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 2	05000 499	1 OF 4

TITLE (4) Inadvertent Test-Mode Starts of Standby Diesel Generators

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
06	07	94	94	-- 005 --	02	02	14	95	South Texas, Unit 1	05000 498
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)					
		20.402(b)	20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	35	20.405(a)(1)(i)	50.36(c)(1)		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)	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)	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 - Staff 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 June 7, 1994, Unit 2 was in Mode 1 at 35% power. Standby Diesel Generator 22 had satisfactorily completed the monthly Standby Diesel Generator Operability test, had been unloaded, and placed in cooldown. Approximately three minutes into the cooldown cycle, the Standby Diesel Generator trouble alarm annunciated, generator voltage and frequency returned to normal operating parameters and it was recognized that Standby Diesel Generator 22 had received an unexpected test-mode start signal. The most likely cause of this event was a voltage spike generated by an emergency fuel oil solenoid deenergizing. The inadvertent test-mode start would not prevent any of the Standby Diesel Generators from performing its safety function of coping with a loss of offsite power.

An additional inadvertent test-mode start occurred on Standby Diesel Generator 22 on July 25, 1994. Also, on August 1, 1994, Unit 1 Standby Diesel Generator 13, received a spurious test-mode start signal that resulted in a start of a portion of the Standby Diesel Generator auxiliaries with no actual start of the Standby Diesel Generator. The causes of these events were also attributed to the previously identified cause.

Corrective actions included installing noise suppression devices to reduce the susceptibility of inadvertent starts due to the noise

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DESCRIPTION OF EVENT:

On June 7, 1994, Unit 2 was in Mode 1 at 35% power. Standby Diesel Generator 22 had satisfactorily completed performance of the monthly Standby Diesel Generator Operability test, had been unloaded, and placed in the cooldown cycle. Approximately three minutes into the cooldown cycle, the Standby Diesel Generator trouble alarm annunciated, generator voltage and frequency returned to normal operating parameters and it was recognized that Standby Diesel Generator 22 had experienced an unexpected start (with no operator action) while in the cooldown cycle. The start was verified to originate from a test-mode start signal and not an emergency start signal. At 1630 hours, the Standby Diesel Generator was placed in the cooldown cycle for the second time, and was successfully secured after a five minute cooldown. This indicated that the voltage spike within the start circuit was intermittent or that it varied in magnitude sufficient to induce a start signal infrequently. The inadvertent start occurred when the first cooldown timer deenergized.

When the cooldown timers deenergized emergency fuel oil solenoid 20-F01 and energized emergency fuel oil solenoid 20-F02 (within the start circuit), a voltage spike was generated within the start circuit. Investigation indicates that the fiber-optic isolation board allowed this voltage spike to induce a start. Previous modifications to reduce the magnitude of external voltage spikes were not intended or capable of eliminating voltage spikes generated within the start circuit.

On July 25, 1994, while Unit 2 was in Mode 1 at 100% power, Standby Diesel Generator 22 experienced a test-mode start for no apparent reason. No testing, maintenance, or other work activities were in progress at the time of the start.

On August 1, 1994, while Unit 1 was in Mode 1 at 100% power, Standby Diesel Generator 13 received an apparent test-mode start signal that resulted in a start of a portion of the Standby Diesel Generator auxiliaries with no start of the Standby Diesel Generator. Current indications are that a "spurious" start command occurred in the non-class portion of one of the two start circuits.

The filtering network and circuit modifications, to reduce susceptibility of the fiber-optic boards to voltage spikes, had been previously installed. These are the first starts from the standby mode since these modifications have been made on each of the diesels. Testing has shown that the fiber-optic boards are sensitive to high temperatures. The higher the ambient temperature, the more likely voltage spikes will trigger the fiber-optic board to start the Standby Diesel Generator. The filtering network reduces the magnitude of the spikes that are getting through; however, voltage spikes cannot be totally eliminated.

CAUSE OF EVENT:

The most likely cause of the June 7, 1994, event was electrical noise generated by the emergency fuel oil solenoid deenergizing which was perceived by the fiber-optic test circuit to be a "Start" signal.

The cause of the July 25, 1994 and August 1, 1994 starts are believed to be the fiber-optic board's susceptibility to noise in conjunction with transient DC spikes. This caused the fiber-optic test circuits to perceive a "Start" signal.

Repeat starts on the Standby Diesel Generators after initial fiber-optic board repair activities were likely caused by damage or aging from soldering of replacement transistors on the fiber-optic boards. This repair was deemed necessary due to the non-availability of new replacements at that time.

ANALYSIS OF EVENT:

The inadvertent test-mode starts of Standby Diesel Generators are classified Engineered Safety Feature Actuations and are reportable pursuant to 10CFR50.73(a)(2)(iv).

The Standby Diesel Generators are part of the Class 1E 4.16 KV AC Power System. The Class 1E 4.16 KV AC Power System is composed of three trains designed to provide a reliable source of power to the safety-related equipment essential to all modes of plant operation including emergency shutdown following any design basis event. Upon a loss of off-site power, each of the three Standby Diesel Generators supply backup power to the associated 4.16 KV bus to mitigate the consequences of postulated accidents. This inadvertent Standby Diesel Generator test-mode start did not affect the ability of the Standby Diesel Generator to perform its intended safety function.

CORRECTIVE ACTIONS:

The following corrective actions have been taken to date to prevent unintentional starts (These corrective actions have been previously described in Unit 2 Licensee Event Report 94-003):

1. Equipment repair, which replaced parts found to be weak or non-functioning.

2. Electrical noise reduction was added to install:

   a) filters to the DC power supplies to reduce the magnitude of external circuit electrical noise seen in the engine control panel, and

   b) ceramic capacitors across the Allen Bradley "Run" relays to reduce the inductive response contributing to the noise induced starts.

This action has been completed on all Standby Diesel Generators. The electrical noise reduction modifications will lower the probability but not eliminate the possibility of additional unexpected test-mode starts.

CORRECTIVE ACTIONS: (Continued)

3. As a result of an ongoing assessment of the Diesel Generator Control Panel reliability, an evaluation will be performed to determine the feasibility of replacing the entire control panel with Programmable Logic Control (current state-of-the-art technology), due to obsolescence of components in the present installation. This evaluation will be completed by December 31, 1995.

