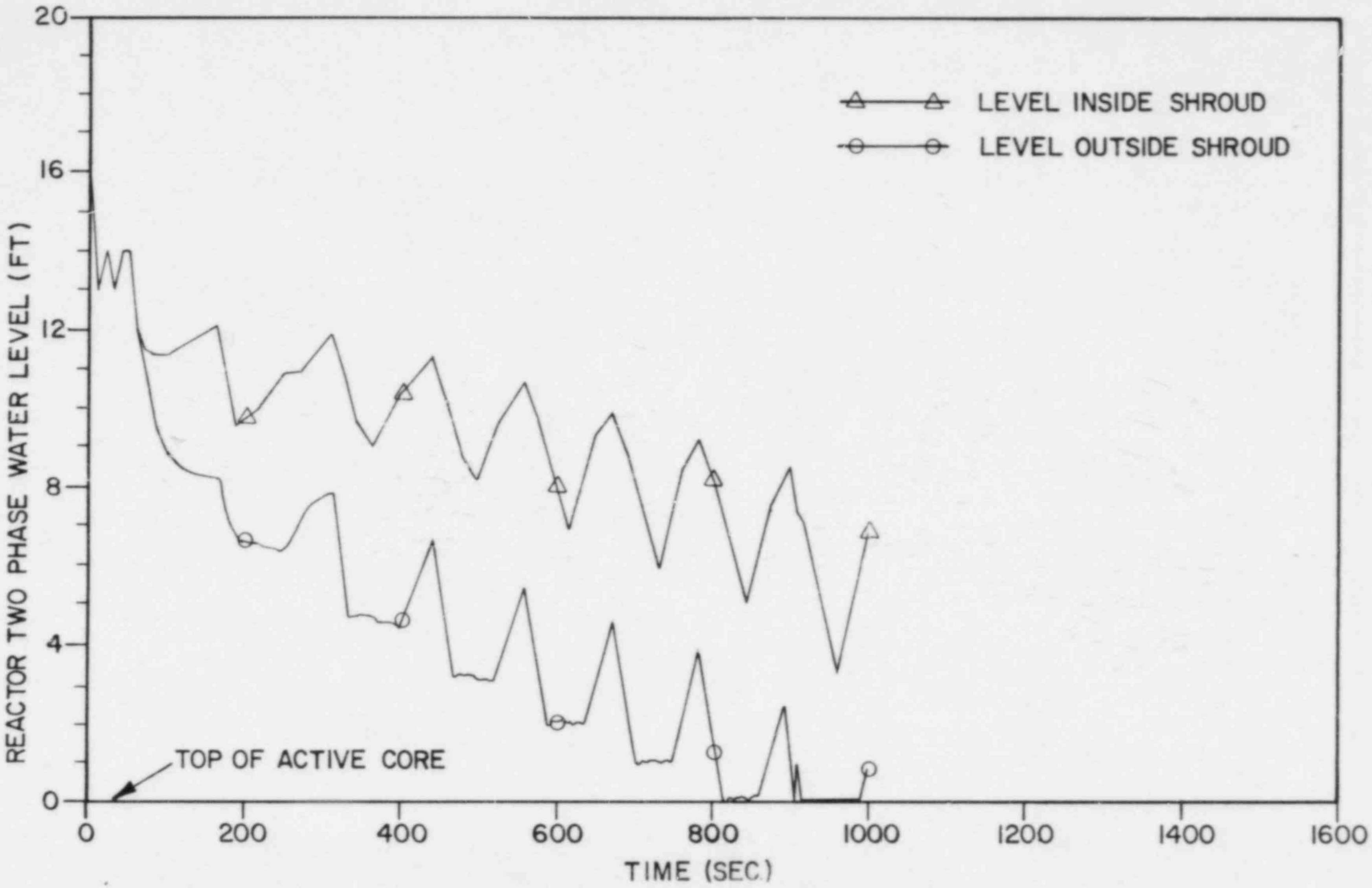

FIGURE 3-8: REACTOR WATER LEVEL RESPONSE FOR AN SBO AT FULL POWER WITH RCIC FAILURE

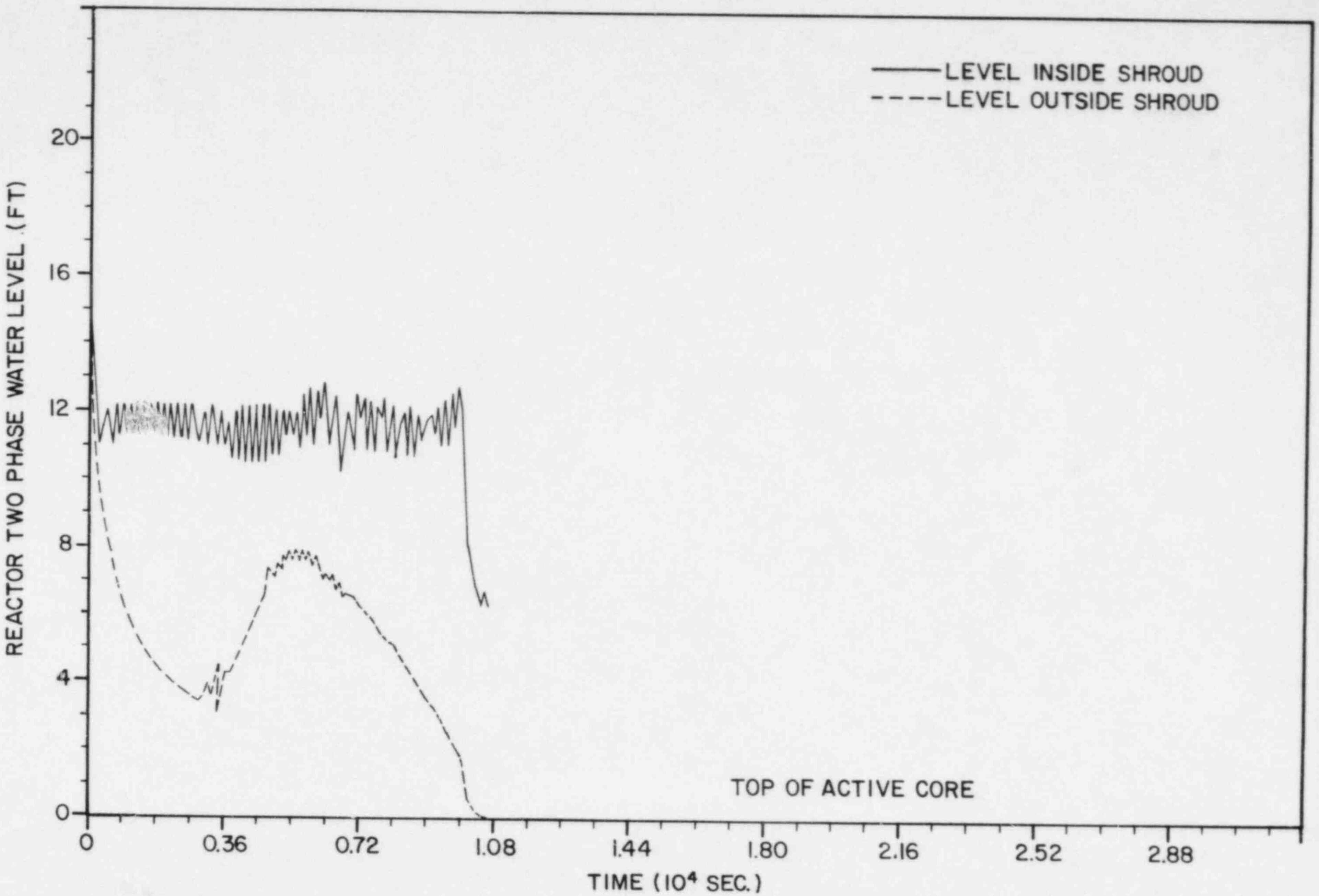

FIGURE 3-9: REACTOR WATER LEVEL RESPONSE FOR AN SBO AT FULL POWER WITH A STUCK OPEN RELIEF VALVE (SORV)

4.0 EVALUATION OF STATION BLACKOUT TESTING

This section describes deficiencies and risks associated with Station Blackout testing, and describes existing tests that are useful in verifying analytically predicted plant responses to an actual SBO.

4.1 TEST DEFICIENCIES AND RISKS

The following subsections describe the disadvantages associated with performing a Station Blackout Test at CPS. The disadvantages associated with the test fall into two categories; deficiencies associated with the test that prevent an accurate simulation of the Station Blackout Event, and risks to plant equipment associated with the test.

4.1.1 TEST DEFICIENCIES IN SIMULATING SBO CONDITIONS

In this subsection, plant responses and operator actions under station blackout conditions are compared to those under SBO test conditions to show deficiencies in the simulation of an SBO by an SBO test.

