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U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. D.G. Eisenhut
Division of Licensing

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR II)
DOCKET NO. STN 50-447

APPENDIX 15E - STATION BLACKOUT CAPABILITY

Attached please find a draft of new GESSAR II Appendix 15E pertaining to station blackout capability. This appendix concludes that the GESSAR II station blackout capability exceeds ten (10) hours. The assessed capability assumes credit for operator actions that are straightforward and where means exists to enable the operator to execute the action. Where features and/or equipment are not present, potential design improvements are recommended. It is anticipated that upon completion of NRC review a formal amendment on the GESSAR II docket will be submitted. This is anticipated to occur in early 1984.

If there are any questions on the information provided herein please contact J.F. Quirk at (408) 925-2606 or J.N. Fox of my staff at (408) 925-5039.

Very truly yours,

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Attachment

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APPENDIX 15E

STATION BLACKOUT CAPABILITY

APPENDIX 15E

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15E.1 INTRODUCTION AND CONCLUSIONS

15E.1.1 Introduction

This appendix is provided to demonstrate that the GESSAR II design has substantial capability to prevent a core damaging event well beyond the two-hour value recommended by NUREG-0626 and assumed in the Probabilistic Risk Assessment (Section 15D.3).

Attachment A contains responses to pertinent questions on station blackout of interest to the staff. These are addressed in more detail in other parts of this appendix.

15E.1.2 Conclusions

The GESSAR II station blackout capability exceeds ten (10) hours. The assessed capability assumes credit for operator actions that are straightforward and where means exist to enable the operator to execute the action. Where features and/or equipment are not present, potential design improvements are recommended. These operator actions and potential design improvements are summarized below:

1. Operator Actions
   a. Manual RPV Water Level Control with RCIC.
   b. Shift of RCIC pump suction to the condensate storage tank.
   c. Vessel depressurization with SRVs to about 200 psig. Maintain vessel pressure above 150 psig with manual SRV control.
2. Potential Design Improvements
   a. Provide manual logic override of the RCIC suction transfer signal and test line closure signal from the control room.

b. Provide Enhanced Water Level Instrumentation (currently under review for Appendix 1D).

c. Provide alternate power supply to RCIC gland compressor.

An ongoing evaluation of the 125 VDC battery capability is in progress. However, if necessary to ensure 10-hour capability, emergency DC bus cross ties, or larger battery capacity, or other methods will be identified.

In addition to the above actions, the following contingency actions could be taken to provide even longer duration capability are:

1. Provide override capability for the RCIC room high temperature isolation logic to be used if room temperature exceeds about 150°F.
2. Extend SRV pheumatic supply by replacing air bottles if depleted. A connection outside the fuel building would be more convenient.

15E.2 DEFINITION OF STATION BLACKOUT

Station blackout refers to the total loss of both off-site and on-site a.c. electrical power. In draft information pertaining to proposed Regulatory Guides, the NRC consultants refer to "Emergency AC" loss in addition to offsite power loss. This could be interpreted as the Division 1 and 2 Standby Emergency Diesel Generators. Both HPCS and RCIC operate at high pressure and can be considered redundant water sources available for maintaining core cooling during design basis assumptions that assume a single failure (i.e., such as a D-G). This configuration is believed to be adequate to comply with the proposed regulatory requirements. For purposes of this assessment,

however, a failure of the HPCS diesel generator has been assumed in addition to loss of offsite power and the division 1 and 2 diesel generators thus providing a more severe impact on plant systems and the station battery.

A one-line diagram of the GESSAR II design is shown in Figure 8.3-1. Three divisions of 6.9 kv on-site power are provided; two by standby emergency diesel generators (in addition to preferred and alternate off-site power sources); the third by an off-site power source and a separate and diverse diesel generator dedicated to division 3 eletrical power. Division 3 supports the High Pressure Core Spray (HPCS) system and all of its supporting auxiliaries.

The GESSAR II design also includes a steam turbine driven Reactor Core Isolation Cooling System (RCIC) which operates in an emergency independently of a.c. electrical power. This system is designed to provide high pressure makeup to the RPV during isolation events and would thus be initiated automatically during a postulated blackout event. The plant response with RCIC alone has been reviewed, and the duration capability of the GESSAR II plant in excess of ten hours has been verified. This configuration is consistent with the station blackout definition in the Probabilistic Risk Assessment (Section 15D.3).

In the evaluation certain assumptions have been made:

- o No Loss of Coolant Accident (LOCA), stuck open relief valve (SRV) or failure to scram concurrent with the station blackout is considered.

- In evaluation of equipment, some capability beyond environmental qualification limits has been assumed. In assessing the ultimate failure capability of equipment the judgement of senior General Electric engineering personnel has been relied upon to provide guidance. Such judgements are explicitly call out in the following sections.
- Operator actions are identified where adequate time and skills would be expected to be available to a typical operating plant staff. No extra-ordinary actions on the part of the operator are assumed; rather, only straightforward, simple actions are allowed.
- No credit for off-site assistance from a utility maintenance crew using portable electric generators or batteries has been assumed for this assessment even though this possibility may exist within the time frame of interest. Such capability might be considered by an applicant to improve the restoration time for on-site emergency a.c. power if the situation warranted.

15E.3 INDICATION OF STATION BLACKOUT

The station blackout event is characterized by a loss of all off-site power (preferred and alternate feeders) and a loss of divisions 1, 2 and 3 of on-site a.c. power. As noted in Section 1D.2.3.33 of the assessment against Regulatory Guide 1.97, the class 1E power distribution system monitors voltage on the three 6.9 kv a.c. buses and the four 125 V d.c. buses. This indication is displayed on panel P800 in the main control room. A potential station blackout event would be first noticed by the plant operators by a change in the control room lighting which would alert him to evaluate both the plant and the electrical distribution system status. By observation of the loss of bus voltage on the 6.9 kv buses "E", "F" and "G" and the breaker position for incoming voltage to these buses, the operator would be alerted to the presence of a potential blackout event. Voltage indication on the d.c. buses E, F, G & H would assure the operator that power is available to control the event.