4. A plan of action has been put into place to replace the "repaired" fiber-optic boards with new ones. To date, the Fiber-optic boards for Standby Diesel Generators 12, 13, 21, and 22, have been replaced with new fiber-optic boards. Replacement of remaining Standby Diesel Generators should occur at the next Limiting Condition for Operation Action Statement following availability of new replacement boards.

5. Preventive maintenance activities for replacement of the fiber-optic boards on a five year cycle have been approved and scheduled.

Should another test-mode start occur on any of the South Texas Project Standby Diesel Generators, a supplement to this License Event Report will be submitted to keep the Nuclear Regulatory Commission informed.

ADDITIONAL INFORMATION:

The Standby Diesel Generators are type KSV-20-T, four stroke, turbocharged engines manufactured by Cooper Energy Services.

During the past three years four events were reported regarding inadvertent starts of Standby Diesel Generators:

Unit 2 Licensee Event Report 93-015 was submitted documenting the inadvertent start of Standby Diesel Generator 22. The cause was attributed to a spurious failure of a transistor.

Unit 1 Licensee Event Report 93-023 was submitted documenting an inadvertent start of Standby Diesel Generator 12 during testing. The cause of this even was most likely the result of an electrical arc between the Standby Diesel Generator panel and test equipment.

Unit 2 Licensee Event Report 94-001 was submitted documenting an inadvertent start of Standby Diesel Generator 21. The inadvertent start was cause by a combination of two component failures: a weakened transistor in the Non-Class 1E fiber-optic start circuits and a faulty power supply external to the start circuit that induced spikes into the fiber-optic start circuits.

Unit 2 Licensee Event Report 94-003 was submitted documenting the inadvertent test-mode starts of Standby Diesel Generators 21, 22, and 13. The cause of these events was determined to the fiber-optic boards' susceptibility to noise in conjunction with transient DC spikes.

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

February 14, 1995
ST-HL-AE-4990
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
Revision 1 to Voluntary Licensee Event Report 92-018
Pressurizer Safety Valve Setpoints and
Main Steam Safety Valve Outside Required Tolerance

Pursuant to 10CFR50.73, Houston Lighting & Power Company submits the attached revision to Unit 1 Voluntary Licensee Event Report 92-018, regarding Pressurizer Safety Valve and Main Steam Safety Valve setpoints outside required tolerance. This event did not have an adverse effect on the health and safety of the public.

The purpose of this revision is to incorporate information regarding several Main Steam Safety Valve and Pressurizer Safety Valve setpoints that were found to be out of tolerance during past outages. The out of tolerance conditions of the Main Steam and Pressurizer Safety Valves were evaluated for impact to the design basis analysis. The safety analysis concluded that all criteria are met, thus, the past safety analyses bounded each situation. This revision includes an expanded safety analysis regarding out of tolerance conditions and their impact on Unit operation. Changes are indicated by revision bars.

This Licensee Event Report was originally submitted prior to the issuance of the guidance which was provided by letter from Mr. S. J. Collins, Nuclear Regulatory Commission Division Director of Reactor Safety, to Mr. W. T. Cottle, Nuclear Group Vice President, Houston Lighting & Power, dated December 8, 1993. This guidance clarified interpretation of reporting requirements related to out of tolerance conditions on primary or secondary safety relief valves. Prior to the issuance of this guidance, Houston Lighting & Power had a different interpretation of the application of this reporting requirement.

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The original Licensee Event Report was focused on Pressurizer Safety Valves and did not include identification of similar conditions on the Main Steam Safety Valves or prior occurrences of these conditions. Additionally, the original Licensee Event Report did not provide an evaluation of the impact of the out of tolerance condition on the design basis analysis. After analysis of the out of tolerance data, it was determined that the out of tolerance conditions experienced as a result of the Pressurizer Safety and the Main Steam Safety Valves did not result in a condition that was unanalyzed or outside the design basis of the plant.

Houston Lighting & Power will ensure that this guidance is appropriately applied to out of tolerance conditions for surveillances in future outages. Personnel have been sensitized to the analysis and reporting requirements for these type of conditions.

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

T. H. Cloninger
Vice President,
Nuclear Engineering

JMP/lf

Attachment: Revision 1 to Voluntary LER 92-018 (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

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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) Pressurizer Safety Valve and Main Steam Safety Valve Setpoints Outside Required Tolerance

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
10	21	92	92	-- 018 --	01	02	14	95		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N/A	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	000	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X OTHER
		20.405(a)(1)(iii)	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)	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)	Voluntary

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jairo Pinzon - Staff 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 October 21, 1992, Unit 1 was defueled during a refueling outage. After being reset at the third refueling outage for Unit 1 (1RE03) to the specified value of 2485 psig ± 1.0%, the setpoints for the Unit 1 Pressurizer Safety Valves were found to be 6.7% below to 4.3% above the required setpoint during the fourth refueling outage (1RE04). This is a deviation from the ± 1.0% Technical Specification requirement. The Unit 1 Pressurizer Safety Valves (PSV-3450, 3451 & 3452) had been sent to Wyle Laboratories for setpoint verification testing. Pressurizer Safety Valve setpoint drift is an industry-wide problem which has been known for some time. The Westinghouse Owners Group has addressed this generic problem and WCAP-12910, which makes specific recommendations relative to PSV setpoint verification testing, has been issued. Corrective actions include modifying the test procedure to test the Pressurizer Safety Valve lift setpoint on saturated steam as recommended by WCAP-12910.

In addition to the above, additional Pressurizer Safety Valve out of tolerance conditions were discovered which were not previously identified. Several Main Steam Safety Valve setpoints were also found to be outside the required tolerances.

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DESCRIPTION OF EVENT:

On October 21, 1992, Unit 1 was defueled during a refueling outage. The three Unit 1 Pressurizer Safety Valves (PSV-3450, 3451 & 3452) were sent to Wyle Laboratories for setpoint verification testing. After being reset at the third refueling outage for Unit 1 (1RE03) in the Spring of 1991 to the specified value of 2485 psig ± 1.0%, the setpoints were determined during the fourth refueling outage (1RE04) to be in the range of 2318 psig to 2591 psig or from 6.7% below to 4.3% above the require setpoint. This is a deviation from the ± 1.0% Technical Specification requirement.