4.1.1.1 RCIC SUCTION SOURCE

Under SBO conditions, RCIC suction would be manually switched from the RCIC storage tank to the suppression pool (SP) early in the event to conserve the cool RCIC tank water for use later in the event. This would be done to maintain RCIC operability as long as possible (as discussed in Section 3.3) by minimizing the potential for RCIC lube oil high temperature problems.

In a Station Blackout Test, water for the RCIC suction would only be taken from the RCIC storage tank. This restriction is necessary in order to avoid injecting suspended solids and impurities from the suppression pool 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.

Since operator action to transfer the RCIC suction source to the suppression pool would not be performed in the SBO test, the test would not provide representative training in this area.

4.1.1.2 DRYWELL TEMPERATURE RESPONSE

If a station blackout were to occur and the RCIC System is confirmed to be running, the operator would initiate a depressurization of the reactor vessel by opening one of the Safety Relief Valves (See Section 3.3).

The cool down rate during an SBO test would need to be kept under 100 °F/hr in order to avoid a thermal cycle on the RPV which may reduce its fatigue life. This is a slower cool down rate than would be encountered during actual SBO conditions and would result in higher drywell air temperatures.

4.1.1.3 SUPPRESSION POOL TEMPERATURE RESPONSE

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

1. Heatup Rate

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

   a. The reactor will be depressurized at a slower rate during the test than during an actual SBO event (See Section 4.1.1.2).

b. Pool water inventory will be increasing due to the addition of RCIC storage tank water earlier in an SBO test than during an actual SBO event (See Section 4.1.1.1).

2. CONDENSATION LOAD DISTRIBUTION

For an SBO test, the reactor would be depressurized using a number of relief valves in order to equalize condensation loads around the suppression pool. In an actual Station 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).

4.1.1.4 BATTERY DEPLETION RATE

Under station blackout conditions, the batteries assume the loads normally carried by their respective chargers. 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 would be energized and on standby during this test. 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 of plant battery depletion would not be representative of station blackout conditions.

4.1.2 RISKS IN SBO TESTING

This subsection describes some of the risks associated with performing an SBO test.

4.1.2.1 EXCESSIVE DRYWELL TEMPERATURE

Termination of drywell cooling due to the SBO test will result in a rapid increase in drywell temperature. Assuming an initial drywell temperature of 135°F (the Technical Specification Limit), 150°F will be reached at approximately 1 minute into the test. 150°F is the maximum normal operating temperature used as an equipment qualification envelope for most areas of the drywell. At approximately 14 minutes, the drywell air pressure will increase to 1.68 psig where a high drywell pressure LOCA signal is generated. This signal would hamper efforts to reestablish drywell cooling, because it isolates the chilled water lines for the drywell coolers. Temperature would continue to rise to a peak of approximately 225°F at 45 minutes into the test. These calculations are for bulk (average) drywell air temperatures. Much higher local temperatures may be experienced in thermal plumes above individual heat sources

and in the drywell head region. (The values listed in the above paragraph are approximate because they are based upon the analysis of the SBO event from Section 3 instead of an analysis of a Station Blackout test.)

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. Reanalysis, and possibly, requalification or replacement of safety-related equipment will be required, which would be a substantial cost to the plant owners.

Operator options to minimize drywell heatup in the event of inability to restore cooling are limited. The reactor could be depressurized rapidly through the use of ADS to reduce drywell heat loads, but this would impose a severe

thermal cycle on the reactor pressure vessel, reducing its design allowable fatigue lifetime and exposing the plant owners to the potential costs of reanalyzing fatigue life and reduced vessel lifetime.

In summary, any test in which drywell cooling is turned off (an expected consequence of SBO) with the primary system at operating temperatures 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 during a station blackout.

4.1.2.2 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 can fail by sticking open when the actuator is deenergized. This would result in an uncontrolled depressurization of the reactor with associated loss of RCIC for inventory makeup.

An SORV presents an uncontrolled loss of primary coolant. Although Emergency Core Cooling Systems would be available to makeup this loss, an SORV still represents a reduction in safety.

4.2 PLANNED TESTING TO SUPPORT ANALYTICAL PREDICTIONS

As indicated in the previous subsection, a single test of the response of multiple systems and components to Station Blackout cannot be formulated without undue risk to plant equipment. However, tests can be performed on single systems or components to determine their performance under Station Blackout conditions. Results from such tests can be used to verify the appropriateness of the analytical modeling techniques used to predict plant response to SBO. The following subsections describe tests that will provide information relevant to plant capabilities and responses under SBO conditions.

