The Light company Houston Lighting & Power P.O. Box 1700 Houston, Texas 77001 (713) 228-9211

October 31, 1985
ST-HL-AE-1442
File No.: G9.17

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Responses to DSER/FSAR Items
Question 440.54

Dear Mr. Knighton:

The attachment enclosed provide STP's response to Draft Safety Evaluation Report (DSER) or Final Safety Analysis Report (FSAR) items.

The item numbers listed below correspond to those assigned on STP's internal list of items for completion which includes open and confirmatory DSER items, STP FSAR open items and open NRC questions. This list was given to your Mr. N. Prasad Kadambi on October 8, 1985 by our Mr. M. E. Powell.

The attachment includes mark-ups of FSAR pages which will be incorporated in a future FSAR amendment unless otherwise noted below.

The items which are attached to this letter are:

Attachment	Item No.*	Subject
1	D 15.0-2 (Q440.54)	Review of STP AOO's and PA's for all modes Note: Small Break and Large Break LOCA are not addressed and will provided in an additional transmittal.

* Legend
D - DSER Open Item
F - FSAR Open Item
C - DSER Confirmatory Item
Q - FSAR Question Response Item

L1/DSER/aw

B001
1/1

If you should have any questions concerning this matter, please contact Mr. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

JSP/bl

Attachments: See above

L1/DSER/aw

cc:

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Robert D. Martin
Regional Administrator, Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

N. Prasad Kadambi, Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Claude E. Johnson
Senior Resident Inspector/STP
c/o U.S. Nuclear Regulatory
Commission
P.O. Box 910
Bay City, TX 77414

M.D. Schwarz, Jr., Esquire
Baker & Botts
One Shell Plaza
Houston, TX 77002

J.R. Newman, Esquire
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

Director, Office of Inspection
and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

E.R. Brooks/R.L. Range
Central Power & Light Company
P.O. Box 2121
Corpus Christi, TX 78403

H.L. Peterson/G. Pokorny
City of Austin
P.O. Box 1088
Austin, TX 78767

J.B. Poston/A. vonRosenberg
City Public Service Board
P.O. Box 1771
San Antonio, TX 78296

Brian E. Berwick, Esquire
Assistant Attorney General for
the State of Texas
P.O. Box 12548, Capitol Station
Austin, TX 78711

Lanny A. Sinkin
3022 Porter Street, N.W. #304
Washington, DC 20008

Oreste R. Pirfo, Esquire
Hearing Attorney
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Charles Bechhoefer, Esquire
Chairman, Atomic Safety &
Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. James C. Lamb, III
313 Woodhaven Road
Chapel Hill, NC 27514

Judge Frederick J. Shon
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Ray Goldstein, Esquire
1001 Vaughn Building
807 Brazos
Austin, TX 78701

Citizens for Equitable Utilities, Inc.
c/o Ms. Peggy Buchorn
Route 1, Box 1684
Brazoria, TX 77422

Docketing & Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555
(3 Copies)

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
1717 H Street
Washington, DC 20555

Revised 9/25/85

L1/DSER/aw

ATTACHMENT 1
ST-HL-AE- 1442
PAGE 1 OF 11

Question 440.54N

State whether the STP AOO and PA analyses were performed for all operational modes. If not, or the assumption is made that Mode 1 bounds all the others, please review each AOO and PA to provide assurance that all equipment and systems relied upon for AOO or PA mitigation whose availability and operability is assured by the STP Technical Specifications in Modes 1 and 2 can also be relied on to provide mitigation in other modes. If this assurance can not be provided, then provide a detailed accounting of what systems, equipment, and protective functions were assumed for these modes, a justification of why the Modes 1 and 2 analyses are bounding, and a confirmation from the applicant that the technical specifications applicable in Modes 3, 4, and 5 will be consistent with and provide the same level of intended protection as the technical specifications in Modes 1 and 2. If differences exist between the Modes 1 and 2 analyses and those for other modes, these should be discussed in detail.

Response

A review of all STP anticipated operational occurrence (AOO) and postulated accident (PA) analyses for all modes and a discussion of the bounding analysis will be completed. A detailed confirmation that the Technical Specifications applicable in Modes 3, 4, and 5 are consistent with Modes 1 and 2 analyses will be done. The results will be available in the third quarter of 1985.

