952

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

RELATED CORRESPONDENCE

DOCKETED USNRC

'86 OCT -3 P4:07

OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

September 30, 1986
ST-HL-AE-1767
File No.: G9.18

Mr. Vincent S. Noonan, Project Director
PWR Project Directorate #5
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499-OL
SER Open Item #16 and
Confirmatory Item #13; Long Term Cooling

Reference: Letter ST-HL-AE-1724 dated August 20, 1986; M. R. Wisenburg to V. S. Noonan

Dear Mr. Noonan:

This letter transmits information with respect to South Texas Project Safety Evaluation Report (SER) open item #16 and confirmatory item #13. These items involve the use of the Westinghouse (W) TREAT computer program to evaluate STP's long term cooling capabilities in the event of a small break loss-of-coolant accident (LOCA), a non-isolable LOCA and a secondary break.

The analysis has been completed and demonstrates that the STP emergency core cooling system (ECCS) meets the requirements of 10CFR50.46. The analysis, which was discussed with the NRC staff in August 1986 (meeting minutes were provided in the reference), is provided in Enclosure 1. Enclosure 2 includes associated revisions to the FSAR which will be incorporated in a future FSAR amendment.

Also provided is a comparison of the TREAT and NOTRUMP Westinghouse programs which was used to benchmark TREAT against applicable 10CFR50.46 criteria. Enclosure 3 contains the following:

1. 5 copies of WCAP-11232, "Comparison of the TREAT and NOTRUMP Small Break LOCA Transient Results." (Proprietary)

2. 5 copies of WCAP-11297, "Comparison of the TREAT and NOTRUMP Small Break LOCA Transient Results." (Non-proprietary)

Also enclosed is a Westinghouse authorization letter (CAW-86-084), Proprietary Information Notice, and accompanying affidavit.

L1/NRC/na

8610060980 860930
PDR ADOCK 05000498
E PDR

DS03

As item (1) of Enclosure 3 contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations. Correspondence with respect to the proprietary aspects of the Application for Withholding or the supporting Westinghouse affidavit should reference CAW-86-084 and should be addressed to R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Based on the information provided in the enclosures Houston Lighting & Power Company considers the aforementioned items to be "closed". If you should have any questions on this matter, please contact Mr. J. S. Phelps at (713) 993-1367.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

JSP/yd

Enclosures 1: Long Term Cooling Analysis for South Texas Project

2: FSAR revisions

3: Comparison of the TREAT and NOTRUMP Small Break Transient Results
Five copies each of WCAP-11232 (Proprietary) and WCAP-11297 (Non-proprietary)

L1/NRC/na

*cc:

Hugh L. Thompson, Jr., Director
Division of PWR Licensing - A
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

T.V. Shockley/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
Christic Institute
1324 North Capitol Street
Washington, D.C. 20002

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

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

*Enclosures 1 & 2 Only

Enclosure 2

monitoring. To accommodate this additional RCS inventory, letdown may be discharged from the reactor vessel head vent line with letdown routed to the pressurizer relief tank. The head vent throttling valves are controlled
through the Qualified Display Processing System (QDPS), described in Section 53
7.5.6.

The safety-related cooldown is accomplished by increasing the steam release from the SG PORVs to attain a rate of primary side cooling of approximately
25°F/hr. The SG PORVs are also controlled through the QDPS. In conjunction 53
with this portion of the cooldown, the charging pumps are used to deliver water to make up for primary system contraction due to cooling. Makeup is also required for inventory control in the event the reactor vessel head vent path is periodically opened to provide head cooling. ~~An alternative to head cooling is to provide an 8 hour soak period following RCS cooldown and prior
to RCS depressurization. The safety grade AFST has adequate capacity to accommodate the identified soak period~~. 54
Upon approaching the end of this phase of cooldown (RCS temperature of approximately 350°F), the RCS is depressurized to approximately 350 psig by venting the pressurizer through the safety-related pressurizer PORVs.

To ensure that the accumulators do not repressurize the RCS, the accumulator discharge valves are closed prior to the RCS pressure dropping below the accumulator discharge pressure. Each accumulator is provided with a Class 1E solenoid-actuated valve to ensure that the accumulator may be vented through the nitrogen supply header should the accumulator discharge isolation valve fail. A branch line inside the Containment with a parallel set of Class 1E
valves allows venting the nitrogen header to Containment atmosphere. 38

Actuation of the SIS is precluded by use of the pressurizer low pressure and excessive cooldown signal blocks.