Prior to conducting the various operator actions needed to mitigate a blackout event, the operator must distinguish between a short duration event and a prolonged blackout. A short duration event would be one in which restoration of an off-site or on-site a.c. power source would occur prior to development of conditions requiring the operator actions

defined later in this supplement. Minimizing the time to recognize this event is important so that the potential drain on the batteries is controlled.

Upon recognition of the a.c. power source failure, an auxiliary operator would be sent to each of the diesel generator rooms to attempt a manual start. Simultaneously, the control room operator should attempt to start each diesel from the main control room. In addition, the system dispatcher would be contacted by the shift supervisor to determine the status and likelihood of off-site power restoration. Accomplishment of these activities in addition to those related to controlling vessel water level and pressure is expected to take about 30 minutes.

Thus recognition of a station blackout event and the initiation of any blackout specific operator actions is expected to be delayed for about 30 minutes.

15E.4 INSTRUMENTATION REQUIREMENTS

Instrumentation required to monitor plant status during a blackout event has been selected from a review of the type A through E variables discussed in ~~Section~~ Appendix 1D which is the response to Reg Guide 1.97 requirements. This list has been augmented slightly to account for specific variables such a room temperatures and certain valve and breaker position indications which are needed to determine plant conditions.

Table 15E-1 lists the variables considered and whether or not they are needed for the blackout sequence. The basis for selection generally is based on the need for the operator to follow Emergency Procedure Guidelines (or take other actions which may later be established) during the period of interest. As such, type A variables are identified as needing indication during the blackout event while variables which are more representative of monitoring core damage or breaks of the reactor coolant boundary or effluent release are excluded.* [redacted]

Table ~~4-2~~ 15E-2 shows the power supplies in the GESSAR II design for the instruments needed. All indications needed to follow the blackout event are or will be powered from 125V d.c. sources.

The applicant could [redacted] provide d.c. ~~backed~~ backup power to the condensate storage tank level indicator and to ensure local control room temperature indication as available.

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* Since releases stemming from a postulated station blackout event are within existing design bases events for upset conditions.

15E.5 PLANT RESPONSE FOLLOWING A STATION BLACKOUT

The key plant areas which could potentially effect the ability of the plant design to accommodate a station blackout are:

- o RCIC room
- o Remote shutdown panel area
- o Suppression pool and containment
- o Drywell
- o Control room
- o Fuel pool

In addition non-electrical a.c. plant energy supplies will be consumed and need to be addressed to assess the plant capability. These are:

- o Pneumatic Air Supply System (ADS)
- o D.C. Power Distribution System

These areas and energy supplies will be discussed in subsequent sub-sections. An estimate of limiting condition, design improvements or operator actions needed are noted in each.

15E.5.1 Areas

~~Area~~ 15.E.5.1.1 RCIC Room

a. Reason for Concern

- o Room temperature increase without area cooling could cause a loss of RCIC control due to equipment failure.
- o Isolation and turbine trip due to leak detection system trip. (Trip setpoint approx. 170°F) could prevent RCIC from operating .
- o Steam line drain valves may fail after air supply becomes exhausted causing system damage on restart.

b. Plant Response

- o Approx. 122°F in 12 hours (w/CST suction)
- o Approx. 133°F in 12 hours (w/SP suction)
- o Approx. 101°F in 12 hours (w/10 lb/hr steam)

} See Attachment B

Critical Components	Limitation
EH Differential Coil	Approx. 170°F water temp.
Magnetic Speed Sensor	225°F
Instrumentation	212°F

Capability >12 hrs

C. Assumed Operator Actions

- o Manual switch of RCIC suction to CST at about 30 min.
- o Override RCIC high temp isolation if room temp > approx. 150°F (not expected)
- o Manual RPV level control of RCIC to avoid L8 trip and restart.

d. Potential Modifications/Actions

- o Ensure override capability exists for RCIC room isolation signal.
- o Ensure override capability for RCIC suction transfer.
- o Provide logic changes to permit low flow RCIC injection. Requires override capability on test line to CST to obtain flow split between CST return and vessel.

15E.5.1.2

Area: Remote Shutdown Panel Area

a. Reason for Concern

- o RCIC electronics could fail if area temperature exceeds 150°F.
- o Access needed if control room uncomfortable or electronics erratic.

b. Plant Response

- o Not evaluated, but very little heat source. ~~S~~since Remote Shutdown Station panel is deenergized until control transfer switch is thrown.
- o Expect area temperature to remain <150°F for 20 hours

Capability >20 hrs

c. Assumed Operator Actions

None.

d. Potential Modifications/Actions

None.

15E.5.1.3

~~a Area:~~ Suppression Pool

a. Reason for Concern

- High suppression pool temperature could cause NPSH limits (approx. 175°F) and reduced lube oil cooling, to RCIC.
- High suppression pool level causes suction transfer.
- High containment air temperature may cause erratic RPV indication.
- High suppression pool temperature and level increases containment loads.

b. Plant Response

time (hrs)	T_{sp} (°F)	T_c (°F)	L_{sp} (ft)
1	135	100	+2
5	190	175	+5
10	220	220	@ weir
15	225	225	@ weir
20	230	230	@ weir

Notes: T_c based on T_{sp} + judgment

T_{sp} based on Table 15D.2-2 + calculation

Capability >10 hours.

Instruments qualified to 185°F; capability likely to 250°F.

c. Assumed Operator Actions

- Manual switchover back to CST within 1 hr. eliminates potential NPSH problem.
- Maintain vessel pressure below heat capacity temperature limit per EPGs - ensure written procedures contain heat capacity temperature limit curve [redacted] - may need to exceed heat capacity temperature limit slightly after approx. 6 hrs, but acceptable because no additional depressurization required. Consistent with EPGs.

d Potential Modifications/Actions

- Ensure manual override capability for RCIC suction transfer

ATTACHMENT A
TO
APPENDIX 15E

ACRS QUESTIONS PERTAINING TO AC/DC POWER SYSTEM RELIABILITY

15EA.1 DC RELIABILITY

<u>Question:</u> The NRC Staff has issued a report (NUREG-0666) on the reliability of d.c. power system in which a 2-train d.c. system found to meet minimum NRC requirements was evaluated. As a result, the d.c. power system was identified as a potentially high contributor to core melt. The applicant could be asked what his assessment of his d.c. system is and what consideration he has given to the recommendations of NUREG-0666.

