The Light company
Houston Lighting & Power South Texas Project Electric Generating Station P. O. Box 289 Wadsworth, Texas 77483

February 7, 1994
ST-HL-AE-4692
File No.: G26
10CFR50.73

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Unit 1
Docket No. STN 50-498
Licensee Event Report 94-001
Regarding Small Gaps in the Reactor
Containment Building Emergency Sumps Screens

Pursuant to 10CFR50.73, Houston Lighting & Power submits the attached Unit 1 Licensee Event Report 94-001 regarding small gaps in the Reactor Containment Building (RCB) emergency sumps screens. This event did not have an adverse effect on the health and safety of the public.

If you should have any questions on this matter, please contact Mr. J. M. Pinzon at (512) 972-8027 or me at (512) 972-8787.

M Sheppard for T.H.C.

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Vice President,
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JMP/eg

Attachment: LER 94-001 (South Texas, Unit 1)

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
South Texas Unit 1	05000 498	1 OF 10

TITLE (4) Small Gaps in the Reactor Containment Building Emergency Sumps Screens

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
01	05	94	94	-- 001 --	00	02	07	94	SOUTH TEXAS, UNIT 2	05000 499
									FACILITY NAME	DOCKET NUMBER 05000

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
OPERATING MODE (9)	5	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	0	20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER
		20.405(a)(1)(iii) X	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iv) X	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jairo Pinzon - Senior Engineer	(512) 972-8027

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 5, 1994, Unit 1 was in Mode 5 at 0% power and Unit 2 was defueled while in a refueling outage. During an inspection of the emergency sumps in the Unit 1 Reactor Containment Building by the NRC Resident Inspectors, openings in the sump screen structures were found that exceeded the designed 1/4" diameter perforations. At each emergency sump screen the frame plate at the floor was warped creating a clearance up to approximately 5/8", and near the solid cover plate at each of the six angle iron supports, cut-out holes approximately 1.5" x 3.5" remained. Some additional gaps, larger than 1/4", were found in the area of the screen seams. Similar conditions were identified in Unit 2. The gaps remained from initial construction fabrication, welding and installation of the screen. The apparent cause of this event is the less than adequate attention to detail during the original design, fabrication, and installation, as well as various surveillance inspections subsequent to original construction. A plant change was issued providing the repair disposition and repairs were implemented. Engineering performed an assessment of the deficiencies and concluded there is no safety significance with regard to these deficiencies.

DESCRIPTION OF EVENT:

On January 5, 1994, Unit 1 was in Mode 5 at 0% power and Unit 2 was defueled while in a refueling outage. During an inspection of the emergency sumps in the Unit 1 Reactor Containment Building (RCB) by the NRC Resident Inspectors, openings in the sump screens were found that exceeded the designed 1/4" perforations. At each emergency sump screen the frame plate at the floor was warped providing a clearance up to approximately 5/8", and near the solid cover plate at each of the six angle iron supports, cut-out holes of approximately 1.5" x 3.5" remained. Engineering and maintenance personnel investigated the extent of the gaps and proposed a resolution. Some additional gaps, larger than 1/4", were found in the area of the screen seams. These gaps remained from initial fabrication, welding and installation of the screen. A plant change was issued providing the repair disposition and repairs were implemented. Similar deficiencies were identified for Unit 2.

On January 6, 1994 the NRC was notified that this event was reportable.

As a result of the investigation of the sump screen deficiencies, on January 20, 1994, the NRC was also notified that the surveillance procedure for the containment sump inspection was inadequate to fully meet the requirements for Technical Specification 4.5.2.d in that it did not require the inspector to physically enter the sump to search for debris. The surveillance was last determined to be performed adequately on March 22, 1991. In August 1993, Unit 1 entered Mode 3 without an adequate surveillance to ensure compliance with Technical Specification 4.5.2.d. This constituted a period of approximately 29 months since the last adequate performance of the containment sump surveillance and entry into Mode 3. This is beyond the 18 months (plus grace period) allowed by Technical Specifications. Therefore, Unit 1 entry into Mode 3 in August 1993, constituted an operation prohibited by Technical Specification 3.5.2.