When the reactor coolant temperature and pressure are reduced to approximately 350° and 350 psig, respectively, the second phase of cooldown starts with the
RHRS being placed in operation. Since loss of the non-safety-grade instrument 53
air system results in a loss of the air supply to the flow control valves that are normally used to limit the initial RHRS cooldown rate, the operator may choose to use only one of the RHR subsystems as a means to control cooldown rate. Should a single failure occur, such as that of an RHRS component, precluding operation of one of the RHR subsystems, the operator could elect to use a fully operational RHR train. Cooldown would continue using the fully operational RHR train(s), until the failed equipment or component could be made available. A failure mode and effects analysis for cold shutdown operations is provided in Table 5.4.A-2.

Cooldown of the RCS is continued using available RHR trains and following cooldown rate limits. The time required to reach the cold shutdown conditions (see definition in Technical Specifications) depends upon the number of RHR trains available, and the CCW and ECW temperatures.

TABLE 5.4.A-1 (Continued)

COMPLIANCE COMPARISON WITH BRANCH TECHNICAL POSITION RSB 5-1

| Design Requirements of BTP RSB 5-1 | Process and [System or Component] | Possible Solution for Full Compliance | Recommended Implementation for Class 2 plants* | Degree of STP Compliance** |
|---|---|---|---|---|
| V. Test requirement<br>Meet RG 1.68 for PWRs, test plus analysis for cooldown under natural circulation to confirm adequate mixing and cooldown within limits specified in Emergency Operating Procedures. | | Run tests and confirming analysis to meet requirement. | Compliance required. | Meets the intent of RG 1.68. Test data and analysis for a plant similar in design to STP will verify adequate mixing and cooldown under natural circulation conditions (Section 14.2). |
| VI. Operational procedure<br>Meet RG 1.33. For PWRs, include specific procedures and information for cooldown under natural circulation. | | Develop procedures and information from tests and analysis. | Compliance required. | Generic Procedures as developed by the Westinghouse Owners Group will be used as the basis for plant specific procedures. 38 |
| VII. Auxiliary Feedwater Supply<br>Seismic Category I supply for auxiliary feedwater for at least four hours at hot ~~standby~~ stet plus cooldown to residual heat removal cut-in temperature followed by an 8 hour soak period based on longest time for only onsite or only offsite power and assumed single failure. | Emergency feedwater supply | From tests and analysis obtain conservative estimate of auxiliary feedwater supply to meet requirments and provide Seismic Category I supply. | Compliance will not be required if it is shown that an adequate alternate Seismic Category I source is available. | The AFST usable capacity of 525,000 gals is adequate to support 4 hrs at hot standby and a 10 hr cooldown to RHR cut-in conditions followed by an 8-hour soak period prior to actual RHR initiation with a margin for contingencies. The AFST meets Seismic Category I requirements (Section 10.4.7) 54 |

NOTES:

* The implementation for Class 2 plants does not result in a major impact while providing additional capability to go to cold shutdown. The major impact results from the requirement for safety-related steam dump valves.

** STP falls within the category of Class 2 plant as defined by Section H, "Implementation," of Branch Technical Position RSB 5-1, Revision 2.





ATTACHMENT 2
ST-HL-AE-1767
PAGE 2 OF 12

ATTACHMENT 2
ST-HL-AE-1767
PAGE 3 OF 12



The AFWS is also designed for the following normal plant operations. 39

10.4.9.1.1 <u>Plant Cold Startup</u>: The AFWS is designed to back up the main FW system during plant startup in the event the main FW system and/or the startup SGFP is unavailable.

10.4.9.1.2 <u>Plant Hot Shutdown</u>: The AFWS is designed to back up the main FW system during plant hot shutdown (or hot standby) in the event the main FW system and/or the startup SGFP is unavailable. The AFWS can be used as a means of continuous FW supply even if this condition is maintained for extended periods. FW is continuously supplied from the AFST, which during normal operation receives required makeup from the demineralized water storage tank (DWST). The DWST in turn is supplied by water from wells through the demineralizers, as shown on Figures 9.2.3-1 and 9.2.6-1. 45

10.4.9.1.3 <u>Plant Cold Shutdown</u>: The AFWS is designed to back up the main FW system when achieving plant cold shutdown.

