UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FACILITY OPERATING LICENSE NO. NPF-76

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

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1 INTRODUCTION

Supplement Nos. 6 and 7 to the Safety Evaluation Report (SER) NUREG-0781 were issued in January and March 1989. The supplements discussed the evaluation of issues pertaining to the operation of South Texas Project, Unit 2 (STP-2) and were issued in support of the low-power and full-power licenses for STP-2. Although the supplements were written to resolve issues for STP-2, many of the issues also apply to South Texas Project, Unit 1 (STP-1). This Safety Evaluation (SE) addresses those issues which were resolved for STP-2 and which also apply to STP-1. Each of the sections is numbered the same as the corresponding SER section. Each section is supplementary to and not in lieu of the discussion in the SER and its supplements.

2 SITE CHARACTERISTICS

2.4 Hydrologic Engineering

2.4.14 Technical Specifications and Emergency Operating Requirements

In Sections 2.4.11.2 and 2.4.14 of the SER, issued April 1986, the staff indicated that South Texas Project (STP) would be permitted to operate only when the water temperature in the Essential Cooling Pond (ECP) was less than a maximum value. At STP, the ECP is the Ultimate Heat Sink (UHS). The April 1986 SER did not specify what the maximum value would be.

In SSER 1, issued September 1986, the staff indicated that, based on the licensee's analysis of the thermal performance of the ECP, the maximum temperature would be 95°F on the intake side of the pond. The staff characterized this temperature as the maximum pond temperature at the start of the design basis accident (DBA). Based on this information the staff recommended that the Limiting Condition for Operation, Technical Specification Section 3.7.5, "Ultimate Heat Sink", should establish a maximum temperature in the ECP of 95°F.

By letters dated January 13 and March 9, 1987, the licensee indicated that the maximum temperature cited in SSER 1, Section 2.4.14, was incorrect and requested that Technical Specification (TS) Section 3.7.5(b) be changed accordingly. In addition, the licensee's January 9, 1987 letter requested that Technical Specification Section 3.7.5 be revised to permit continued operation of the plant for 72 hours with the UHS inoperable.

The licensee indicated that 95°F is the normal operating temperature of the ECP. This temperature is based on the design of the ECP and normal meteorological conditions at the STP site. To determine the maximum water temperature in the ECP at the start of the DBA, a 20-day initialization period using worst case historical meteorological data for the STP site was applied to the 95°F normal operating temperature. This 20-day initialization period produced a maximum water temperature of 99°F at the intake side of the pond. The analyses used by the licensee to assess the thermal performance of the ECP are discussed in Final Safety Analysis Report (FSAR) Section 9.2.4.3.3. The licensee's analyses comply with the regulatory positions of Regulatory Guide (R.G.) 1.27 and were previously accepted by the staff.

The licensee has performed a thermal analysis of the ECP during design accident conditions which indicates that with a temperature of 99°F at the pump suction side of the ECP the maximum outlet temperature of the component cooling water (CCW) Heat Exchanger (for the unit experiencing DBA) will be 120.5°F. The licensee has also identified the Reactor Containment Fan Coolers as the most temperature sensitive equipment cooled by CCW, with a limiting temperature of 125°F from the CCW system. The staff's interpretation of the meteorological requirements would have used a slightly different alignment of the meteorological historical record which would result in a maximum outlet temperature of the CCW Heat Exchanger (LOCA unit) of about 123°F which is still well within the 125°F limit for the Reactor Containment Fan Coolers.

Since the licensee's analysis indicates that, during periods of extreme hot weather, the temperature at the intake side of the ECP may reach 99°F and that this temperature will not affect the operation of the Reactor Containment Fan Coolers, the licensee requested that Section 3.7.5 of the TS be revised. This request was granted in the combined TS issued with the STP-2 low-power license since the maximum temperature of 99°F will not violate the CCW system design criterion.

Technical Specification Section 3.7.5, "Ultimate Heat Sink", contains two criteria which determine if the UHS is operable. These criteria include a minimum water level at or above 25.5 feet mean sea level, and an average water temperature equal to or less than 99°F. If the criteria are not met, the existing TS Action Statement requires that the plant be in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. By letter dated January 13, 1987, the licensee requested that the Action Statement be revised to permit continued operation for 72 hours with the UHS inoperable prior to initiating shutdown. The licensee's request was denied. Technical Specification Section 4.7.5 provides surveillance requirements for the UHS. This technical specification section requires the plant operator to determine if the UHS is operable at least once every 24 hours by verifying that the water level and temperature are within the limits set by Technical Specification Section 3.7.5. During plant operations, periodic surveillance of the UHS will indicate if the water level and temperature in the UHS are approaching the limits set in TS Section 3.7.5. This surveillance will give the plant operators sufficient time to take appropriate corrective actions to prevent the water level and/or temperature in the UHS from reaching their limits. If the corrective actions taken by plant operators are not successful, such that the UHS becomes inoperable, an additional 72 hours of operation with continuing corrective actions would not be justified.