Table 1 presents the setpoint data for Unit 1. The results were obtained at the first, third and fourth refueling outages. No testing was conducted in the second refueling outage of Unit 1.

The Pressurizer Safety Valves were originally tested (during 1RE01, 1RE03, and first setpoint test during 1RE04) with a nitrogen backed loop seal (water). The valve was installed on the test stand in a normal operating position and test equipment (thermocouples, pressure transducers, etc.) installed. The valve and loop seal are then thermally stabilized to specified temperatures. The valve is considered thermally stable when bonnet temperature changes differ no more than 5°F between two consecutive readings taken 30 minutes apart and when the valve body temperature changes no more than 10°F in 30 minutes. The inlet pressure during stabilization is 2236 ± 10 psig (90% of setpoint pressure). A leak check is then performed. Following the leak check, the setpoint actuation is performed by increasing the inlet pressure at ramp rate until the valve actuates. Following actuation, the pressure is returned to the leakage test pressure, ± 10 psig, as soon as practical. The inlet pressure is maintained at this pressure until the next actuation. There is a minimum of 10 minutes between each run and a minimum of three set pressure actuations.

The Pressurizer Safety Valves were reset on steam in accordance with WCAP-12910 prior to the end of 1RE04.

Crosby HB-BP-86 6N8 Pressurizer Safety Valves are spring-loaded with an adjusting bolt nut which increases or decreases the spring force to achieve the relief valve opening. Once a safety valve is set and the adjusting bolt nut is torqued, the lift setpoint should change only if the valve experiences leakage or if the valve is actuated. Neither of these events had occurred with the South Texas Project valves.

DESCRIPTION OF EVENT: (Continued)

WCAP-12910 summarizes industry data (EPRI Report NP-2770-LD) and presents the results of the Westinghouse Owners Group Program which further evaluates the Pressurizer Safety Valve set pressure drift phenomena. WCAP-12910 concludes that (for the specific valves tested) the set pressure shift did not exceed 1.0% for spring-loaded safety valves based on a very accurate spindle measurement (0.1-05 mil) movement and a pressure curve slope change (0.1-0.5 mil). For South Texas Project Pressurizer Safety Valves, there is fair agreement between "pops" in the test stand for a series of tests, but a significant variance is seen when the data is taken about 18 months apart. The South Texas Project 6N8 type PSV is not included in the WCAP summary. The 6M6 type PSV safety valve is included, and except for differences in the spring and orifice, is expected to respond in a similar manner.

The WCAP recommends that Pressurizer Safety Valves be set on a steam environment, regardless of the medium used during operation. With a steam setting, the spindle movement is sharper, more visible and more reproducible.

Earlier Pressurizer Safety Valve setpoint testing for the South Texas Project (up to and including 1RE03, 2RE01 and 2RE02) was conducted at the Westinghouse Western Service Center while the 1RE04 testing was conducted at Wyle Laboratories. The same test procedure was used and no data bias can be seen in favor of either test facility.

On January 16, 1995, an issue was raised by a Nuclear Regulatory Commission Inspector regarding the safety analysis of operation of the Unit with the Pressurizer Safety Valves in the out of tolerance condition. The issue related to the fact that this Voluntary Licensee Event Report did not document the impact of the Pressurizer Safety Valve out of tolerance condition in the "Analysis of Event" section. In addition, it was discovered that there have been instances where a number of the Main Steam Safety Valves have been found to be out of tolerance during the past refueling outage surveillances. Table 2 documents the as found condition for the tested valves during each Unit 1 outage.

A reportability review was immediately initiated to evaluate the as found condition of the Pressurizer Safety Valves and the Main Steam Safety Valves. Discussions with the Nuclear Regulatory Commission, Region IV, Engineering Management, indicated that the reportability of these past events should hinge on whether or not the condition resulted in the plant being outside the design basis. The reportability review concluded that although these valves were found to be out of tolerance, these conditions were bounded by the design basis of the plant. This is further discussed in the "Analysis of Event" section.

CAUSE OF EVENT:

The observed setpoint drift on the South Texas Project Pressurizer Safety Valves is apparently the result of the inherent valve characteristics (i.e. the massive spring assembly) and the limited capability to accurately determine the first point of valve spindle movement. Inherent valve characteristics include the differences between the test and operating conditions and especially the effect of temperature in the valve seat, plug area and on the spring. Pressurizer Safety Valve setpoint drift is an industry-wide problem which has been known for some time. The Westinghouse Owners Group has addressed this generic problem in WCAP-12910 which makes specific recommendations relative to Pressurizer Safety Valve setpoint verification testing.

The cause of the Main Steam Safety Valve out of tolerance conditions is an industry known problem regarding setpoint drift on the large pressure release devices.

ANALYSIS OF EVENT:

Houston Lighting & Power recognizes the guidance provided by letter from Mr. S. J. Collins, Nuclear Regulatory Commission Division Director of Reactor Safety, to Mr. W. T. Cottle, Nuclear Group Vice President, Houston Lighting & Power, dated December 8, 1993. This guidance clarified interpretation of reporting requirements of out of tolerance conditions of primary or secondary safety relief valves. Prior to the issuance of this guidance, Houston Lighting & Power had a different interpretation of the application of this reporting requirement. After analysis of the out of tolerance data, it was determined that the out of tolerance conditions experienced as a result of the Pressurizer Safety and the Main Steam Safety Valves did not result in a condition that was unanalyzed or outside the design basis of the plant.

Pressurizer Safety Valves

The Pressurizer Safety Valves protect the integrity of the Reactor Coolant Systems from overpressurization. The integrity of the Reactor Coolant System is required to prevent the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere. The reactor vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the American Society of Mechanical Engineers (ASME) Code for Nuclear Power Plants. The ASME Code permits a maximum transient pressure of 110% (2735 psig) of design pressure. Technical Specification Bases 2.1.2 and 3/4.4.2 identify 2735 psig as the safety limit for the Reactor Coolant System.