CAUSE OF EVENT:

The apparent cause of the screen deficiencies is the less than adequate attention to detail during original construction design, fabrication, installation.

- The design drawing should have included an additional note limiting the size of fit-up gaps to less than normal installation tolerances.

- The fabricator, installer and quality control acceptance should have questioned the cutouts in the screen around the angle iron frames in addition to the other gaps/holes following initial installation.

There have been several NRC communications issued to the industry addressing sump screen blockage and debris intrusion into pump suctions. Most of this correspondence, with the exception of information Notice (IN) 89-77, addressed types of debris and their effect on sump suction blockage. IN 89-77 addressed both debris and inadequate sump screens. The South Texas Project Electric Generating Station

CAUSE OF EVENT: (Cont'd)

review of this IN was focused primarily on cleanliness and removal of debris present in the Reactor Containment Building and did not consider sump screen integrity.

Subsequent surveillance inspections did not identify the screen deficiencies since the surveillance procedure did not have guidance to prompt plant personnel to identify holes and gaps during the various surveillance inspections of the emergency sump.

The cause of the inadequate surveillance was the lack of detailed inspection instructions. Regulatory Guide 1.82 does not provide specific inspection criteria. The cause for the operation prohibited by Technical Specifications was as a result of the inadequate surveillance procedure.

ANALYSIS OF EVENT:

The small gaps in the Unit 1 and Unit 2 sump screens created the potential for injected debris to compromise the ability of the containment spray and injection systems to perform their design functions. Therefore this event is considered to be reportable pursuant to 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis of the plant and 10CFR50.73(a)(2)(v) as a condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident for Unit 1 and Unit 2. The inadequate surveillance procedure is reportable for both Units pursuant to 10CFR50.73(a)(2)(i)(B) as an operation prohibited by Technical Specification. The surveillance procedure was inadequate to meet the requirements of Technical Specification 4.5.2.d. In addition, Unit 1 operated in Mode 3 without a surveillance being adequately performed within the time limits established by the Technical Specifications.

The following provides the safety analysis of this event:

1. GENERATION OF DEBRIS

Debris generated from the jet impingement of a loss of coolant accident (LOCA) is the most likely source of material that could bypass the sump screens via the identified deficiencies. RCB walkdowns, in accordance with 0PSP03-XC-0002, provide reasonable assurance that the Containment is clean and free of other foreign material which could migrate to the emergency sumps.

Insulation has traditionally been considered one of the major contributors to post-LOCA debris. Mirror and NUKON insulation are utilized predominantly in the RCBs at STP. Topical Report

ANALYSIS OF EVENT: (Cont'd)

OCF-1 concludes that NUKON will not deteriorate or lose its mechanical integrity in a post-LOCA environment. Similarly, mirror insulation, if dislodged in a LOCA, would settle in relatively large pieces and would not be a factor adverse to emergency core cooling system (ECCS) recirculation.

2. TRANSPORTATION OF DEBRIS

The six 1.5" x 3.5" holes near the top of the structure do not significantly degrade the integrity of the screen because of the arrangement of the screen design and the normal flow configuration. A 4" x 4" structural angle on the outer edge of the screen structure directs the normal flow path below the 1.5" x 3.5" openings and entrained matter is caught on the 1/4" perforated screen as intended. See Figure 1.

The potential impact of the remaining holes and gaps is twofold. The 5/8" and 3/8" gaps and holes provide potential paths for an increase in particulate size over the 1/4" design criteria and increase the total available flow area through which suspended particulate matter may pass.

From the aspect of increased particulate size, there is little impact because of low design flow velocities. The approach velocity at the first screen is 0.12 ft/sec, which is well below the recommended design velocity of 0.2 ft/sec in Regulatory Guide (RG) 1.82. Since the flow upstream of the screens would be less than or equal to 0.12 ft/sec, particulate matter with a specific gravity of 1.05 or greater would settle out on the containment floor and would not migrate to the screens.