SEE Attached

Response:

Most AOOs and PAs are analyzed for occurrences in in Modes 1 and 2, since these are usually the modes in which the most severe consequences could possibly result. Since the Technical Specifications are based upon these analyses, and are written to ensure the availability of required protection logic and equipment in Modes 1 and 2, this response will concentrate on transients which are postulated to occur while the plant is in any of the subcritical operational modes (Modes 3, 4, and 5).

Generally, the occurrence of an AOO or PA when the plant is in a subcritical mode will not result in consequences more severe than those which would result in Modes 1 and 2. This is due mainly to the reduced temperature and pressure conditions characteristic of subcritical modes. In some cases, certain AOOs or PAs cannot occur, or cannot produce a significant transient (i.e., a transient which would challenge plant safety limits), and therefore, protection is not always required to the same degree as in Modes 1 and 2.

Each AOO and PA has been reviewed with attention to occurrences in modes which are not identified in the FSAR, and to the protection operability requirements in these modes. This review included consideration of the applicable Technical Specification to assure that the protection and operability required by the analysis was assured by the Technical Specifications. The results and conclusions of this review are given below.

Feedwater System Malfunction

This AOO increases the core heat removal rate, which reduces the core temperature, leading to an increase in power generation (due to the negative moderator temperature coefficient) and a consequential reduction in thermal margin. The heat removal rate may be increased either by an increase in feedwater flow, or a decrease in feedwater temperature. Analyses or evaluations for both cases, in Modes 1 and 2, are presented in ~~the FSARR~~ chapter 15.

In Modes 1 and 2, protection is available from the power range high neutron flux trip. If the increased heat removal is due to abnormally high feedwater flow, then turbine trip and feedwater isolation will occur when a high steam generator water level setpoint is reached.

The feedwater malfunction, associated with a drop in feedwater temperature, is not a concern below Mode 2, because there is no pre-heating of feedwater in those modes.

The feedwater control malfunction which causes an increase in feedwater flow, postulated to result from the failing open of a feedwater control valve, is ill-defined below Mode 2, since the main feedwater system would probably be secured. Even if the main feedwater system were in operation in Mode 3, the flow would normally be controlled via the feedwater bypass valves, not the feedwater control valves, since the (smaller) bypass valves provide much better control under low flow conditions. Therefore, failure of a main feedwater control valve in Mode 3 is not likely. The assumption of a failed-open feedwater bypass valve, in Mode 3 and below, would result in a relatively slow transient due to the lower feedwater flow rate.

In subcritical modes, the increased heat removal cause rate would cause an increase in the audible count rate and possibly the extended range flux multiplication alarm to sound, alerting the operator to the increase in neutron flux. If no operator action is taken, then any withdrawn rods will be automatically inserted when the source range high neutron flux trip setpoint is reached.

In Modes 5 and 6, when the RCS is cold, any increase in heat removal rate would not be meaningful, nor would there be any viable feedback to the core, since the main heat removal path will be via the RHR system.

In general, the potential for serious consequences, resulting from cooldown events in the subcritical modes is low, since the RCS is relatively cool (usually less than the no-load temperature); and the core is shutdown.

Excessive Load Increase

The excessive load increase, in Mode 2, will not be as severe as either the Mode 1 excessive load increase, or the opening of a steam generator safety or relief valve (which is analyzed in Mode 2). This AOO is an increase in steam flow (load), usually 10 percent, which may or may not generate a reactor trip signal, depending upon the plant and protection system characteristics. Mode 1 analyses are presented in ~~the~~ chapter 15. ~~FSAR.~~ An excessive load increase in Mode 1 is considered limiting, since an excessive load increase at full power would put the plant at the highest achievable power level. Load increases at less than full power, or during startup (Mode 2), would not reach as high a power level before trip.

In Mode 3, the excessive load increase may be considered to be a simple steam release, since there can be no load, per se, when the turbine is off-line and the core is subcritical. The Mode 3 load increase would be less limiting than the Mode 1 or Mode 2 case, since the core is already subcritical. Automatic safety injection actuation may not be available, if it is blocked by the operator (blocking is necessary to depressurize). However, the RCS must be borated to the cold shutdown concentration prior to blocking SI, in order to prevent a return to criticality in the event of a cooldown.

The Mode 4 situation is bounded by Mode 3, since pressure and temperature conditions in the primary and secondary systems are reduced. Also, a cooldown in Mode 4 will not be aggaravated by the addition of auxiliary feedwater. At some point in Mode 4, the RHR system will be placed in service, disconnecting the steam generators from the heat removal path.

In Modes 5 and 6, the residual heat removal system should be in operation. Any steam release, if possible, would have little or no effect upon the core.