An assessment of the safety consequences and implication of the Pressurizer Safety Valves out of tolerance conditions was performed. Table 1 shows the as found values and the percentage of out of tolerance of the Pressurizer Safety Valves for each loop. These measurements were taken at each of the Unit 1 refueling outages.

ANALYSIS OF EVENT: (Continued)

Table 1 identifies that for Unit 1 all the Pressurizer Safety Valves were out of tolerance during the first, third and fourth refueling outages. Of the nine out of tolerance conditions, four were high (or positive) and the remaining five were low (or negative). Positive out of tolerance conditions may result in increasing the calculated peak Reactor Coolant System pressure results for accidents that take credit for the Pressurizer Safety Valve actuation. Negative out of tolerance conditions may result in decreasing the calculated minimum Departure From Nucleate Boiling Ratio for accidents that take credit for Pressurizer Power Operated Relief Valve actuation.

The accidents that positive out of tolerance conditions may impact are:

1) Uncontrolled Rod Cluster Control Assembly withdrawal from a subcritical condition and at power (Condition II event) (FSAR section 15.4.1)
2) Loss of external electrical load (Condition II event) (FSAR section 15.2.2)
3) Turbine trip (Condition II event) (FSAR section 15.2.3)
4) Locked rotor (Condition IV event) (FSAR section 15.3.3)
5) Rod Cluster Control Assembly ejection (Condition IV event) (FSAR section 15.4.8)

The single most severe positive out of tolerance condition was 4.27 % or 2591 psig, which was identified during the fourth refueling outage. The other two Pressurizer Safety Valves were shown to be -2.94 % and -6.72%, which more than offsets the effect of the one being high by 4.27%. The worst case scenario for the Pressurizer Safety Valves was measured during the third refueling outage when all three Pressurizer Safety Valves were out of tolerance high simultaneously. Averaging the tolerances of the three safety valves yields a group out of tolerance of 2.7%. The past safety analysis assumes a 3% accumulation added to the setpoint. An evaluation was made that showed the past safety analyses bounded the as measured Pressurizer Safety Valve setpoint cases. Therefore, the past safety analyses bounds the most severe positive out of tolerance condition.

The accidents that the negative out of tolerance conditions may impact are:

1) Uncontrolled Rod Cluster Control Assembly withdrawal from a subcritical condition and at power (Condition II event) (FSAR section 15.4.1)
2) Complete loss of flow (Condition III event) (FSAR section 15.3.2)
3) Rod out of position (Condition II event) (FSAR 15.4.3)
4) Hypothetical steamline break (Condition IV event) (FSAR section 15.1.5)
5) Locked rotor (Condition IV event) (FSAR section 15.3.3)
6) Dropped Rod Cluster Control Assembly (Condition II event) (FSAR section 15.4.3)

ANALYSIS OF EVENT: (Continued)

All of the Pressurizer Safety Valve negative out of tolerance conditions, except one, are bounded by the Pressurizer Power Operated Relief Valve setpoint of 2235 psig. The exception was found during the fourth refueling outage, which actuated at 2318 psig or 6.72% below the safety valve setpoint. A more limiting Departure From Nucleate Boiling Ratio is calculated when credit is taken in the analysis for the Pressurizer Power Operated Relief Valve to operate and limit the pressure rise to a lower peak than if only the Pressurizer Safety Valves are assumed to be available. The complete loss of flow accident represents the most limiting accident with respect to Departure From Nucleate Boiling Ratio for the fourth refueling outage conditions and the only Departure From Nucleate Boiling Ratio analysis where the calculated minimum Departure From Nucleate Boiling Ratio occurs at a Pressurizer pressure greater than 2318 psig. During a complete loss of flow accident, pressure is rising as power and flow are decreasing. The pressure at which minimum Departure From Nucleate Boiling Ratio is calculated for the Unit 1 fourth refueling outage complete loss of flow accident is approximately 2347.1 psig. Therefore, with a Pressurizer Safety Valve possibly actuating before the Power Operated Relief Valve and yielding a lower statepoint pressure than analyzed at minimum Departure From Nucleate Boiling Ratio, a lower Departure From Nucleate Boiling Ratio may result.

With the Pressurizer Safety Valve actuating before the Power Operated Relief Valve would have actuated for the worst case Departure From Nucleate Boiling Ratio, the accident analysis did not specifically bound this out of tolerance condition. A penalty can be applied to the past complete loss of flow accident analysis to account for the lower pressure. This penalty is determined by evaluating the WRB-1 Critical Heat Flux correlation for both the analyzed and the out of tolerance conditions. The difference in calculated Critical Heat Flux is 3.3%, which also corresponds to a decrease in the Departure From Nucleate Boiling Ratio of approximately 3.3%. The Unit 1 Fourth refueling outage safety analysis included an available 3.9% margin in Departure From Nucleate Boiling Ratio, which is sufficient to offset this out of tolerance penalty. Therefore, the most negative Pressurizer Safety Valve out of tolerance condition is not a concern with the Departure From Nucleate Boiling Ratio for the Unit 1 fourth refueling outage.

Main Steam Safety Valves

The safety function of the Main Steam Safety Valves is to protect the Steam Generator and the piping upstream of the Main Steam Isolation Valves from overpressurization under all modes of operation. During a loss of load/turbine trip event, operability of the Main Steam Safety Valves ensure that the secondary system pressure will be limited to within 110% (1413.5 psig) of its design pressure of 1285 psig.

An assessment of the safety consequences and implication of the Main Steam Safety Valves out of tolerance conditions was performed. Table 2 shows the as found values and the percentage of out of tolerance of the Main Steam Safety Valves for each loop. These measurements were taken at each of the past refueling outages.

ANALYSIS OF EVENT: (Continued)

The primary concern addressed in this safety analysis is Main Steam System overpressurization. First, the single most positive out of tolerance condition of the Main Steam Safety Valves occurred during the third Unit 1 refueling outage. Second, four Main Steam Safety Valves on one Steam Generator were out of tolerance during the first and third refueling outage; however, only one of the four out of tolerance conditions for the first refueling outage were positive. Finally, during the third Unit 1 refueling outage, 12 of 20 Main Steam Safety Valves were discovered to be out of tolerance with only two being positive.