From the aspect of an increase in total flow area, the gaps and holes have been conservatively estimated to represent a total maximum increase of 1.36% in the current available sump screen flow area. This increase is considered insignificant compared with the total flow area.

Due to the low flow velocities, the concentrated area of potential debris that would be available for ingestion to the sump is extremely small. Given the location and geometry of the emergency sumps, it is highly unlikely that remote debris would propagate to the emergency sumps. With the exception of the aforementioned deficiencies, the emergency sumps meet the intent of the RG 1.82.

3. IMPACT ON EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEM (ECCS & CS) PUMPS

The ECCS & CS pump design and fabrication requirements specifically address pump operation during recirculation of Containment sump water. The pump manufacturer was required to allow

ANALYSIS OF EVENT: (Cont'd)

for sump water to contain solid particles of concrete, insulation and paint flakes which could pass through the screen and into the suction of the pumps. This was satisfactorily demonstrated by a thermal transient test with injection of suspended particles (less than 1/4 inch). The pump performance variation between pre and post transient test was minimal and well within specifications. The head curve deviated less than 1% between tests and vibration levels were unchanged. Additionally, the seal area showed no sign of leakage before, during, or after the transient. Any entrained particle must enter the outer barrel and make a 180° flow direction change prior to entering the first stage impeller. A dense particle (unlikely to be transported to the screen or migrate from the sump) would be forced against the bottom of the outer barrel. Light objects would enter the first stage impeller and pass through the pump.

Although transport of debris in excess of design is unlikely, the suspended particle pump performance tests and inherent geometry substantiate that there would be no adverse impact on the ability of the ECCS and CS pumps to perform their design functions.

4. NO SIGNIFICANT HAZARDS (CONTAINMENT HEAT REMOVAL, RADIOLOGICAL OR CORE COOLING)

(i) CONTAINMENT HEAT REMOVAL

One of the functions of the Containment Spray System (CSS) is heat removal from Containment following a design basis accident (DBA). Per UFSAR Section 6.2.1, the Containment DBA is a double-ended pump suction guillotine rupture (LOCA 2). The energy inventory for this accident is depicted in Figure 2 (UFSAR Fig. 6.2.1.1-1A) and the Containment vapor and sump temperatures for this accident are depicted in Figure 3 (FSAR Fig. 6.2.1.1-11).

At the assumed time of initiation of recirculation with suction from the Containment sumps (1216 seconds), the Containment sump temperature is significantly higher than Containment vapor temperature and heat removal by Containment spray begins to decrease (and is essentially zero at 8000 sec) as shown in Figure 2.

A loss of Containment spray capability by total blockage of the spray header nozzles at the initiation of recirculation flow will not have a negative impact on energy removal capability following a DBA and is, therefore, not a safety concern.

ANALYSIS OF EVENT: (Cont'd)

(ii) RADIOLOGICAL

In addition to reducing the pressure and temperature in the Containment following a postulated LOCA, the CSS also provides a mechanism to scrub iodine from the Containment atmosphere. The recirculation phase of the LOCA begins after approximately 20 minutes. It is at this time that any degradation of the CSS due to loss of sump screen integrity might begin to potentially impact the design basis dose calculation.

An examination of the dose calculations performed for the LOCA reveals that spray removal of elemental iodine ends at 16.8 minutes after the accident. Spray removal of particulate iodine ends at 6.3 hours. The sprays are not assumed to remove organic iodine from the Containment atmosphere. Therefore, the impact of potentially reduced CSS flow would be limited to a decrease in the removal rate for particulate iodine. Since approximately 42% of the offsite thyroid dose is due to elemental iodine and 54% due to organic iodine, the impact of reduced CSS flow on the offsite doses is small.

Calculation NE-CE-94-01-00 indicates that a reduction of two-train CSS flow (during the recirculation phase) of up to 75% may be tolerated without exceeding 10CFR100 limits. In the bounding hypothetical case considering the two train CSS flow reduced to zero, the offsite dose would only increase by approximately 15%. The control room and Technical Support Center (TSC) thyroid doses would increase by only 11%. The offsite and control room doses would not exceed 10CFR100 dose limits. Additionally, the thyroid dose to TSC personnel would exceed 10CFR100 limit of 30 rem by only approximately 3 rem.