Spurious Opening of a Steam Generator Safety or Relief Valve

The Condition II steam line break, or the spurious opening of a steam generator safety or relief valve, also affects the core like a load increase; but the analysis assumptions that are applied are different. The Condition II steam line break is usually assumed to be an unisolatable, uncontrolled steam release which causes a non-uniform core cooldown (typical of an open safety valve) during the period immediately following a reactor trip which inserts all but the most reactive RCCA. The resulting reactivity excursion may be large enough to overcome the shutdown margin and return the core to critical, especially when there is little or no decay heat (with power peaking in the region of the stuck RCCA). The Condition II steam line break is analyzed in Mode 2, and the assumptions used lead to a more severe transient than would result from a load increase in Mode 1.

or other reactor trip signals.

In Mode 1, prior to reactor trip, the transient characteristics of this AOO are similar to the excessive load increase. A reactor trip signal, if needed, will result from the overpower delta-T logic. After the reactor trip, the concern becomes a possible return to criticality with the most reactive RCCA stuck in the fully withdrawn position, leading to high local power levels. However, a post-trip return to criticality is less likely when this AOO occurs in Mode 1 than in Mode 2, because there will be more decay heat present, which tends to retard the cooldown. Mode 1 steamline break is discussed in WCAP-9226.

In Mode 3, results are expected to be better than the Mode 2 case, since pressure, temperature and flow conditions would be less limiting. An occurrence in Mode 4 would be even less severe than in Modes 2 or 3, due to the lower initial RCS temperature, and an effective decoupling of the secondary system from the primary system as the reactor coolant pumps are removed from service and the residual heat removal system is started. Automatic SI actuation is available through Mode 3, until the RCS is borated and the SI is blocked (see excessive load increase discussion). One high-head SI pump must be operable in Mode 4, available for activation by the operator, if needed.

Any cooldown in Modes 5 and 6 is meaningless, since the RCS is already cold, and the RHR system effectively decouples the steam generators from the core.

Steam Line Rupture

The steam line rupture is a Condition IV event, producing a greater uncontrolled steam release than the spurious opening of a steam generator safety or relief valve (above); but the relative effects in the various modes, and requirements for protection equipment are the same. This is the most severe cooldown event.

Steam Flow Reduction

In the case of South Texas, there are no steam pressure regulators whose malfunction or failure could cause a steam flow transient. No safety analyses or protection, in any mode, are required.

Loss of Electrical Load

This AOO can occur only in Mode 1, since the turbine is off-line in all other modes. Loss of electrical load is bounded by the turbine trip (below), which is analyzed and reported in ~~the FSAR~~ Chapter 15 for Mode 1.

Turbine Trip

This AOO is defined only in Mode 1, since the turbine would be off-line below Mode 1, and bounds the loss of electrical load (above), since steam flow is terminated more rapidly by a turbine trip than by a loss of load.

Spurious MSIV Closure

The Mode 1 case is limiting, which itself is bounded by the turbine trip AOO (above). In Modes 2, 3, and 4, the plant may be cooling down via steam dumping to the condenser. MSIV closure in these modes may prevent the use of the condenser, and require atmospheric steam dumping. There is no steam flow below Mode 4, since cooldown is continued via the residual heat removal system. Only the availability of a means to dump steam to the atmosphere (including PORVs and safety valves) is required, for decay heat removal, in the event that the MSIVs close while the plant is in any mode above Mode 5.

Loss of Condenser Vacuum

The full power case is bounded by the turbine trip AOO. Loss of the condenser vacuum, while the plant is below Mode 1, may require decay heat removal via atmospheric steam dumping, until some time in Mode 4, when the residual heat removal system is placed in operation.

Loss of AC Power

The loss of AC power results in the loss of primary coolant flow and main feedwater flow. It must be shown that decay heat can be removed, via natural circulation in the reactor coolant system, to the steam generators, which are supplied with auxiliary feedwater. Therefore, the full power case (maximum decay heat) is limiting. At least one auxiliary feedwater pump is required (and is available), in Modes 1 through 3, for decay heat removal.

In Mode 4, the transition is made from steam dumping to the residual heat removal system for further cooldown. Although the auxiliary feedwater pumps are not required to be available in this mode, it is reasonable to assume that, during cooldown operations, the reactor operator would continue to feed the steam generators with auxiliary feedwater, well into Mode 4, until the RCS pressure decreases to a level low enough to activate the Residual Heat Removal System.