All of the Main Steam Safety Valve measurements are within the ± 3% of the setpoint. The single most positive out of tolerance condition discovered in Unit 1 was 2.54% during the third Unit 1 refueling outage. Similar to that of the Pressurizer Safety Valves, past safety analyses assumed a 3% accumulation added to the setpoint. This condition is bounded by the 3% accumulation assumption. The most severe set of four Main Steam Safety Valves being out of tolerance included only one that was positive (i.e. 1.68%), which is also bounded by the 3% accumulation assumption. For the case where 12 of 20 Main Steam Safety Valves were out of tolerance, it is again bounded by the 3% accumulation assumption due to the fact that only two Main Steam Safety Valves had a positive out of tolerance condition.

The Main Steam Safety Valves setpoints measured during past Unit 1 refueling outages all measured within the 3%. Although the Main Steam Safety Valves were discovered to be outside the allowable 1% according to Technical Specifications, the plant was operating within its safety analysis limits.

CORRECTIVE ACTIONS:

1. The three Unit 1 Pressurizer Safety Valves have been recently setpoint tested in accordance with the best technology available at this time. They should function to protect the Reactor Coolant System as designed until the next scheduled setpoint verification testing. No further action is warranted at this time.

2. Houston Lighting & Power has modified the test procedure to test the Pressurizer Safety Valve lift setpoint on saturated steam as recommended by WCAP-12910.

3. Houston Lighting & Power has been following industry attempts to resolve the Main Steam Safety Valve drift issue.

ADDITIONAL INFORMATION:

The Pressurizer Safety Valves are model number HB-BP-86 6N8 manufactured by Crosby. The Pressurizer Safety Valves are spring-loaded with an adjusting bolt nut which increases or decreases the spring force to achieve the relief valve opening.

The Main Steam Safety Valves are model 6-3707RAX-RT-25-XNC1012 manufactured by Consolidated/Dresser. The Main Steam Safety Valves are spring-loaded with an adjusting bolt which increases or decreases the spring force to achieve the relief valve opening.

The South Texas Project is pursuing a Technical Specification change to relax the lift setting tolerance of the Pressurizer Safety Valves from ±1% to ±2% and the Main Steam Safety Valves from ±1% to ±3%. The proposed setpoint tolerance of ±2% for the Pressurizer Safety Valves and ±3% for the Main Steam Safety Valves "as found" conditions was previously evaluated as part of the transition to VANTAGE 5H fuel.

Unit 2 Voluntary Licensee Event Report 93-011 was submitted documenting Pressurizer Safety Valve setpoints outside the required tolerance.

Table 1

Unit 1 Pressurizer Safety Valves Setpoint- 2485.00

	1RE01 08/05/89 AS FOUND/% ERR	1RE03 03/29/91 AS FOUND/% ERR	1RE04 09/19/92 AS FOUND/% ERR
PSV-3450	2381.70/-4.16	2523.00/+1.53	2591.00/+4.27
PSV-3451	2404.40/-3.24	2590.00/+4.23	2412.00/-2.94
PSV-3452	2447.40/-1.51	2542.00/+2.29	2318.00/-6.72

Table 2

Unit 1 Main Steam Safety Valves

TRAIN A	7410	7410A	7410B	7410C	7410D
TRAIN B	7420	7420A	7420B	7420C	7420D
TRAIN C	7430	7430A	7430B	7430C	7430D
TRAIN D	7440	7440A	7440B	7440C	7440D
SETPOINT	1285.0	1295.0	1305.0	1315.0	1325.0

1RE01 - 08/5/89

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	1251.75/-2.59	1282.70/-0.95	1301.50/-0.27	1306.10/-0.68	1316.00/-0.68
TRAIN B	1258.90/-2.03	1283.20/-0.91	1267.00/-2.91	1337.10/+1.68	1295.40/-2.23
TRAIN C	1289.00/+0.31	1270.10/-1.92	1293.40/-0.89	1344.70/+2.26	1314.10/-0.82
TRAIN D	1281.60/-0.26	1275.80/-1.48	1289.20/-1.21	1288.00/-2.05	1286.80/-2.88

1RE02 - 06/13/90

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	*	*	1313.93/+0.68	*	*
TRAIN B	*	*	*	*	*
TRAIN C	*	*	*	*	*
TRAIN D	*	1280.10/-1.15	1322.70/+1.36	*	*

* - NOT REQUIRED TO BE TESTED PER TECHNICAL SPECIFICATIONS

1RE03 - 03/29/91

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	1277.40/-0.59	1270.20/-1.92	1283.10/-1.68	1308.60/-0.49	1305.90/-1.44
TRAIN B	1269.10/-1.24	1274.00/-1.62	1278.40/-2.04	1291.40/-1.79	1321.90/-0.23
TRAIN C	1289.70/+0.37	1273.50/-1.66	1281.50/-1.80	1303.30/-0.89	1306.90/-1.37
TRAIN D	1275.90/-0.71	1303.30/+0.64	1338.10/+2.54	1312.90/-0.16	1345.20/+1.52

1RE04 - 09/19/92

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	1275.70/-0.72	1293.60/-0.11	*	1304.70/-0.78	*
TRAIN B	*	1293.50/-0.12	1302.70/-0.18	*	1324.00/-0.08
TRAIN C	*	*	*	*	*
TRAIN D	*	*	*	*	*

* - NOT REQUIRED TO BE TESTED PER TECHNICAL SPECIFICATIONS

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

February 14, 1995
ST-HL-AE-4991
File No.: G26
10CFR50.73

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

South Texas Project
Unit 2
Docket No. STN 50-499
Revision 1 to Voluntary Licensee Event Report 93-011
Pressurizer Safety Valve and
Main Steam Safety Valve Setpoints Outside Required Tolerance

Pursuant to 10CFR50.73, Houston Lighting & Power Company submits the attached revision to Unit 2 Voluntary Licensee Event Report 93-011, regarding Pressurizer Safety Valve and Main Steam Safety Valve setpoints outside required tolerance. This event did not have an adverse effect on the health and safety of the public.