10.4.9.2 <u>System Description</u>. One AFWS is provided for each unit. The piping diagram is shown on Figure 10.4.9-1. The system includes an adequate water storage, redundant pumping capacity to supply the SGs, associated piping, valves, and instrumentation. 39

The AFWS supplies water to the SGs, where it is converted into steam by the heat transferred from the primary coolant that removes decay heat from the reactor core and heat generated in the primary coolant loop by the reactor coolant pumps. 39

The AFST provides water to the AFW pumps. It is a concrete, stainless steel lined, tank with a usable capacity of 525,000 gallons based on the following plus a margin for contingencies; 5-

- maintaining the plant in hot standby for four hours, then 31
- cooling down the primary system to 350°F. delete
- ~~followed by an eight hour soak period, after which the residual heat removal system may be initiated.~~ 54

The cooldown rate is 50°F/hr with one RCP operating or 25°F/hr with natural circulation. During normal cooldown the rate is limited to 100°F/hr due to structural limits of the RCS components. 39

Four AFW pumps, each with independent motive power supplies, are provided to comply with redundancy requirements of the safety standards, both for equipment and power supplies. Pump characteristics are given in Table 10.1-1. 46 39

Three horizontal, centrifugal, multistage, electric motor-driven pumps supply one SG each. Each pump motor is supplied power from a separate engineered safety bus, and the power supply is separated throughout.

7. Uncoupling of Human Errors

This study assumes that test and maintenance activities are staggered. That is, redundant AFWS components are not tested by the same personnel on the same shift, but in general, tests and/or maintenance of redundant components involve time and/or personnel changes (e.g., different personnel and shifts, or the same personnel on a different day, etc.) In addition, a double-check procedure is assumed to assure the correct status of locked open valves after test and maintenance. This significantly reduces the probability of human error in two or more trains simultaneously. Given that test and maintenance activities are staggered and the use of a double check-procedure, it is reasonable to assume that human errors for test and maintenance are uncoupled.

For the above reasons, the evaluation does not consider concurrent disabling of multiple trains because of human error in conjunction with test or maintenance to be a credible failure scenario.

8. Technical Specification

The auxiliary feedwater system design is evaluated in accordance with the STP Technical Specifications (Ref. 7).

| | |
|---|---|
| Train A - | Availability is assumed to be degraded since there is no Technical Specification requirement on Train A. |
| Trains B, C, and D | - Operable except for the scenarios illustrated in the fault trees in Section 10A.3.2. |

54

9. HVAC Support

The motor driven auxiliary feedwater pump rooms are cooled by safety-related HVAC units powered by their respective trains. The turbine driven pump room is cooled by a Train A HVAC unit, however, the turbine driven pump is qualified for operation following the loss of all HVAC. Consistent with NUREG-0611 methodology, HVAC support to the pumps is not considered in this evaluation.

10. Auxiliary Feedwater Storage Tank

The AFWST capacity is sufficient to allow the RCS to remain at hot standby for 4 hours followed by a 10 hour cooldown, ~~and an 8-6 hour soak period~~ at which point further RCS cooldown is performed by the residual heat removal system. If additional quantities are needed, water can be provided to the AFWST from the demineralized water storage tank, the condenser hot well, or an alternate onsite source. The AFWST has level instrumentation with control room indication and annunciation to warn operators of low AFWST water inventory.

tank, with a usable volume of 525,000 gallons. The



4. Auxiliary Feedwater Storage Tank

The Seismic Category I auxiliary feedwater storage tank provides water to the AFW pumps. It is a concrete, stainless steel lined, ~~518,000 gallon~~ tank ~~which~~ has sufficient capacity to allow the RCS to remain at hot standby for 4 hours followed by a 10 hour cooldown ~~and an 8 hour soak period~~ at which point further RCS cooldown is performed by the residual heat removal system.

The AFWST is designed to withstand environmental design conditions, including floods, earthquakes, hurricanes, tornado loadings, and tornado missiles. The AFWST is designed so that no single active failure will preclude the ability to provide water to the AFW system. Each train has a dedicated suction line from the AFWST to the AFW pumps. The water level in the AFWST is indicated in the control room as well as at the auxiliary shutdown panel. A low level alarm is also provided in the control room.