<u>Response:</u> We do not favor the use of such a minimum system as considered in NUREG-0666. [redacted] For example, it has a single bus tie breaker with too much potential for common cause failure. ~~Our~~ The original GESSAR II design allowed d.c. cross-connection capability with <u>dual</u> cross-tie breakers and double key interlocks. GE agreed to delete the d.c. cross-connection capability from the GESSAR II design until such time it can be shown that this capability does not contribute to d.c. system unreliability.

The following is provided in response to the recommendations in NUREG-0666:

(1) Prohibits certain design and operational features of the d.c. power system such as use of a tie breaker which could compromise divisional independence. As noted above, GESSAR II complies although we believe cross-connection capability is appropriate for specific conditions during shutdown and occurrences which require last resort flexibility (such as station blackout). GESSAR II has four

safety-related batteries, each of which has two chargers so that charger maintenance does not require use of cross-connections nor cause draw-down on the battery.

(2) Addresses testing and maintenance activities. These are accomplished by the applicant. We agree with these recommendations, and the GESSAR II design allows their implementation.

(3) Requires staggered test and maintenance activities to minimize the potential for human error related common cause failure. This is controlled in the field, but we agree that these actions are appropriate.

(4) Requires design and operational features to be adequate to maintain reactor core cooling in the hot standby condition following the loss of any one d.c. power bus and a single independent failure of any other system required for shutdown cooling. Although we cannot disagree with the intent of this recommendation, a judgment as to what features are needed should be tempered with an assessment of the reliability of the d.c. power loads and sources. We have concentrated on maintaining full separation and independence between division 1 and division 2 d.c. systems to provide this reliability. [redacted] With four independent d.c. systems and with three independent a.c. systems, the GESSAR II design shown ~~we expect~~ substantial capability in meeting the NUREG-0666 recommendation. For example, a potentially adverse capability loss would follow from the loss of both RHR systems, but the suppression pool can

store decay heat for several hours, during which it may be possible to recover active decay heat removal.

15EA.2 Grid Reliability

Question: What is the applicant's assessment of grid reliability and what procedures exist for restoring offsite power to the plant in the event of this loss.

Response: The grid is the responsibility of the applicant, and we assume he will meet the NRC requirements in this area. On loss of normal preferred offsite power, there is automatic transfer to the alternate offsite power source and, if necessary, to the onsite diesel generators. Restoring preferred power is accomplished manually by the control room operator. The specific procedures for restoration of power in the switchyard or transmission systems would be developed by the applicant.

Station Blackout Analysis

Question: What are the results of the applicant's station blackout analyses? Has the applicant made a best-estimate analysis of the accident sequence and evaluated what might be done to improve the plant, or has a conservative analysis been made with a core melt assumed after some specified degradation of the battery?

Response: This evaluation responds to both questions. [redacted] Our best-estimate analysis to the extent that it is complete is the primary subject of this supplement. We have identified potential system design and procedural improvements, and we will implement them upon concurrence from the NRC that they [redacted] satisfactorily resolve the issue.

[redacted] Our probabilistic risk assessment considered station blackout capability in a conservative manner (core cooling lost in two hours due to battery depletion and loss of RCIC control). We believe the more realistic treatment considering automatic and manual d.c. load shedding shows a substantially longer capability.

15EA.3 Diesel Generators

Question: What is the applicant's assessment of his diesel generator system? To what extent has LER and operating experiences been used to improve the design?

Response: Our HPCS diesel generator has undergone extensive testing (including 300 tests without failure) which has been documented for the NRC. From this testing and from field experience we have high confidence in the design. Extensive review of the design specifiation, the installation design and the auxiliary system design for the larger diesel generators (division 1 and 2) demonstrates [redacted] high availability from these units.

15EA.4 Low Power Testing/Simulated Loss of Offsite Power

Question: Has the applicant performed low power testing and a simulated loss of offsite power test? If so, what are the results and what has the applicant learned?

Response: [redacted]
This is the responsibility of the Applicant.
[redacted]

156.5.1.4

~~Area~~ Drywell

a. <u>Reason for Concern</u>

- o High drywell temperature could cause RPV level instrument reference leg boiloff.
- o High drywell temperature might exceed qualification levels for drywell equipment.
- o High drywell temperature could cause SRV solenoid failure.

b. <u>Plant Response</u>

Approx. 135°F during plant operation

<270°F prior to depressurization at 30 min.

<200°F after depressurization to 200 psi

Capability: unlimited

Drywell equipment qualified for >300°F

c. <u>Assumed Operator Actions</u>

- o Depressurization to approx. 200 psi to limit drywell heatup.
- o Maintain pressure >118 psi to avoid reference leg flooding.
- o Maintain RPV water level approx. + 20" on Enhanced Level Instrument.

d. <u>Recommended Modifications/Actions</u>

- o Enhanced water level instrument (ELI) compensates for drywell and containment temperature effects. (Previously recommended. See [redacted] Appendix 1D.)

15E.5.1.5

~~Area~~ Control Room

a. Reason for Concern

- High control room temperature could cause computer/microprocessor controls to fail.
- High temperature could make the control room uninhabitable.

b. Plant Response

- PGCC floor section heat sinks expected to prevent heatup above 105°F.

Capability: unlimited.

- Humidity could become uncomfortable but not uninhabitable.

Microprocessors (ELI, ERIS, etc.) unreliable above approx. 105°F but backup information is available at Remote Shutdown Station (RSS).

c. Assumed Operator Actions

- Transfer control to remote shutdown station (RSS) if control room becomes uninhabitable. (not expected)

d. Potential Modifications/Actions

None.

156.5.1.6

~~Area:~~ Fuel Pool

a. Reason for Concern

- Loss of fuel pool cooling could cause fuel pool to boil away.

b. Plant Response

- Approx. 14 hrs to boiling
- Approx. 77 hrs to fuel uncovery

Basis: Judgment probably longer with less hot fuel

Capability >75 hrs.

c. Assumed Operator Actions

None, but SRV air bottle replacement (see pneumatic supply) could be hampered by fuel building environment after approx. 14 hrs.

d. Potential Modifications/Actions

Consideration of moving extra air bottles to corridor outside fuel building. Not required for station blackout.