In conclusion, the licensee was permitted to use 99°F as the maximum water temperature of the ECP at the intake structure. This temperature is the maximum permitted by the CCW system design criteria and is in compliance with the guidelines of R.G. 1.27 and GDC 44 of Appendix A to 10 CFR Part 50.

Additionally, the licensee was not permitted to continue plant operations for 72 hours after determining that the UHS is inoperable. Continued operation with the UHS inoperable would be a violation of GDC 44 of Appendix A to 10 CFR Part 50.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

3.9.3.1.1 Thermal Stresses in Piping Connected to the RCS-Bulletin 88-08

Bulletin 88-08 (Thermal Stresses in Piping Connected to Reactor Coolant Systems) requested that each licensee (1) review the reactor coolant system to identify any connected, unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses, and (2) take action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses.

On September 28, 1988, HL&P provided its response to the bulletin action items in accordance with the reporting requirements as stated in the bulletin. In response to Action 1, HL&P stated that all systems connected to the RCS in Unit 1 were reviewed, and sections which cannot be isolated and are susceptible to thermal stress oscillation were identified. These included a portion of the auxiliary spray line and the chemical and volume control system (CVCS) normal and alternate charging lines.

In response to Action Item 2, HL&P intends to perform a non-destructive examination of the sections which cannot be isolated to verify that there are no existing flaws.

In response to Action Item 3, HL&P committed to develop a program to provide the required continuing assurance by December 1, 1988 and have it implemented prior to startup from the first refueling outage.

HL&P has satisfied Reporting Requirement No. 1 of the bulletin in its letter dated September 29, 1988. Reporting Requirement No. 2 will be satisfied within 30 days of the completion of Actions 2 and 3. The staff finds this acceptable.

3.9.3.5 Non-Conforming Materials - Bulletin 88-05

NRC Bulletin No. 88-05, issued May 6, 1988, required holders of construction permits (CP) and operating licenses (OL) to submit information regarding materials supplied by Piping Supplies, Incorporated (PSI) at Folsom, New Jersey, West Jersey Manufacturing Company (WJM) at Williamstown, New Jersey, and Chews Landing Manufacturing Incorporated (CL). The bulletin requested that licensees: (1) take action to assure that materials comply with ASME Code and design specification requirements or are suitable for their intended service; or (2) replace such materials. The NRC action was precipitated by the discovery that certified material test reports (CMTRs) for material supplied by PSI and WJM contained false information about material supplied to the nuclear industry.

A number of CMTRs were apparently used to certify that commercial-grade steel met the requirements of ASME Code Section III, Subarticle NCA-3800, by using a domestic forging company's letterhead.

The licensee's response consisted of a letter dated September 8, 1988. The report described the methodology used to identify and test the nonconforming parts, contained a summary of the test results, and presented the engineering evaluations and analyses.

The licensee conducted a multi-faceted program to identify and locate materials supplied by the suppliers identified. HL&P undertook a comprehensive evaluation in concert with the STP contractors, other utilities, and the Nuclear Management and Resources Council (NUMARC). The overall effort involved extensive document searches, field walkdowns, extensive field and laboratory testing, nationwide coordination of information, and engineering evaluation of results.

Following receipt of the bulletin, the site bulk material heat logs were reviewed to determine which heats of bulk material were manufactured by West Jersey Manufacturing (WJM) or Piping Supplies, Inc. (PSI). The review identified each heat of bulk material which was field installed in safety-related piping systems. This review revealed that only flange material was supplied in bulk by these manufacturers.

In order to substantiate the accuracy of that review, a complete review of installation documentation was performed to identify the specified manufacturer of each flange installed in the plant. This review validated the accuracy of the warehouse heat logs and provided the specific installed location of each WJM/PSI flange. No Chews Landing (CL) material was identified in either review.

To complete the search, a comprehensive review was conducted of vendor component data packages to identify WJM/PSI materials supplied with those components. This review identified a number of valves, heat exchangers, strainers, and skid-mounting piping systems containing WJM/PSI flanges. There were no blind flanges identified.

An additional search was conducted for non-flange product forms as described in the bulletin and supplements. This review effort concluded that no non-flange product forms were supplied to STP by the suspect suppliers.

When completed, the record review effort identified that a total of 110 flanges supplied by WJM or PSI were installed in STP-1. Two types of flange material were identified - SA105 and SA350/LF2. All remaining warehouse stock of WJM/PSI material was segregated in the warehouse pending final resolution of this issue.

On the basis of its review, the staff finds that HL&P conducted a thorough and comprehensive search to identify and locate nonconforming flanges and fittings supplied by PSI/WJM/CL in response to the requirements of Bulletin 88-05, and Supplements 1 and 2. The staff also finds that HL&P was responsive to the action and reporting requirements of Bulletin 88-05, Supplements 1 and 2, and that there is a high probability that all nonconforming flanges and fittings have been identified. The staff concludes that HL&P's identification efforts provide an adequate basis to resolve the nonconforming material identification concerns described in Bulletin 88-05 and are acceptable.