(iii) CORE COOLING

The analysis of a large break LOCA is divided into three phases: blowdown, refill and reflood. Blowdown is the time between full power operation until zero break flow is first calculated; refill is from the end of blowdown to the time the ECCS fills the vessel lower plenum; and reflood begins when water starts moving into the core and continues until the end of the transient.

Depending on the specific accident assumptions, the core reflood ends at 100 to 280 seconds (UFSAR Table 6.2.1.1-10). At this time, the transient has ended and long term core cooling has been established. Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady-state levels associated with dissipation of residual heat generation. The recirculation phase of a LOCA begins at approximately 20 minutes (1200 seconds) after the accident. After the water level

ANALYSIS OF EVENT: (Cont'd)

of the RWST reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by automatically switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the Containment emergency sumps by the low-head and high-head safety injection pumps and returned to the RCS cold legs. The main objective of the ECCS now is to keep the core covered. The recirculation phase of a LOCA begins when the ECCS suction is switched from the RWST to the containment sumps. The identified sump screen defects would not prevent the low-head and high-head safety injection from performing the long-term cooling functions, i.e.,the core would remain flooded.

(5) CONCLUSION

- The low flow velocities and design conservatism would inhibit the propagation of debris to the emergency sumps and preclude debris from entering into the ECCS/CS suction piping.

- In the highly unlikely event of a 75% loss of the two-train CSS design flow due to blockage, there are no negative consequences to containment pressure/temperature mitigation or core cooling and only minimal impact on the available design margin for control room, TSC, and offsite doses.

- Therefore, the gaps found in the Unit 1 and Unit 2 emergency sumps have no adverse effect on the operation of the plant.

CORRECTIVE ACTIONS

The following corrective actions have been or will be taken as a result of this event:

1. Repair dispositions were initiated to correct the noted deficiencies in the ECCS sump as well as the design drawings. The repairs on Unit 1 are complete and the Unit 2 repairs will be completed prior to entry into Mode 4 from the current refueling outage.

2. The emergency sump inspection surveillance procedure has been corrected to require sump entry and has been enhanced to include quantitative inspection criteria for gaps and holes in the screen structure.

ADDITIONAL INFORMATION

Unit 1 LER 88-063 was previously submitted regarding the failure to install vortex breakers in the containment emergency sumps. The cause of this event was the failure to clearly establish responsibility and accountability for the installation of the vortex breakers during initial installation.