ISE.5.2 Energy Supplies

~~Energy Supply~~ ISE.5.2.1 Pneumatic Supply

a. **Major Sources of Consumption**

- ADS/SRV
- Drywell and containment vacuum breakers

b. **Estimated Duration** (5000 CF available)

SRV Depressurization approx. 50 actuations @ 8 CF/actuation = 400 CF

Ongoing SRV use approx. $\frac{\text{1 actuation}}{\text{2 min.}} \times \frac{\text{60 min.}}{\text{hr.}} \times \text{8 CF} = \text{240 CFH}$

Leakage @ 1 CFH/valve x 8 valves = 8 CFH

DW Vacuum Breakers approx. $\frac{\text{1 act}}{\text{7 hrs}}$ @ $\frac{\text{15 CF}}{\text{act}} \times \text{2 VB} =$ 4 CFH

total approx. 250 CFH

$\frac{5000 - 400}{250} = \text{18 hrs.}$

$\boxed{\text{Capacity >18 hrs}}$

c. **Operator Actions to Extend Duration**

- Air bottle replacement after depletion possible if necessary (not expected).
- Rotate use of ADS/SRV valves to permit time for accumulators to recharge and give preference to Division 2 ADS/SRV values.
- Monitor SRV position indication to indicate need for switch to other values (valves close when air supply lost).

d. **Potential Modifications**

None

15E.5.2.2

~~Energy Supply~~ – 125 VDC – Bus E

a. Major Sources of Consumption

See Table 8.3-6

b. Estimated Duration (1950 amp hours (AH), 2 hr)

RCIC Gland Compressor Modification (see below) delete 58A

Shed load approx 35A (see below)

Steady state load approx. 251-58-35 = 158A

Capability
> __*__ hrs.

c. Operator Actions to Extend Duration

- Shed the following loads (at approx. 30 min.)
  - NMS panel H13-P669 (NSPS) - 25A from NSPS inverter
  - Emergency lighting (fuel building) - 10A

d. Potential Modifications

- Power RCIC gland compressor from an alternate source.
- Delete 125 VDC emergency lighting system except for control building or move to Bus J.
- Provide Emergency crosstie capability with dual crosstie breakers and double key interlocks if needed for longer duration.*
- Provide larger capacity battery if needed for longer duration*.

*The capability of this battery with load shedding is being evaluated. If the estimated duration is less than about 10 hours, the addition of crossties or expanded battery size will be reviewed to determine the optimum configuration for achieving a 10-hour capability.

15E.5.2.3
~~Energy Supply~~ 125 VDC - Bus F

a. Major Sources of Consumption

See Table 8.3-7

b. Estimated Duration (1500AH, 2hr)

Shed Loads approx. 40A (see below)
Steady State Load = 175 -40 = 135A

Capability
> _*_ hrs.

c. Operator Actions to Extend Duration

Shed the following loads at approx. 30 min.

- NMS panel H13-P670 (NSPS) -25A
- Emergency lighting -15A

d. Potential Modifications

o Delete 125 VDC emergency lighting in auxiliary building

o Provide larger capacity battery if needed for longer duration.*

*The capability of this battery with load shedding is being evaluated. If the estimated duration is less than about ~~8~~ 10 hours, the addition of crossties or expanded battery size will be reviewed to determine the optimum configuration for achieving a 10-hour capability.

15E.5.2.4

~~Energy Supply~~ 125 VDC Bus G

Major Sources of Consumption

See Table 8.3-8

Estimated Duration (400 AH, 6 hr)

Shed Loads = 25A (see below)
SS load = 78 -25 = 53A

Capability
> * hrs.

Operator Actions to Extend Duration

Shed the following load at approx. 30 min.

NMS panel H13-P671 (NSPS) -25A

Potential Modifications

Larger capacity battery if needed for longer duration.*

*The capability of this battery with load shedding is being evaluated. If the estimated duration is less than about 10 ~~6~~ hours, the addition of crossties or expanded battery size will be reviewed to determine the optimum configuration for achieving a 10-hour capability.

15E.5.2.5

~~Energy Supply~~ 125 VDC Bus H

Major Sources of Consumption

See Table 8.3-9

Estimated Duration (425 AH, 2 hr)

Load Shed = 25A
SS Load = 100 -25 = 75A

Capability > * hrs.

Operator Actions to Extend Duration

Shed the following load at approx. 30 min.
Shed NMS Panel H13-P672 (NSPS) -25A

Potential Modifications

None

*The capability of this battery with load shedding is being evaluated. If the estimated duration is less than about [redacted] 10 hours, the addition of crossties or expanded battery size will be reviewed to determine the optimum configuration for achieving a 10-hour capability.

TABLE 15E-1

VARIABLES ASSESSED FOR STATION BLACKOUT ASSESSMENT

Variable	RG 1.97 Type	RG 1.97 Category	Discussion Subsection	Needed in Blackout Sequence?
Reactivity Control				
Neutron Flux (value, rate, trend)	A,B	1	1D.2.3.1	No*
Control Rod Position	B	3	1D.2.3.2	No*
Boron Concentration (sample)	B	3	1D.2.3.3	No
Core Cooling				
Coolant Level in the Reactor (value, trend)	A,B,C	1	1D.2.3.4	Yes
Maintaining Reactor Coolant System Integrity				
RCS Pressure (value + alarm)	A,B,C	1	1D,2.3.5	Yes
Drywell Sump Level (value + alarm)	B,C	3	1D.2.3.6	No
Drywell Pressure	B,C,D	1,2	1D.2.3.7	No
Primary Containment Area Radiation	E C	1 3	1D.2.3.8	No
Suppression Pool Water Level	A,C,D	1.2	1D.2.3.9	Yes
Maintaining Containment Integrity				
Primary Containment Isolation Valve Position (Excluding Check Valves)	B	1	1D.2.3.10	Yes**
Primary Containment Temperature	A	1	1D.2.3.11	Yes

*ATWS plus blackout is not considered in this study. Failure to scram can be inferred from abnormal water level and pressure response.