10A.2.3 Emergency Operation

The AFWS is designed for automatic actuation in an emergency. Any of the following conditions automatically starts the three Class 1E motor-driven pumps: 54

1. Two out of four channels showing low-low water level in any steam generator

2. Safety injection signal

3. 4.16 kV bus undervoltage. The AFW pump is started in conjunction with diesel generator starting and load sequencing. Water is not automatically fed to the steam generator until condition 1 or 2 above exists.

The turbine-driven auxiliary feedwater pump starts automatically on any of the following signals:

1. Two out of four channels showing low-low water level in any steam generator

2. Safety injection signal

A one-inch bypass line with a normally closed solenoid operated valve (FV0143) and orifice is provided around the steam inlet valve (MS0143). This bypass valve (FV0143) opens upon receipt of either of the above signals to supply steam to the turbine and allow the turbine to reach governor control speed. After a time delay to allow governor control speed to be reached, the steam inlet valve is opened which allows rated steam flow to the turbine. This arrangement precludes an overspeed trip due to excessive steam flow prior to governor warmup. This bypass line is not dependent upon AC power to operate.

ATTACHMENT 2
ST-HL-AE-1767
PAGE 6 OF 12

Question 440.30N

With regard to the information in Appendix 5.4A "Cold Shutdown Capability" identify the most limiting single failure with regard to cooldown capability and verify that the statement of Table 5.4A-1 that the auxiliary feedwater storage tank (AFST) "capacity of 500,000 gallons is adequate to support 4 hours at hot standby conditions followed by 10 hours cooldown to RHR cut in condition with a margin for contingencies" considers this failure.

Response

The most limiting failure regarding cooldown time is the loss of "A" train AC power, which results in the loss of two steam generator PORVs. RHR cut-in conditions can be achieved with this failure ~~22~~ 14 hours after reactor trip based on maintaining hot standby for four hours followed by a ten hour natural circulation cooldown. ~~and then an eight hour soak period.~~ Approximately 360 ~~445~~,000 gallons of water would be added to the effective steam generators during this period. 54

Specifically the AFST sizing considers: 4 hours at hot standby and a 10 ~~8~~ hour natural circulation cooldown. ~~8 hour soak period.~~ It also considers possible level instrument error, water delivered to a faulted steam generator, water lost through the turbine lube oil cooler, various small system water losses (ie., flange or pump seal leakage), a soak period (up to 8 hours, if necessary), and a margin against vortex formation. The usable volume in the AFST above the suction nozzles is 525,000 gallons.

Question 440.38N

a. Demonstrate that the STP ECCS meets 10 CFR Part 50.46 criteria for long term decay heat removal in the event of a small break LOCA of a size such that recirculation would be required but the RCS pressure either remains above the low-head safety injection (LHSI) pump shutoff head or recovers after loss of the secondary heat sink. An examination of Figures 6.3-1 through 6.3-5 does not indicate that the STP ECCS is designed for high-head recirculation combined with decay heat removal by the RHR heat exchangers, i.e., there are no apparent provisions for routing recirculation flow from the RHR heat exchangers to the HHSI pumps. Also, as described in Appendix 5.4.A "Cold Shutdown Capability," the steam generators have a limited supply of safety grade secondary water supply, since there is not a safety grade backup to the auxiliary feedwater storage tank (AFST). Therefore, provide long term analyses for a spectrum of small break LOCAs that demonstrate that decay heat can be adequately removed and the RCS depressurized using only safety grade equipment and water sources, assuming loss of offsite power and the most severe single failure. If credit is taken for operator actions, the STP emergency response guideline (ERG) sequence of operator actions should be followed. Justify the timing of operator actions if they are less conservative than those recommended in ANSI N-660 for a condition IV event.

b. In a conference call held on March 8, 1985, the applicant indicated to NRC that for small break LOCAs the combined heat sink capacity of the RWST and the steam generators would provide core cooling for approximately 18 hours, after which the reactor containment fan coolers (RCFCs) would provide an adequate heat sink for decay heat removal. No credit is taken for heat removal by the RHR heat exchangers. Provide a detailed explanation of the mechanism of energy removal from the RCS after loss of the secondary heat sink and supporting analyses that demonstrate that energy can be adequately removed to meet the acceptance criteria of 10 CFR Part 50.46. We are concerned that for very small break LOCAs (e.g., 1 inch) energy would not be adequately removed from the RCS for a considerable period of time after the accident. Thus, WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System" June 1979, indicates that for 1 inch breaks the break can remove all the decay heat only after about 24 hours, and that prior to that time, auxiliary feedwater is required to maintain the heat sink.