The purpose of this revision is to incorporate information regarding several Main Steam Safety Valve and Pressurizer Safety Valve setpoints that were found to be out of tolerance during past outages. The out of tolerance conditions of the Main Steam and Pressurizer Safety Valves were evaluated for impact to the design basis analysis. The safety analysis concluded that all criteria are met, thus, the past safety analyses bounded each situation. This revision includes an expanded safety analysis regarding out of tolerance conditions and their impact on Unit operation. Changes are indicated by revision bars.

This Licensee Event Report was originally submitted prior to the issuance of the guidance which was provided by letter from Mr. S. J. Collins, Nuclear Regulatory Commission Division Director of Reactor Safety, to Mr. W. T. Cottle, Nuclear Group Vice President, Houston Lighting & Power, dated December 8, 1993. This guidance clarified interpretation of reporting requirements related to out of tolerance conditions on primary or secondary safety relief valves. Prior to the issuance of this guidance, Houston Lighting & Power had a different interpretation of the application of this reporting requirement.

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The original Licensee Event Report was focused on Pressurizer Safety Valves and did not include identification of similar conditions on the Main Steam Safety Valves or prior occurrences of these conditions. Additionally, the original Licensee Event Report did not provide an evaluation of the impact of the out of tolerance condition on the design basis analysis. After analysis of the out of tolerance data, it was determined that the out of tolerance conditions experienced as a result of the Pressurizer Safety and the Main Steam Safety Valves did not result in a condition that was unanalyzed or outside the design basis of the plant.

Houston Lighting & Power will ensure that this guidance is appropriately applied to out of tolerance conditions for surveillances in future outages. Personnel have been sensitized to the analysis and reporting requirements for these type of conditions.

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.

T. H. Cloninger
Vice President,
Nuclear Engineering

JMP/lf

Attachment: Revision 1 to Voluntary LER 93-011 (South Texas, Unit 2)

c:

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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 2	05000 499	1 OF 9

TITLE (4) Pressurizer Safety Valve and Main Steam Safety Valve Setpoints Outside Required Tolerance

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
04	01	93	93	-- 011 --	01	02	14	95		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N/A	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	000	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X OTHER
		20.405(a)(1)(iii)	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)	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)	Voluntary

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jairo Pinzon - Staff 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 April 1, 1993, Unit 2 was defueled during a refueling outage. After being reset at the second refueling outage for Unit 2 (2RE02) to the specified value of 2485 psig ± 1.0%, the setpoint for the Unit 2 Pressurizer Safety Valves were found to range from 3.8% below to 2.1% above the required setpoint during the third refueling outage (2RE03). This is a deviation from the ± 1.0% Technical Specification requirement. The Unit 2 Pressurizer Safety Valves (PSV-3450, 3451 & 3452) had been sent to Wyle Lab[illegible] for setpoint verification testing. Pressurizer Safety Valve setpoint drift is an industry-wide [illegible]h has been known for some time. The Westinghouse Owners Group has addressed this generic issue in WCAP-12910, which makes specific recommendations relative to Pressurizer Safety Valve setpoint verification testing. Corrective actions include modifying the test procedure to test the Pressurizer Safety Valve lift setpoint on saturated steam as recommended by WCAP-12910.

In addition to the above, additional Pressurizer Safety Valve out of tolerance conditions were discovered which were not previously identified. Several Main Steam Safety Valve setpoints were also found to be outside the required tolerances.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
South Texas, Unit 2	05000 499	93	-- 0 11 --	01	2 OF 9

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

DESCRIPTION OF EVENT:

On April 1, 1993, Unit 2 was defueled during its third refueling outage. The three Unit 2 Pressurizer Safety Valves (PSV-3450, 3451 & 3452) were sent to Wyle Laboratories for setpoint verification testing. After being reset at the second refueling outage for Unit 2 (2RE02), the setpoints were determined during the third refueling outage (2RE03) to be in the range of 2390 psig to 2536 psig or from 3.8% below to 2.1% above the required setpoint. This is a deviation from the ± 1.0% Technical Specification requirement. Table 1 presents the setpoint data for Unit 2.

Pressurizer Safety Valves are required to provide American Society of Mechanical Engineers (ASME) Code over-pressure protection for the Reactor Coolant System during plant transients. The valves are designed and tested to open at a nominal set pressure of 2500 psia. During normal plant operating conditions, the Pressurizer Safety Valves must maintain their leak tightness as part of the reactor coolant pressure boundary. At the South Texas Project, the valves are installed on water-filled loop seals to provide maximum seat tightness which keeps hydrogen from the valve seats and keeps the valve cool.

The South Texas Project utilizes Crosby HB-BP-86 6N8 Pressurizer Safety Valves which are spring-loaded with an adjusting bolt which increases or decreases the spring pressure to achieve the relief valve opening. Once a safety valve is set and the adjusting bolt is torqued, the lift set pressure should change only if the valve experiences leakage or if the valve is actuated. Neither of these conditions had occurred with the South Texas Project valves.

WCAP-12910 summarizes industry data (EPRI Report NP-2770-LD) and the results of the Westinghouse Owners Group Program which evaluates the Pressurizer Safety Valve set pressure shift phenomena. The WCAP primarily addresses the pressure shift noted between the setting conditions and the operating conditions. Our set pressure differences or deviations are between consecutive set pressure verification tests with the same test configuration. WCAP-12910 does, however, present a significant number of data points which can provide an understanding of the set pressure test repeatability.

After being set on steam and then re-tested on water loop seal, WCAP-12910 statistically evaluates the data and concludes that (for the specific valves tested), the set pressure shift did not exceed ± 1.0%. This is based on a very accurate measurement (0.1 to 0.5 mil) of spindle movement and a pressure curve slope change. For the South Texas Project Pressurizer Safety Valves, there is fair agreement between "pops" in the test stand on a water loop seal for a series of tests. However, a significant difference is seen when the data are taken about 18 months apart. The South Texas Project 6N8 type PSV is not included in the WCAP summary but the 6M6 type safety valve is included, and it would be expected to respond in a similar manner.