**Plus RCIC minimum flow.

TABLE ~~4-1~~ 15E-1

VARIABLES ASSESSED FOR STATION BLACKOUT ASSESSMENT (Continued)

Variable	RG 1.97 Type	RG 1.97 Category	Discussion Subsection	Needed in Blackout Sequence?
Maintaining Containment Integrity (Continued				
Primary Containment Pressure (value, rate, trend, + alarm)	A,B,C	1	1D.2.3.12	Yes
Drywell/Containment Hydrogen Concentration (value)	A,C	1	1D.2.3.13	No
Secondary Containment Area Radiation (value)	C,E	2	1D.2.3.14	No
Secondary Containment Noble Gas Effluent	C,E	2	1D.2.3.15	No
Primary Containment Noble Gas Effluent	C	3	1D.2.3.16	No
Suppression Pool Temperature	A,D	1,2	1D.2.3.17	Yes
Drywell Air Temperature	A,D	1,2	1D.2.3.18	Yes
Fuel Cladding Barrier Monitoring				
Coolant Radiation (value + alarm)	N/A	N/A	1D.2.3.19	-
Coolant Gamma (1 sample/6 hours) results within 72 hr	C	3	1D.2.3.20	No
System Operation				
Main Steam Line Isolation Valve Leakage Control System Pressure	D	2	1D.2.3.21	No
Containment Spray Flow	D	2	1D.2.3.22	No

TABLE 15E-1

VARIABLES ASSESSED FOR STATION BLACKOUT ASSESSMENT (Continued)

Variable	RG 1.97 Type	RG 1.97 Category	Discussion Subsection	Needed in Black-out Sequence?
System Operation (Continued)				
Residual Heat Removal (RHR) System Flow	D	2	1D.2.3.22	No
RHR Service Water Flow	D	2	1D.2.3.23	No
Low Pressure Coolant Injection System Flow	D	2	1D.2.3.22	No
Reactor Core Isolation Cooling System Flow	D	2	1D.2.3.24	Yes
RCIC Room Temp.	-	-	-	Yes
Control Room Temp.	-	-	-	Yes
High Pressure Coolant Spray System Flow	D	2	1D.2.3.24	No
Core Spray System Flow	D	2	1D.2.3.24	No
Standby Liquid Control System (SLCS) Flow	D	2	1D.2.3.25	No
SLCS Storage Tank Level	D	3	1D.2.3.26	No
SRV Position	D	2	1D.2.3.27	Yes
Feedwater Flow	D	3	1D.2.3.28	No
CST Level	D	3	1D.2.3.29	Yes
ESF Cooling Water Flow	D	2	1D.2.3.30	No
ESF Cooling Water Temperature	D	2	1D.2.3.30	No
High Radioactivity Tank Level	D	3	1D.2.3.31	No
Emergency Vent Damper Position	D	2	1D.2.3.32	Yes
Standby Energy Status	D	2	1D.2.3.33	Yes*

*Including breaker position.

TABLE ~~4-2~~ 15E-1

VARIABLES ASSESSED FOR STATION BLACKOUT ASSESSMENT (Continued)

Variable	RG 1.97 Type	RG 1.97 Category	Discussion Subsection	Needed in Blackout Sequence?
Effluent Monitoring				
SGTS Ventilation Flow Rate	E	2	1D.2.3.34	No
Other Ventilation Flow Rates	E	3	1D.2.3.34	No
Particulate/Halogen Release (sample)	E	3	1D.2.3.35	No
Environs Radioactivity Monitoring	E	3	1D.2.3.36	No
Meteorology	E	3	1D.2.3.37	No
Post-Accident Sampling (sample)	E	3	1D.2.3.38	No

TABLE ~~#~~ 15E-2

POWER SUPPLIES TO INSTRUMENTS NEEDED FOR A BLACKOUT

Variable	Control Room Indicator	Power Supply	Available?	Notes
RPV Level	B21 R623A R623B	120 Inst. Bus A 120 Inst. Bus B	Yes	1
RPV Pressure	B21 R623A R623B	120 Inst. Bus A 120 Inst. Bus B	Yes Yes	1 1
Suppression Pool Water Level	P50-R600A,B	125 VDC	Yes	3
Pri. Containment Isol. Valve Position	Indication Lights	RPS	Yes	
Pri. Containment Temperature	T41-RR613A,B	125 VDC	Yes	3
Pri. Containment Pressure	T41-RR618A,B	125 VDC	Yes	3
Suppression Pool Temperature	P50-R600A,B	125 VDC	Yes	3
Drywell Air Temperature	T41-RR611A,B	125 VDC	Yes	3
RCIC Flow	E51-R606	RPS	Yes	
RCIC Room Temperature	E31-R608	RPS	Yes	
Control Room Temperature	-	-	Yes	5
SRV Position	Indicating Lights	125 VDC	Yes	
CST Level	By applicant	By applicant	Yes	2
Emergency Vent Damper Position	Indicating Lights	125 VDC	Yes	3
Standby Energy Status 619 kv AC	Voltmeters	Source	Yes	
DC	Voltmeters	Source	Yes	
Air	P53-R606A,B	125 VDC	Yes	3

Notes to Table ~~4-2~~ 15E-2

1. Enhanced Water Level Instrument to be powered from d.c. power.

2. D.C. power to be provided by applicant.

3. Power Supply from 125V d.c. to Reactor Island Logic Panels P881 or P882.

4. Exhaust air measurement may be unreliable. Local thermometer to be supplied by applicant.

GESSAR II
238 NUCLEAR ISLAND

ATTACHMENT B
TO
APPENDIX 15E

RCIC ROOM HEATUP DURING A STATION BLACKOUT

15EB.1 PURPOSE

The purpose of this ~~memorandum~~ attachment is to document the results of analysis performed by Containment and Radiological Engineering on Reactor Core Isolation Cooling System (RCIC) room temperature response during a station blackout for ~~the~~ GESSAR ~~standard~~ II. ~~plant~~. The results indicate that a station blackout imposes no threat to the operation of RCIC with the RCIC room temperature reaching 122°F 12 hours into the transient, well below the point above which RCIC performance would be degraded. Sensitivity results for some of the most important parameters are also given.