Response

~~This response will be provided in a later amendment.~~

see Insert X

# Insert X

## Large Break LOCA

For large break LOCA (breaks greater than 1 sq. ft.) the break will cause a significant Reactor Coolant System (RCS) depressurization. Breaks of this size are not isolable so the sump is used for long term cooling and makeup. Breaks considered large breaks will have sufficient energy removal through the break to sump flow path to remove decay heat energy. Sufficient make-up capability to keep the core adequately cooled and to meet 10CFR Part 50.46 (b) (5) requirements is provided. Containment heat removal will be provided in the STP design by both containment fan coolers and low head safety injection (LHSI) recirculation flow which is cooled by the RHR heat exchangers. Equipment relied upon is fully qualified for the environmental conditions that prevail during the accident.

## Small Break LOCA

As result of the accident at Three Mile Island Unit 2, Westinghouse performed extensive analyses that focused on the behavior of small break loss of coolant accidents (SBLOCA) for the Westinghouse NSSS. The purpose of the analyses was to demonstrate adequacy of the Westinghouse NSSS design in mitigation and long term recovery from a range of breaks classified as small breaks (less than 1 sq. ft. area).

The results of the analyses were reported in WCAP-9600, "Report on Small Break Accident for Westinghouse NSSS System," dated June 1979. The "Small Break Evaluation Model" at that time consisted of the WFLASH thermal-hydraulic code and the LOCTA fuel rod model. The analyses were performed for generic application using a standard 4-loop Westinghouse designed, a standard 3 loop and standard 2 loop depending on the nature of the study and which plant type was expected to be bounding. The conclusions are applicable for all Westinghouse designs, including STP with exceptions as described in the following.

## STP SBLOCA Design Features

STP has a three train low pressure SI system consisting of three high head SI (HHSI) pumps, three LHSI pumps, and three accumulators. Each train is aligned to a separate RCS loop. The pressure ranges for the SI pumps follow:

HHSI: 0 - 1445 psig
LHSI: 0 - 283 psig

Insert X (Continued)

For recirculation, the LHSI and HHSI pumps take suction directly from the sump. The LHSI pump flow passes through the RHR heat exchanger and is cooled before entering the RCS.

The plant has three motor driven auxiliary feedwater (AFW) pumps and one turbine driven auxiliary feedwater pump. The normal system alignment connects each AFW pump directly to one steam generator. The steam does not have a common header, but cross connections exist in the AFW lines. The valves in the cross connections are normally closed and fail closed. Two motor driven AFW pumps and the turbine driven AFW pump are required operable by the Technical Specifications.

The Auxiliary Feedwater Storage Tank (AFST) has a useable capacity of 525,000 gallons.

Non-safety grade sources of condensate grade make-up to the AFST are:

- o Demineralized Water Storage Tank - One 1,000,000 gallon storage tank shared between units.
- o Secondary Make-up Tank - One 300,000 gallon storage tank per unit
- o Condenser Hotwell - about 100,000 gallons per unit

While these tank volumes are not covered by Technical Specifications or other administrative controls, it would be very improbable to have less than 500,000 gallons of condensate grade water available for each unit.

The limiting single failure for the STP design will result in the loss of one train of safety injection (1 LHSI and HHSI pump) and one AFW pump. Since one AFW pump is allowed out-of-service for maintenance, this will result in the ability to feed two steam generators.

The STP design provides means to remove energy through the steam generators (AFW and atmospheric relief valves), through Containment steam condensation (fan coolers) and through the RHR heat exchangers (LHSI pumps and RHR heat exchangers). In this way energy is removed from containment sump water (RHR heat exchangers) so that relatively cool water will be continued to be supplied as make-up and for decay heat removal.