DESCRIPTION OF EVENT: (Continued)

WCAP-12910 recommends that Pressurizer Safety Valve set points should be established on a saturated steam environment, regardless of the medium used during operation. With a steam setting, the spindle movement is sharper, more visible and more reproducible.

Earlier Pressurizer Safety Valve set pressure testing for the South Texas Project (up to and including 1RE03 and 2RE02) and the WCAP-12910 set pressure testing was conducted at the Westinghouse Western Service Center while the 2RE03 testing was conducted at the Wyle Laboratories. The same test procedure was used and no data bias can be seen in favor of either test facility.

On January 16, 1995, an issue was raised by a Nuclear Regulatory Commission Inspector regarding the safety analysis of operation of the Unit with the Pressurizer Safety Valves in the out of tolerance condition. The issue related to the fact that this Voluntary Licensee Event Report did not document the impact of the Pressurizer Safety Valve out of tolerance condition in the "Analysis of Event" section. In addition, it was discovered that there have been instances where a number of the Main Steam Safety Valves have been found to be out of tolerance during the past refueling outage surveillances. Table 2 documents the as found condition for the tested valves during each Unit 2 outage.

A reportability review was immediately initiated to evaluate the as found condition of the Pressurizer Safety Valves and the Main Steam Safety Valves. Discussions with the Nuclear Regulatory Commission, Region IV, Engineering Management, indicated that the reportability of these past events should hinge on whether or not the condition resulted in the plant being outside the design basis. The reportability review concluded that although these valves were found to be out of tolerance, these conditions were bounded by the design basis of the plant.

CAUSE OF EVENT:

The observed setpoint drift on the South Texas Project Pressurizer Safety Valves is apparently the result of the inherent valve characteristics (i.e. the massive spring assembly) and the limited capability to accurately determine the first point of valve spindle movement. Inherent valve characteristics include the differences between the test conditions and especially the effect of temperature in the valve seat, plug area and on the spring.

The cause of the Main Steam Safety Valve out of tolerance conditions is an industry known problem regarding setpoint drift on the large pressure release devices.

ANALYSIS OF EVENT:

Houston Lighting & Power recognizes the guidance provided by letter from Mr. S. J. Collins, Nuclear Regulatory Commission Division Director of Reactor Safety, to Mr. W. T. Cottle, Nuclear Group Vice President, Houston Lighting & Power, dated December 8, 1993. This guidance clarified interpretation of reporting requirements of out of tolerance conditions of primary or secondary safety relief valves. Prior to the issuance of this guidance, Houston Lighting & Power had a different interpretation of the application of this reporting requirement. After analysis of the out of tolerance data, it was determined that the out of tolerance conditions experienced as a result of the Pressurizer Safety and the Main Steam Safety Valves did not result in a condition that was unanalyzed or outside the design basis of the plant.

Pressurizer Safety Valves

The Pressurizer Safety Valves protect the integrity of the Reactor Coolant Systems from overpressurization. The integrity of the Reactor Coolant System is required to prevent the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere. The reactor vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the American Society of Mechanical Engineers (ASME) Code for Nuclear Power Plants. The ASME Code permits a maximum transient pressure of 110% (2735 psig) of design pressure. Technical Specification Bases 2.1.2 and 3/4.4.2 identify 2735 psig as the safety limit for the Reactor Coolant System.

An assessment of the safety consequences and implication of the Pressurizer Safety Valves out of tolerance conditions was performed. Table 1 shows the as found values and the percentage of out of tolerance of the Pressurizer Safety Valves for each loop. These measurements were taken at each of the Unit 2 refueling outages.

ANALYSIS OF EVENT: (Continued)

Table 1 identifies that for Unit 2 all the Pressurizer Safety Valves were out of tolerance during the second and third refueling outages. Of the four out of tolerance conditions, one was high (or positive) and the remaining three were low (or negative). Positive out of tolerance conditions may result in increasing the calculated peak Reactor Coolant System pressure results for accidents that take credit for the Pressurizer Safety Valve actuation. Negative out of tolerance conditions may result in decreasing the calculated minimum Departure From Nucleate Boiling Ratio for accidents that take credit for Pressurizer Power Operated Relief Valve actuation.

The accidents that positive out of tolerance conditions may impact are:

1) Uncontrolled Rod Cluster Control Assembly withdrawal from a subcritical condition and at power (Condition II event) (FSAR section 15.4.1)
2) Loss of external electrical load (Condition II event) (FSAR section 15.2.2)
3) Turbine trip (Condition II event) (FSAR section 15.2.3)
4) Locked rotor (Condition IV event) (FSAR section 15.3.3)
5) Rod Cluster Control Assembly ejection (Condition IV event) (FSAR section 15.4.8)

The only positive out of tolerance condition was 2.05% or 2536 psig, which was identified during the third Unit 2 refueling outage. The other two Pressurizer Safety Valves were shown to be -3.82 % and -1.49%, which offsets the effect of the one being high by 2.05%. Past safety analyses included a 3% accumulation in the setpoint. An evaluation was made that showed the past safety analyses bounded the as measured Pressurizer Safety Valve setpoint cases. Therefore, the past safety analysis bounds the most severe positive out of tolerance condition.

The accidents that the negative out of tolerance conditions may impact are:

1) Uncontrolled Rod Cluster Control Assembly withdrawal from a subcritical condition and at power (Condition II event) (FSAR section 15.4.1)
2) Complete loss of flow (Condition III event) (FSAR section 15.3.2)
3) Rod out of position (Condition II event) (FSAR 15.4.3)
4) Hypothetical steamline break (Condition IV event) (FSAR section 15.1.5)
5) Locked rotor (Condition IV event) (FSAR section 15.3.3)
6) Dropped Rod Cluster Control Assembly (Condition II event) (FSAR section 15.4.3)

All of the Pressurizer Safety Valve negative out of tolerance conditions are bounded by the Pressurizer Power Operated Relief Valve setpoint of 2235 psig. Therefore, the past safety analyses bounds the most severe negative out of tolerance condition.