15EB.2 INTRODUCTION

A station blackout results in loss of all A.C. power (both offsite and onsite sources), initiating reactor isolation and scram. For this analysis all three diesel generators of a BWR plant are assumed inoperative, i.e., no Emergency Core Cooling System (ECCS) pumps are available: this leaves the battery operated RCIC as the only system available for core cooling. Thus, it is essential that the RCIC remains operational. An important requirement for the proper functioning of the RCIC is that the RCIC room temperature be maintained below the equipment operational limit.

The loss of all A.C. power also means the loss of lighting, auxiliary equipment operation, area HVAC and drywell fan coolers, resulting in a drywell heatup. At some point reactor depressurization will be initiated to reduce the heat input to the drywell, although the reactor is assumed to be depressurized only to the point sufficiently above the RCIC shutoff pressure so that operation of the RCIC can be maintained.

RCIC initially draws water from the Condensate Storage Tank (CST). However, an automatic switchover to the suppression pool as the water

source would occur if the CST water level drops too low or the suppression pool water level rises above a certain point. Since the suppression pool heats up as a result of SRV discharges and subsequent reactor depressurization, and since the design temperature for the RCIC pump is 140°F, a manual switch back to the CST from the suppression pool as the RCIC water source is required when the pool temperature approaches 140°F. Since the time period when the RCIC takes suction from the suppression pool is relatively short (about 30 minutes) compared to the transient period of interest (up to 20 hours), the impact on RCIC room temperature in assuming that RCIC draws all water from the CST is insignificant.

SEB.3

MODELING AND ASSUMPTIONS

To model the RCIC room temperature response, thermodynamic properties of steam and air in the room are evaluated based on mass and energy balances. Heat sources and heat sinks were considered. In addition, some steam has leaked into the room through the RCIC turbine gland seal. The room is conservatively assumed to be isolated from the adjacent rooms.

Heat Sources - The following heat sources are modeled:

- Steam Pipes - there is a six inch steam pipe upstream of the RCIC turbine, 60 ft long, with three inches of insulation, with the pipe temperature assumed equal to the reactor steam temperature of 552°F under normal operating conditions, and 388°F after reactor depressurization to 200 psig): and a sixteen inch exhaust steam pipe downstream of the RCIC turbine, 40 ft long, with two inches of insulation, with pipe temperature at 250°F because steam pressure downstream of the turbine is held at 25 psia.

- Water Pipes - two uninsulated water pipes, one suction pipe and the other discharge pipe, with dimensions of 8''X 38 ft and 6''X 36 ft, carry water from the water source and inject it into the reactor. As mentioned previously, the water source may be either the CST or the suppression pool, thus the water temperature may vary from the CST temperature of 90°F up to the suppression pool temperature. Depending on the RCIC room temperature at a particular time, these water pipes may be either heat sources or heat sinks.

- Turbine - the RCIC turbine is insulated. The turbine temperature is taken as the average upstream and downstream steam temperatures. Small portions of turbine that are not insulated are not modeled.

- RCIC Pump - the RCIC pump weighs 6600 lbm and is not insulated. As in the case of water pipes, the RCIC pump may become a heat sink depending on the room temperature and the water temperature.

Heat Sinks - The following heat sinks are modeled:

- Concrete Walls, Floor and Ceiling - the walls are 26 ft tall, with widths varying from 18 ft to 31 ft. Thicknesses vary from 1 ft to 3 ft. These structures were conservatively assumed to be insulated on the outer surface.

- Turbine Base Plate - it weighs 900 lbm and is uninsulated.

- Room Cooler - it weighs 2000 lbm and is uninsulated.

- As mentioned previously, the water pipes and RCIC pump become heat sinks if the RCIC room temperature is higher than the RCIC water temperature.

Analytical Assumptions - The following assumptions were made in the analysis, with justifications for these assumptions given subsequently:

- Air and steam are uniformly mixed at all times.

- Air behaves like an ideal gas.

- No condensation on structural surfaces.

- The RCIC room is isolated from the surroundings.

- Heat conduction is one dimensional through structures and walls.

Since the period of interest is several hours, steam leaked into the room has sufficient time to diffuse and mix with air, therefore, the uniform mixing assumption is a good approximation. Also, since only low pressures and temperatures are encountered, the ideal gas law holds true for air.

Assumptions of no condensation on structural surfaces is conservative because the free-convection heat transfer coefficient used in the absence of condensation is smaller than the condensing heat transfer coefficient. Isolating the RCIC room is another conservatism, because mass and energy are preventd from leaving the room through conduction, convection and radiation. Finally, the one-dimensional heat conduction assumption is correct except at the corners of the walls, but the impact is negligible.

5.B.4

INPUT PARAMETERS

The following initial conditions and key parameters were used in the analysis:

- Initial room temperature was 90°F.

- Steam leakage rate was 70 lbm/hr.

* No reactor depressurization for the first 30 inutes (as the operator is trying to determine appropriate actions) and the reactor was cooled down at 100°F/hr.

* Temperature of RCIC water was 90°F, which is the technical specification CST temperature, because the RCIC can take suction from the suppression pool for only a short period of time and the operator will switch the sucton back to the CST as the pool approaches 140°F.

15EB.5 RESULTS AND DISCUSSIONS

A timeshare computer program has been developed to carry out the calculations described above.

The RCIC room temperature response following a station blackout is given in Figure 15EB-1. ~~1.~~ The temperature increases rapidly during the first hour of the transient, then the rate of increase levels off subsequently. The room temperature rises to 119°F at eight hours of transient and 122°F at twelve hours of transient.

Figures 15EB-2 and 15EB-3 ~~2 and 3~~ show the sensitivity results at high water temperature and low steam leakage rate, respectively. With the water temperature at 140°F, the RCIC room temperature rises to 133°F at twelve hours, while at the steam leakage rate of 10 lbm/hr (which corresponds to new turbine gland seal condition) the room temperature reaches only 101°F at twelve hours. The high sensitivity to the steam leakage rate is due to the large latent heat of steam which is released upon condensing in the RCIC room. The sensitivity study also indicates that there is no impact of reactor cooldown rate on the RCIC room temperature response.

The above results indicate that the RCIC room temperature twelve hours following a station blackout to be substantially below the equipment quaification limits of 212°F for the first six hours and 150°F between six and twelve hours following a station blackout. This shows that proper operation of the RCIC can be maintained for many hours during a station blackout to provide adequate core cooling.