Loss of Feedwater

This AOO results in a heatup and pressurization of the RCS. Therefore, an occurrence at full power would result in the most severe consequences.

In Modes 2 and 3, the auxiliary feedwater pumps are available ~~(and~~ t______ for startup and decay heat removal. Not all the auxiliary feedwater pumps are required for decay heat removal, and loss of all auxiliary feedwater pumps is not likely. If none of the auxiliary feedwater pumps are operable, then the tech specs require the operator to restore at least one pump as soon as possible. spell out

At some time in Mode 4, the Residual Heat Removal System will be placed in service, and cooldown via the steam generators will not be necessary. Prior to the operation of the RHR System, the operator is assumed to be using the start up feedwater pumps or auxiliary feedwater pumps, even though they are not required to be available in Mode 4 (see Loss of AC Power).

Feedwater Line Rupture

This PA may occur any time the steam generator is pressurized. An occurrence in Mode 1 would cause the greatest RCS heatup and pressurization. Therefore, the Mode 1 case is analyzed, and bounds events in Modes 2, 3, and 4.

Auxiliary feedwater is required through Mode 3 for decay heat removal. In Mode 4, the low levels of decay heat and primary and secondary side temperature and pressure will result in a relatively minor, slow transient.

Below Mode 4, the question becomes moot, since the steam generators are no longer required for decay heat removal.

Partial Loss of Flow

The loss of a reactor coolant pump reduces the heat removal rate from the primary to the secondary coolant system, thereby causing a heatup in the RCS. An occurrence at full power would produce a greater heatup than would an occurrence at no-load (Mode 2). Below Mode 2, when the core is subcritical, it is common to have one or more reactor coolant pumps out of service, since full flow is no longer required. Loss of a reactor coolant pump below Mode 2, even if it is the only pump in service, would still be bounded by either the partial loss of flow in Mode 1, or the complete loss of flow in Mode 1 (below).

Loss of Flow

As in the partial loss of flow, the most severe case is an occurrence in Mode 1. However, the loss of all reactor coolant pumps means that the only mechanism available for decay heat removal from the core is via natural circulation. Therefore, adequate natural circulation and auxiliary feedwater are required through Mode 3. Auxiliary feedwater, although not required by tech specs, is assumed to be available until the residual heat removal system can be placed in service (Mode 4). This event, and its protection requirements are similar to the Loss of AC Power event (discussed previously). spell out

Locked Rotor and Reactor Coolant Pump Shaft Break

These PAs are similar to the partial loss of flow (above) as far as the limiting modes and required protection equipment are concerned.

RCCA Withdrawal from Subcritical

Chapter 15
~~The FSAR~~ presents an analysis for this AOO in Mode 2. An occurrence in Mode 3, 4, or 5, with two or more reactor coolant pumps in operation, would be bounded by the analysis in Mode 2. This is based upon the ~~FSAR~~ analysis assumption that reactor trip does not occur until the power-range (low setting) high neutron flux setpoint is reached, and that two banks are withdrawn sequentially at maximum speed (72 step/min). These conservative assumptions result in the core returning to critical and generating some power prior to trip. Therefore, the primary system flow rate becomes an important consideration, as a factor in DNB evaluation. (Note that, in Mode 3, the tech specs require two reactor coolant pumps to be in operation whenever the reactor trip breakers are closed). spell out

However, in Modes 3, 4, and 5, the source range high neutron flux trip will be available to terminate the event, by tripping any withdrawn and withdrawing rods, before any significant power level could be attained. Therefore, DNB and primary system flow rate need not be considered. Also, the reactivity insertion rate would be slower, when in any of the subcritical modes, since a single failure in the rod control system could cause the withdrawal of only one bank, and its withdrawal rate would be expected to be slower than the maximum rod speed which is possible when in automatic rod control (and is assumed in the ~~FSAR~~ Chapter 15 analysis).

RCCA Withdrawal at Power

This AOO is defined only in Mode 1.

Dropped RCCA Bank

Since the dropping of an RCCA bank will perturb the core only if there is some significant neutron flux level, this event is analyzed only in Mode 1. A less severe case can be postulated at the low power level of Mode 2. Dropping an RCCA bank while in any of the subcritical modes, if any are withdrawn, would have no effect (i.e., no DNB concern).

Dropped RCCAs

See dropped RCCA bank (above).