Figure 1

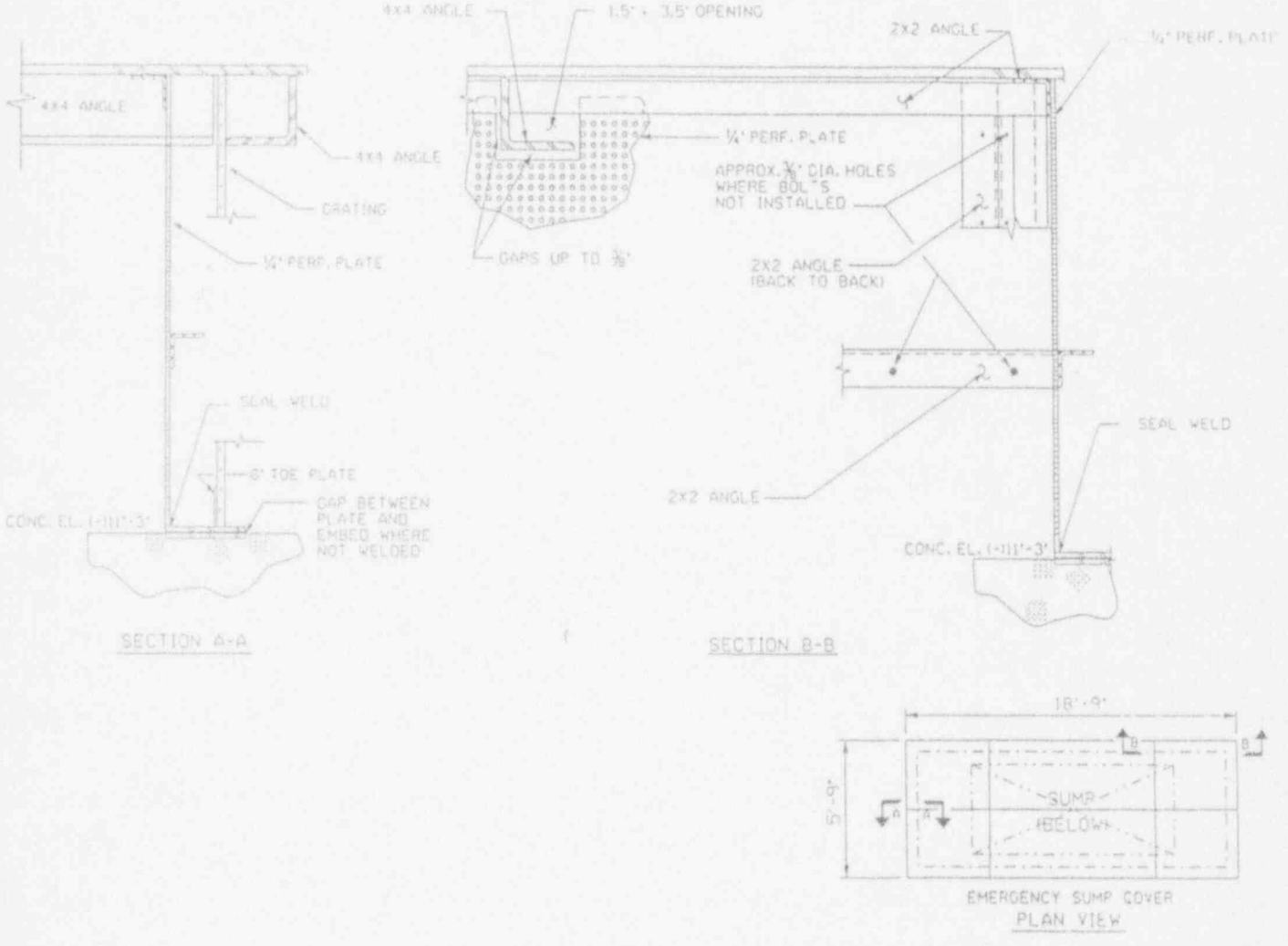

4X4 ANGLE
1.5" x 3.5" OPENING
2X2 ANGLE
¼" PERF. PLATE
4X4 ANGLE
¼" PERF. PLATE
APPROX. ⅜" DIA. HOLES WHERE BOLTS NOT INSTALLED
4X4 ANGLE
GRATING
¼" PERF. PLATE
GAPS UP TO ⅜"
2X2 ANGLE (BACK TO BACK)
SEAL WELD
6" TOE PLATE
CONC. EL. (-)111'-3"
GAP BETWEEN PLATE AND EMBED WHERE NOT WELDED
2X2 ANGLE
SEAL WELD
CONC. EL. (-)111'-3"
SECTION A-A
SECTION B-B
18'-9"
5'-9"
SUMP
BELOW
EMERGENCY SUMP COVER
PLAN VIEW