For all break sizes, heat is removed from the core by the break and steam generators. AFW is required for secondary inventory and heat removal until the break is able to remove all the decay heat or the RHR System is placed in operation. The break removes energy from the RCS because the makeup water from the RWST is relatively cold and can absorb energy before exiting the RCS. The WCAP-9600 analyses with consideration of STP design features and STP analyses of long term cooling discussed in the report titled "Long Term Cooling Analysis for South Texas Project" demonstrate decay heat removal capability for SBLOCA. The Long Term Cooling Report was transmitted in HL&P letter ST-HL-AE-1767 dated September 30, 1986.

Insert X (Continued)

SBLOCA Response

The initiating event is the break. If the break is 3/8" or less equivalent diameter and the charging system and feedwater system are available, the event is classified as a leak since normal charging flow would be sufficient to keep up with leak flow without a significant RCS depressurization. There would not be an automatic reactor trip or safety injection signal.

For breaks larger than 3/8", automatic reactor trip and safety injection will occur due to RCS depressurization caused by the loss of primary inventory. After reactor trip and safety injection initiation, safety injection pump flow provides makeup to the RCS and maximum peak clad temperature will remain below 10CFR50.46 Appendix K criteria.

For breaks greater than 3/8" and less than 1.5", SI flow can match break flow so no significant RCS depressurization or core uncovery will occur. At the point where SI flow matches break flow, the mitigation phase of the accident ends and a long term decay heat removal phase begins. The operator will cool down and depressurize to below the shutoff head pressure of the LHSI pumps (283 psig). This will be accomplished using the steam generator PORVs for cooldown and pressurizer PORVs in combination with HHSI flow termination for depressurization. The detailed actions will be provided in the STP Emergency Procedures which are based on the WOG Emergency Response Guidelines. The RHRS will be available to provide heat removal at RCS pressures below 350 psig and temperatures below 350$^{o}$F. Adequate long term decay heat removal will be provided by LHSI pump flow through an RHR heat exchanger in addition to RHRS operation.

For breaks from 1.5" to 4", the operator will cool down and depressurize the RCS to a pressure below the shutoff head pressure of the LHSI. The combined heat sink capacity of the Refueling Water Storage Tank and the steam generators would provide core cooling until the containment fan coolers and the RHR heat exchangers via LHSI pumps provide an adequate heat sink for decay removal.

For breaks greater than 4", the decay heat will be removed by the break and the containment fan coolers and the RHR heat exchangers via LHSI pumps. No operator action is required.

For isolable breaks, the operator will cool down and depressurize the RCS via a sufficient quantity of auxiliary feedwater to RHRS cut-in conditions of 350 psig and 350$^{o}$F. Adequate long term decay heat removal will then be provided via the Residual Heat Removal System.

Question 440.39N

a. It is stated in 10 CFR Part 50.46(b)(5) that, for long term cooling, "the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-term radioactivity remaining in the core." In order to assure this, heat removal for this extended period must utilize equipment that is fully qualified for the environmental conditions that prevail during the accident. Please demonstrate that decay heat can be removed from the STP core with qualified equipment only, following all sizes of LOCAs, including all LOCAs which could be subsequently isolated by the operator. Include consideration of the post-LOCA cooldown period in your response, and the fact that for isolated LOCAs, the sump would not be available for long term cooling.

b. Discuss whether the RHR pumps are qualified for the environmental effects of the large and small break LOCAs and steam line breaks. If the RHR pumps are not qualified discuss how long term mitigation of these accidents would be accomplished.

Response

~~This response will be provided in a later amendment.~~

a. See the response to NRC Question 440.038N.

b. The RHR pumps are qualified for the containment environment following a DBA, including small LOCA and secondary breaks. The NRC was notified regarding the qualification of the RHR pumps via ST-HL-AE-1684 dated June 17, 1986.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.6 RHR

LIMITING CONDITION FOR OPERATION

3.5.6 Three independent Residual Heat Removal (RHR) loops shall be OPERABLE with each loop comprised of:

a. One OPERABLE RHR pump,
b. one OPERABLE RHR heat exchanger, and
c. one OPERABLE flowpath capable of taking suction from its associated RCS, hot leg and discharging to its associated RCS cold leg.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

a. With one RHR loop inoperable, restore the required loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
b. With two RHR loops inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.
c. With three RHR loops inoperable, immediately initiate corrective action to restore at least one RHR loop to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.5.6 Each RHR loop shall be demonstrated OPERABLE pursuant to the requirements of specification 4.0.5

Enclosure 1