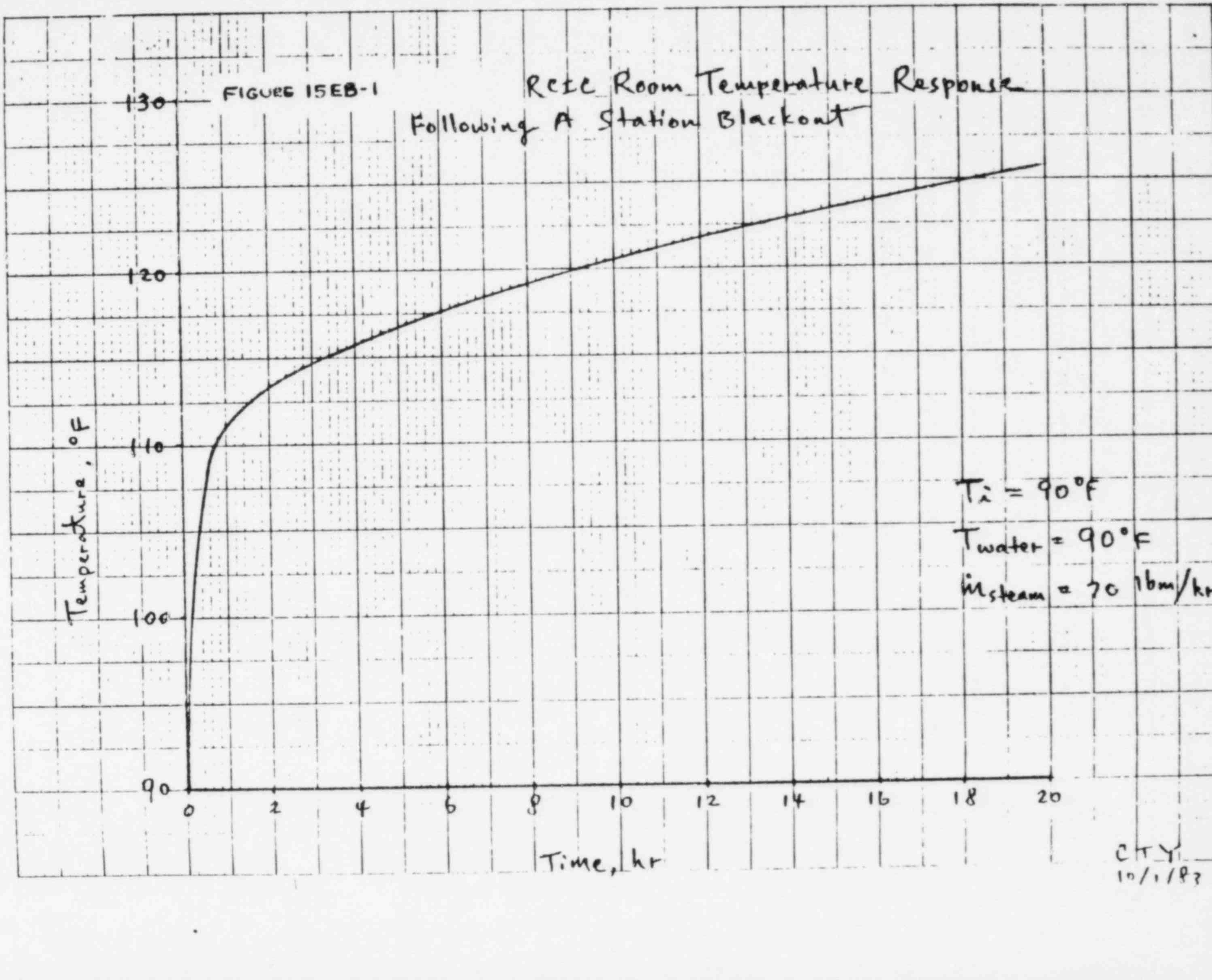

FIGURE 15EB-1 RCIC Room Temperature Response Following A Station Blackout

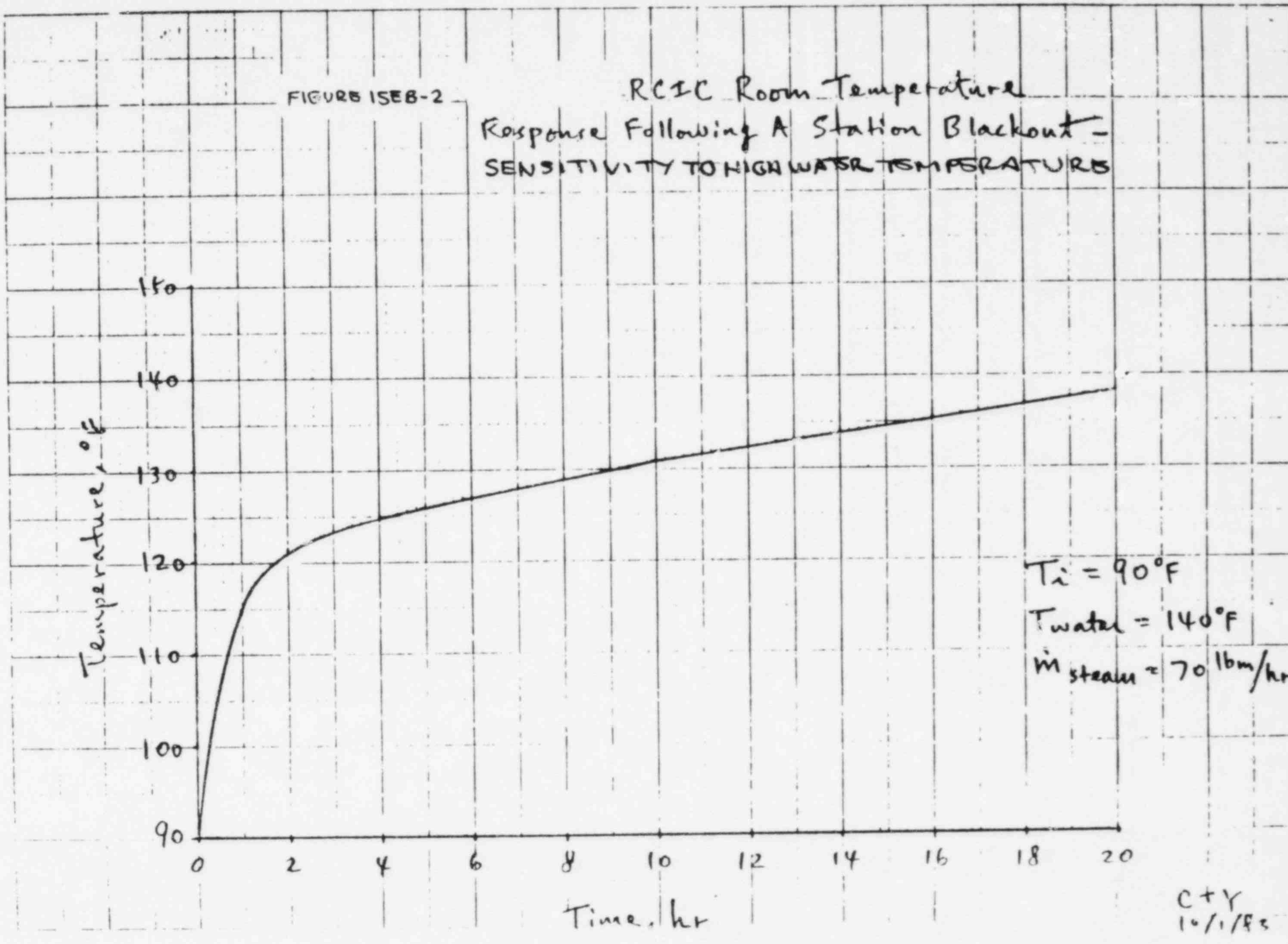
FIGURE ISEB-2
RCIC Room Temperature
Response Following A Station Blackout –
SENSITIVITY TO HIGH WATER TEMPERATURE
Temperature, °F
150
140
130
120
110
100
90
0
2
4
6
8
10
12
14
16