Figure 2

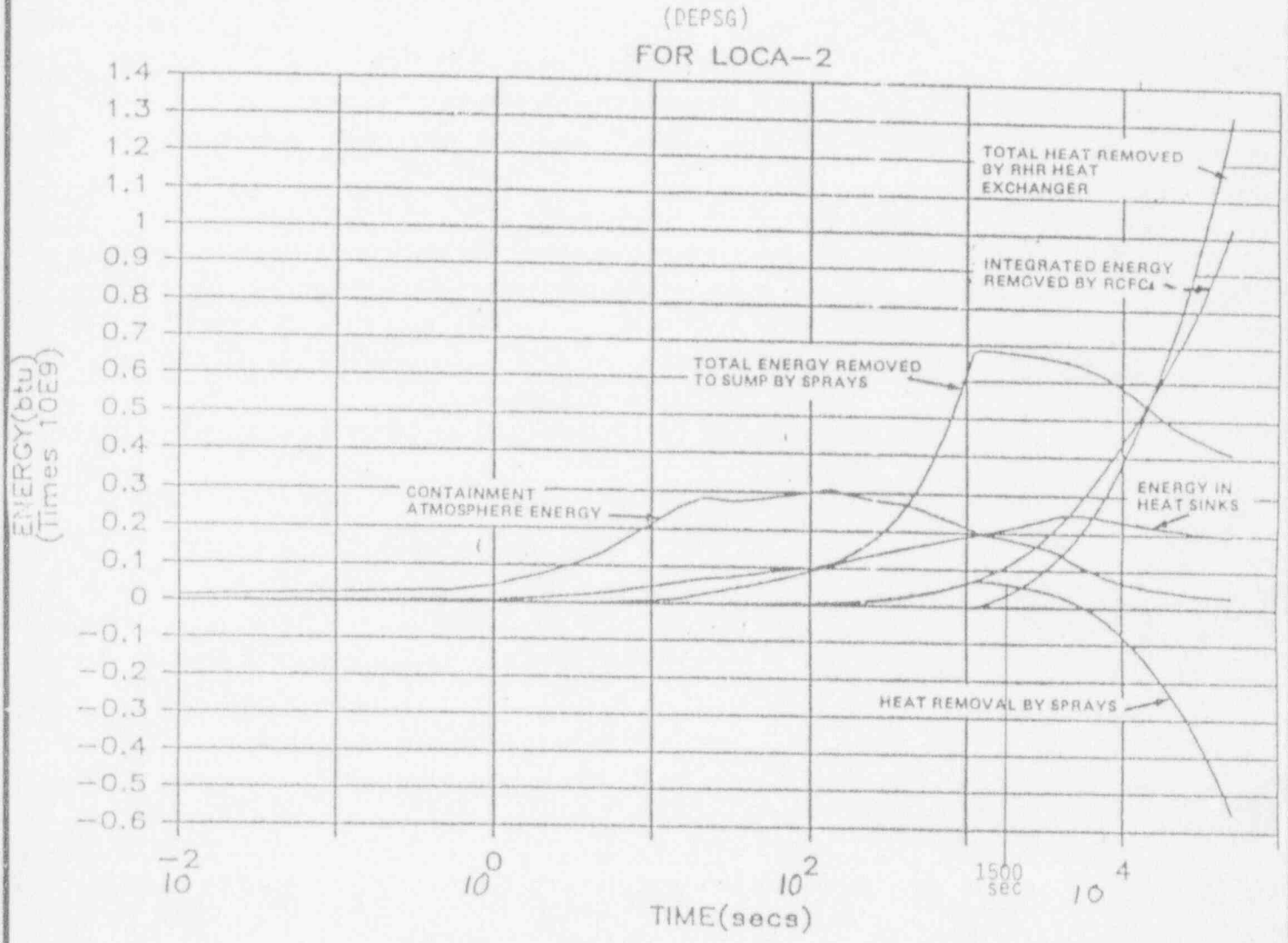
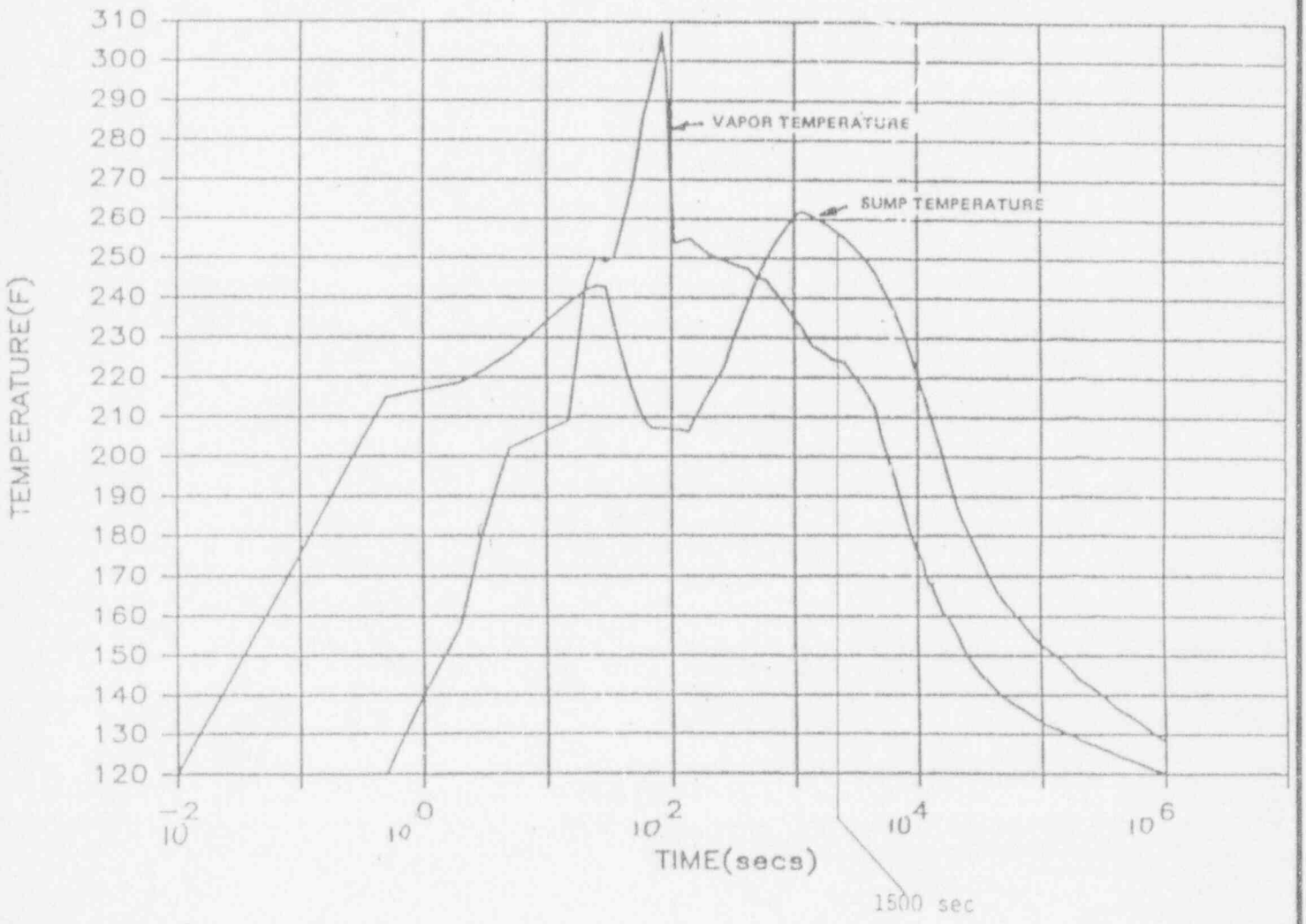

(DEPSG)
FOR LOCA-2
ENERGY(btu)
(Times 10E9)
1.4
1.3
1.2
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-0.1
-0.2
-0.3
-0.4
-0.5
-0.6
TOTAL HEAT REMOVED BY RHR HEAT EXCHANGER
INTEGRATED ENERGY REMOVED BY RCFC
TOTAL ENERGY REMOVED TO SUMP BY SPRAYS
CONTAINMENT ATMOSPHERE ENERGY
ENERGY IN HEAT SINKS
HEAT REMOVAL BY SPRAYS
10^-2
10^0
10^2
1500 sec
10^4
TIME(secs)

Figure 3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250
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200
190
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160
150
140
130
120
TEMPERATURE(F)
VAPOR TEMPERATURE
SUMP TEMPERATURE
10^-2
10^0
10^2
10^4
10^6
TIME(secs)
1500 sec