Single Rod Withdrawal

The limiting case is an occurrence while in Mode 1. An occurrence in any of the subcritical modes would have no effect. If the shutdown margin requirements are satisfied, then no single rod withdrawal would insert enough reactivity to attain criticality, since the shutdown margin requirements are determined assuming the most reactive RCCA is fully withdrawn.

Static Rod Misalignment

As in the dropped RCCAs and dropped RCCA bank, this event would have no effect in the absence of a critical neutron flux. The limiting case, and analysis, is for Mode 1, which bounds Mode 2. There is no DNB concern in any of the subcritical modes.

Startup of an Inactive Loop

For plants without loop isolation valves (e.g. South Texas), the consequences of this event are directly related to the temperature difference between the inactive loop vessel inlet and the core. Relatively cold water would enter the core, after the reactor coolant pump in the inactive loop is started up, and cause a reactivity excursion. Therefore, the most severe consequences are incurred when the plant is operating at the maximum permissable power level with a loop out of service. This is the Mode 1 case which appears in the FSAR. The Mode 2 case, when starting up, is bounded. Startup of an inactive

loop while in any of the subcritical modes would have relatively little effect upon the core temperature, since there would be little or no temperature difference between active and inactive loops. Below Mode 4, the RHR System would be in operation.

Boron Dilution

ATTACHMENT 1
ST-HL-AE-1442
PAGE 10 OF 11

This AOO is analyzed or evaluated in every mode.

Fuel Assembly Misloading

This event, like the rod misalignment events, is meaningful only in the presence of a critical neutron flux. Mode 1 behavior is presented in ~~the FSAR,~~ Chapter 15 which bounds the Mode 2 startup case.

RCCA Ejection

Mode 1 and 2 cases are analyzed ~~for the FSAR~~ in Chapter 15. If shutdown margin requirements are met, then the ejection of a rod while in Modes 3 and 4 would not insert sufficient reactivity to attain criticality, since the shutdown margin requirements are determined assuming that the most reactive RCCA is fully withdrawn. In Mode 5, ejection is impossible, since the RCS is depressurized.

Accidental ECCS Actuation

In Mode 1, this event is analyzed for its effect upon the core. A spurious SI signal should cause an immediate reactor trip. Delivery of safety injection fluid to the core would also cause shutdown. However, if the SI signal does not generate a reactor trip signal, then there would be no effect, since the South Texas HHSI pumps do not have sufficient head (rated at only 1600 psi) to inject into the RCS at normal operating pressure.

At pressures below the normal operating pressure, in Mode 3, the SI system has the potential to pressurize the RCS to the shutoff head of the HHSI pumps (1600 psia). When RCS temperature drops to the level necessary to arm COMS (Mode 4)), then only one high-head SI pump is permitted to be available, and the RCS may become pressurized to the pressurizer relief valve setpoint, which is set to provide cold overpressure protection. In Mode 4 spurious SI actuation is not likely, since most automatic SI signals are blocked.

CVCS Malfunction

The boron dilution aspects of this event are covered in the boron dilution AOO (above). As a mass addition transient, this event is addressed by the cold overpressure tech specs (Modes 4 and 5).

Spurious Opening of a Pressurizer Relief or Safety Valve

When analyzed as a depressurization event, the concern becomes a possible violation of the minimum DNBR criterion. Therefore, this AOO is analyzed in Mode 1, which also bounds Mode 2. DNB is not a realistic concern in any of the subcritical modes. In Mode 5, this AOO is inconsequential, since the RCS is already depressurized.

The loss of RCS inventory aspects of this AOO are considered as part of the Small Break Loss of Coolant Accident (below).

Small Break Loss of Coolant Accident

LATER

Steam Generator Tube Rupture

The tube rupture question is deferred, pending the findings of the Westinghouse Owners Group study in progress. Results are expected around the end of this year.

Large Break Loss of Coolant

LATER

CONCLUSION

Therefore it is concluded that appropriate protection for all applicable AOOs and PAs (except where noted), is available and assured by Technical Specifications for all operable modes. The reduction in the operability requirements of Technical Specifications below Modes 1 and 2 are consistent with the reduction in the severity and potential consequences of each AOO and PA in these lower modes.

Many components and systems, in which transient-initiating failures are postulated to occur when in Modes 1 and 2, are removed from service as the RCS temperature and pressure is reduced below Mode 2. This permits the disarming of protection systems when they are no longer required. Nevertheless, it is our conclusion that adequate protection is assured, and would be available by automatic actuation or operator action at a level that is consistent with the protection available in Modes 1 and 2, considering the reduction in the protection requirements below Mode 2.