18
20
Time, hr
T_i = 90°F
T_{water} = 140°F
$\dot{m}_{steam}$ = 70 lbm/hr
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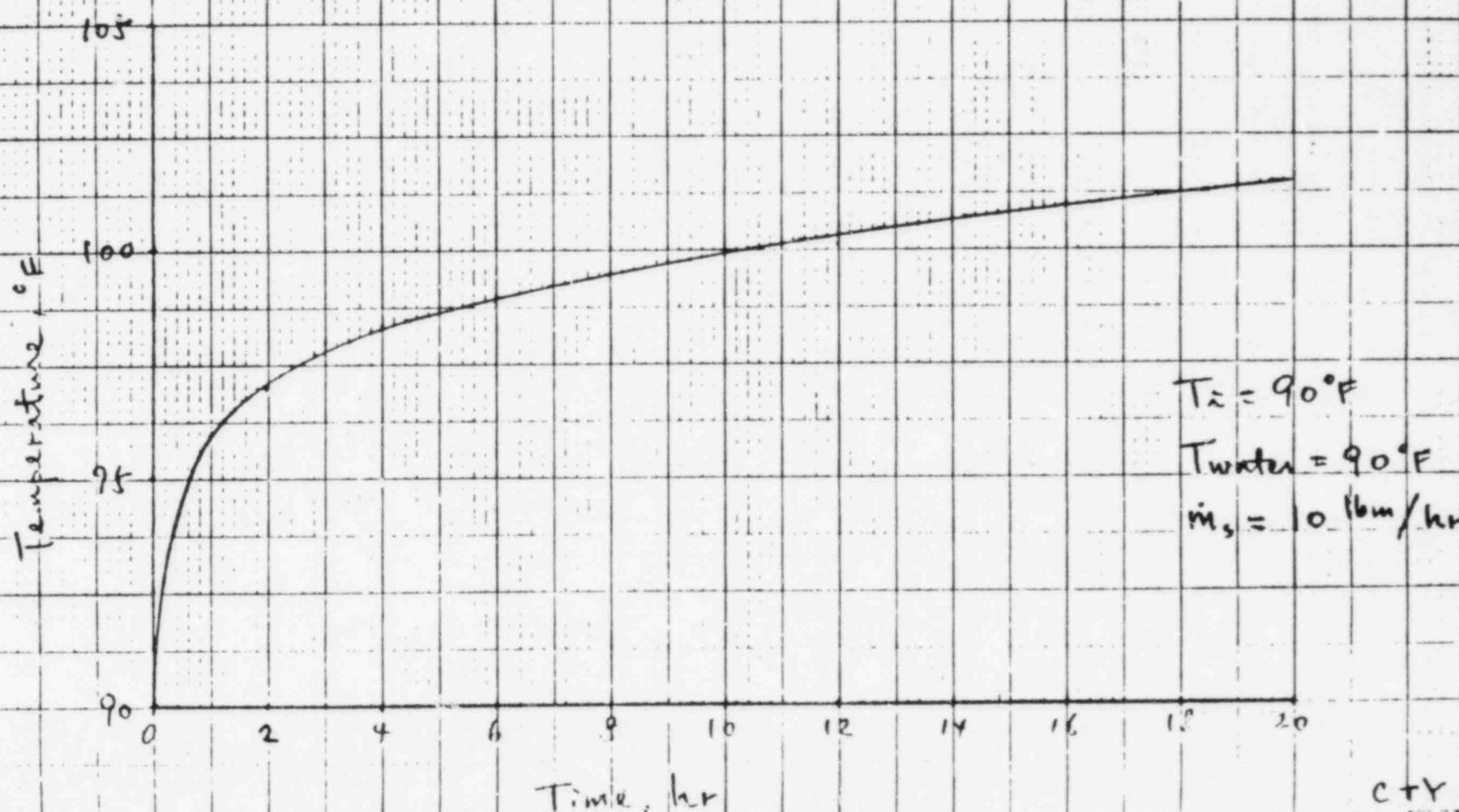
FIGURE 15EB-3

RCIC Room Temperature Responce Following A Station Blackout – SENSITIVITY TO LOW STEAM LEAKAGE RATE



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