ANALYSIS OF EVENT: (Continued)

Main Steam Safety Valves

The safety function of the Main Steam Safety Valves is to protect the Steam Generator and the piping upstream of the Main Steam Isolation Valves from overpressurization under all modes of operation. During a loss of load/turbine trip event, operability of the Main Steam Safety Valves ensure that the secondary system pressure will be limited to within 110% (1413.5 psig) of its design pressure of 1285 psig.

An assessment of the safety consequences and implication of the Main Steam Safety Valves out of tolerance conditions was performed. Table 2 shows the as found values and the percentage of out of tolerance of the Main Steam Safety Valves for each loop. These measurements were taken at each of the past refueling outages.

The primary concern addressed in this safety analysis is Main Steam System overpressurization. First, the single most positive out of tolerance condition of the Main Steam Safety Valves occurred during the third Unit 2 refueling outage which was declared inoperable at the time of discovery. The high flux setpoint was adjusted per the Technical Specification. Only one other Main Steam Safety Valve was measured to be out of tolerance high at the time. Prior to this discovery, the past safety analyses still bounded the Main Steam Safety Valves because only five of the 20 Main Steam Safety Valves are actually required to provide peak steam relief following the secondary side pressure limiting transient.

All of the Main Steam Safety Valve measurements are within the ± 3% of the setpoint, with the exception of the one in Unit 2 mentioned in the previous paragraph. The single most positive out of tolerance condition discovered in Unit 2 was 1.52% during the third Unit 2 refueling outage. Similar to that of the Pressurizer Safety Valves, past safety analysis assumed a 3% accumulation added to the setpoint. This condition is also bounded by the 3% accumulation assumption. Therefore, the past safety analysis bounded the Main Steam Safety Valves.

The Main Steam Safety Valves setpoints measured during past Unit 2 refueling outages all measured within the 3%. Although the Main Steam Safety Valves were discovered to be outside the allowable 1% according to Technical Specifications, the plant was operating within its safety analysis limits.

CORRECTIVE ACTIONS:

1. The three Unit 2 Pressurizer Safety Valves have been recently setpoint tested in accordance with the best technology available.

2. Houston Lighting & Power has modified the test procedure to test the Pressurizer Safety Valve lift setpoint on saturated steam as recommended by WCAP-12910.

3. Houston Lighting and Power has been following industry attempts to resolve the Main Steam Safety Valve drift issue.

ADDITIONAL INFORMATION:

The Pressurizer Safety Valves are model number HB-BP-86 6N8 manufactured by Crosby. The Pressurizer Safety Valves are spring-loaded with an adjusting bolt which increases or decreases the spring force to achieve the relief valve opening.

The Main Steam Safety Valves are model 6-3707RAX-RT-25-XNC1012 manufactured by Consolidated/Dresser. The Main Steam Safety Valves are spring-loaded with an adjusting bolt which increases or decreases the spring force to achieve the relief valve opening.

The South Texas Project is pursuing a Technical Specification change to relax the lift setting tolerance of the Pressurizer Safety Valves from ±1% to ±2% and the Main Steam Safety Valves from ±1% to ±3%. The proposed setpoint tolerance of ±2% for the Pressurizer Safety Valves and ±3% for the Main Steam Safety Valves "as found" conditions was previously evaluated as part of the transition to VANTAGE 5H fuel.

Unit 1 Voluntary Licensee Event Report 92-018 was previously submitted documenting Pressurizer Safety Valve setpoints outside the required tolerance. This Licensee Event Report documented both Unit 1 Pressurizer Safety Valve setpoints outside required tolerances discovered during previous outages.

Table 1

Unit 2 Pressurizer Safety Valves Setpoint- 2485.00

	2RE01 08/05/89 AS FOUND/% ERR	2RE02 03/29/91 AS FOUND/% ERR	2RE03 09/19/92 AS FOUND/% ERR
PSV-3450	2488.00/+0.12	*	2390.00/-3.82
PSV-3451	*	2392.00/-3.74	2536.00/+2.05
PSV-3452	*	2478.00/-0.28	2448.00/-1.49

* - NOT REQUIRED TO BE TESTED PER TECHNICAL SPECIFICATIONS

Table 2

Unit 2 Main Steam Safety Valves

TRAIN A	7410	7410A	7410B	7410C	7410D
TRAIN B	7420	7420A	7420B	7420C	7420D
TRAIN C	7430	7430A	7430B	7430C	7430D
TRAIN D	7440	7440A	7440B	7440C	7440D
SETPOINT	1285.00	1295.00	1305.00	1315.00	1325.00

2RE01 - 10/11/90

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	1255.10/-2.33	1296.90/+0.15	1298.30/-0.51	1309.10/-0.45	1293.30/-2.39
TRAIN B	1277.60/-0.58	1282.20/-0.99	1283.80/-1.62	1304.70/-0.78	1295.10/-2.26
TRAIN C	1289.40/+0.34	1284.30/-0.83	1281.20/-1.82	1289.50/-1.94	1310.70/-1.08
TRAIN D	1294.70/+0.75	1258.60/-2.81	1306.80/+0.14	1326.30/+0.86	1315.60/-0.71

2RE02 - 09/14/91

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	*	*	*	*	*
TRAIN B	*	*	*	*	*
TRAIN C	*	*	*	1321.30/+0.48	1320.40/-0.35
TRAIN D	1295.40/+0.81	1274.00/-1.62	*	*	*

* - NOT REQUIRED TO BE TESTED PER TECHNICAL SPECIFICATIONS

2RE03 - 02/28/93

	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR	AS FOUND/% ERR
TRAIN A	1271.50/-1.05	1283.70/-0.87	1303.60/-0.11	1301.60/-1.02	1334.50/+0.72
TRAIN B	1284.80/-0.02	1295.90/+0.07	1304.00/-0.08	1307.60/-0.56	1345.20/+1.52
TRAIN C	**1357.00/+5.60	1294.40/-0.05	1307.50/+0.19	1315.70/+0.05	1295.40/-2.23
TRAIN D	1281.20/-0.30	1287.30-0.59	1297.40/-0.58	1302.50/-0.95	1321.80/-0.24

** - VALVE 7430 DID NOT ACTUATE ON 2/28/93, AND WAS DECLARED INOPERABLE AND HI-FLUX SETPOINT ADJUSTED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.