4.2.1 EXISTING TESTS

1. Preoperational Tests:

   -PTP-RI-01 Reactor Core Isolation Cooling

   The objectives of this test are:

   1. To verify the operation of the controls, alarms, interlocks, and valves associated with the RCIC system.

2. To verify automatic and manual system activation.
3. To verify automatic and manual system isolation.
4. RCIC turbine operation and pump flow parameters will be determined to the extent possible with an auxiliary steam supply.

-PTP-DC-01, PTP-DC-02, PTP-DC-03, PTP-DC-04 125 VDC Systems

These tests will verify the battery capacity of the four emergency DC battery systems, and verify equipment operability at minimum voltage levels. This is done by discharging the batteries.

-PTP-MS-01 Main Steam

As part of this test, operation of the SRVs using accumulator air alone will be demonstrated.

2. Startup Tests:

-STP-14 RCIC System

The purpose of this test is to verify the proper operation of the RCIC system over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic starting from cold standby when the reactor is at power.

-STP-26 Relief Valves

This test is intended to verify that the SRVs function properly, through SRV actuations at rated reactor pressure.

-STP-27 Turbine Trip and Generator Load Reject

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator. Reactor pressure and SRV response (including the SRV low-low setpoint logic) will be monitored.

-STP-31 Loss of Off-Site Power

This test is conducted to demonstrate plant performance under loss of off-site power conditions. Until diesel generators start and assume their loads during this test, plant response is identical to Station Blackout conditions.

4.2.2 TESTS FROM THE BWROG PROGRAM FOR COMPLIANCE WITH NUREG-0737 ITEM I.G.1 "TRAINING DURING LOW POWER TESTING"

The following are tests from Appendix E of the Boiling Water Reactor Owners Group program and will be performed at Clinton Power Station. They will verify aspects of the ability of CPS to cope with a Station Blackout as well as provide operators with additional training and experience with plant functions.

-<u>Startup of the RCIC system after loss of AC power to the system</u>.

This test is intended to demonstrate the ability of the system to start without the aid of AC power with the exception of that supplied from battery/inverter arrangements.

-<u>Operation of the RCIC system with a sustained loss of AC power to the system</u>.

The purpose of this test is to evaluate the limits of RCIC system operation with extended loss of AC power to it and support systems with the exception of that supplied from battery/inverter arrangements.

-<u>RCIC operation to prove DC separation</u>.

The purpose of this test is to verify proper operation of the RCIC DC components when batteries not required for RCIC operation are disconnected.

5.0 CONCLUSIONS

The conclusions that can be drawn from this report are as follows:

1. Station Blackout is an event that involves multiple failures of both onsite and offsite power sources.
2. Clinton Power Station has the capability to safely withstand a Station Blackout event for at least 2.4 hours if RCIC remains available. If RCIC is not available core cooling can be maintained for at least 30 minutes through boiloff of the core inventory.
3. Performance of a CPS Station Blackout Test is not desirable because of deficiencies the test would have in being able to accurately duplicate a Station Blackout event and because of risks to plant equipment arising from the test. Tests on individual systems will be performed that will provide information relative to the ability of equipment to function under Station Blackout conditions.

6.0 REFERENCES

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2. NUREG-0606, Unresolved Safety Issues Summary, February 13, 1981.
3. NUREG-0694, TMI Related Requirements for New Operating Licenses, June, 1980.
4. Tedesco, R. R., Assistant Director for Licensing, NRC, "TMI Task Action Item I.G.1, Special Low Power Test Program for BWRs", letter to Wuller, G. E., Supervisor-Licensing, IPC, 10/27/81.
5. Waters, D. B., Chairman BWR Owners Group, "BWR Owners' Group Evaluation of NUREG-0737 Requirement I.G.1, Training During Low Power Testing", letter BWROG-8120 to Eisenhut, D. G., Director Division of Licensing, NRC, February 4, 1981.
6. Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants, Rev. 2, August 1978.
7. S. Levy Incorporated, "Inadequate Core Cooling Detection In Boiling Water Reactors", Report for Boiling Water Reactor Owners Group, SLI-8218, November, 1982.

8. Eisenhut, D. G., Director of Licensing, NRC, "TMI Task Action Plan Item 1.G.1, "Special Low Power Testing and Training," Recommendations for BWRs (Generic Letter 83-24)", June 29, 1983.