Description of Licensee's Test Program

HL&P conducted field Equotip tests for each installed flange and provided a number of WJM/PSI supplied SA105 flanges from warehouse inventory to NUMARC for laboratory analysis. They also provided the data obtained from their record reviews and field testing for use in the NUMARC data base. At the request of the NRC, additional chemical analyses were performed on flanges that had hardness readings below a Brinell Hardness Number (BHN) of 137.

There were six different heats of field installed SA105 flanges. Each piece was field hardness tested and at least one piece from each heat was laboratory tested (12 pieces total). Additional laboratory chemical analyses were performed on randomly selected warehouse specimens from the six heats. All test results were within the SA105 chemistry ranges.

There were nine different heats of vendor installed flanges, one of the heats is common to one of the above mentioned field installed heats. Of the eight other heats, four had at least one flange outside the BHN range of 137-187. Chemical analyses were performed on filings removed from the shoulder of six flanges whose Equotip test results were suspect. These six samples represented the four heats which had flanges outside the 137-187 BHN range. Two of the six flanges had chemistry outside the specification range. All of the flanges that were nonconforming due to low hardness or to chemistry were subject to a structural analysis which assumed reduced strength properties.

There were four heats of field installed SA350/LF2 flanges, one heat was laboratory tested as SA105 material (it was certified for both specifications). Another of the heats was laboratory tested for hardness and met specifications. Additional laboratory chemical and mechanical testing on one flange from each of the four heats indicated they were all within specification.

The staff was concerned with an apparent pattern of consistently high hardness readings on one heat of SA350/LF2 material (heat number 1705), but additional warehouse and laboratory hardness tests on cross sectioned flanges of this heat showed acceptable hardness across the entire volume of the flange and on the flange face. The high hardness readings in the field were determined to be due to a surface hardness condition on the shoulder of the flange (the only accessible surface for field testing). The surface hardening is the result of a quenching and tempering operation. For flanges that were outside the acceptable BHN range and had nonconforming chemistry, the staff examined mechanical property data from actual tensile tests and chemical sample data from the same heat of material and concluded that, in view of the conservative nature of the ASTM A370 hardness conversion used, the strength reduction assumptions used in the structural analysis were conservative. The tensile tests indicated that the actual strength ranges were not at a level where stress corrosion cracking would be a concern. Further, there is no weldability concern on an already installed and inspected flange.

Based on the above described material mechanical property and chemical testing which was supported by the NUMARC industry wide testing program, and the structural analyses below, the staff concludes that the components in question are acceptable for their intended use.

Evaluation of Licensee's Structural Analyses of Nonconforming Parts

Structural evaluation of the nonconforming flanges was based on the assumption that the reduced flange capacity is linearly dependent on the yield strength of the material. ANSI B16.5 indicates that flange pressure-temperature ratings are proportional to the yield strength of the material. ASME NC/ND-3658 contains equations which indicate that the maximum flange moment capacity is linearly dependent on the yield strength of the flange material. Table 3 of HL&P's report identified flanges with computed ultimate tensile strengths less than the required 66 thousand pounds per square inch (ksi). Reduced allowable moments and flange pressure ratings were presented along with the design values determined in the original piping analyses. In all cases, the moment loadings were found to be substantially below the reduced allowable values. The actual flange design pressures were found to be less than the reduced allowable values.

Conclusions

On the basis of its review of the licensee submittals, the staff finds that HL&P conducted an adequate material property and structural analysis of the nonconforming flanges and fittings using acceptable and conservative analytical methods, and evaluation criteria. The staff also finds that HL&P was responsive to the action and reporting requirements of Bulletin 88-05, Supplements 1 and 2, and that HL&P has qualified all nonconforming parts as being suitable for the intended service. The staff concludes that the analytical procedures used by HL&P to qualify the nonconforming parts located in STP-1 and the results of the analyses provide an adequate basis for qualifying the nonconforming parts as being suitable for the intended service. The staff concludes that the analytical procedures used by HL&P to qualify the nonconforming parts and the results of the analyses provide an adequate basis for resolving the concerns with respect to demonstrating adequacy for service. The staff does not consider the nonconforming parts to be ASME Code material. The use of this material is an acceptable alternative in accordance with 10 CFR 50.55a(3)(ii) because full compliance with all specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality or safety.

3.9.6 Inservice Testing of Pumps and Valves

The staff's review of the IST program differences between Unit 1 and Unit 2 indicated that the Unit 2 IST Valve List contains valves, FV-1025, -1026, -1027, and -1028, which are missing from the Unit 1 IST Valve List. A review of the piping & instrumentation diagrams (P&IDs) by the staff's contractor, EG&G Idaho, showed that these valves perform safety-related containment isolation functions in both Units. During a conference call on December 4, 1988, the licensee agreed to incorporate these valves into the Unit 1 IST program.

4 REACTOR

4.4 Thermal Hydraulics Design

4.4.3 Design Abnormalities

Resistance Temperature Detector Time Constant

By letter dated August 2, 1985, the licensee indicated that the reactor coolant temperature measurement system for the hot legs would be modified. The modification eliminated the resistance temperature detector (RTD) bypass manifold and implemented a new method of measuring hot leg temperatures by using RTDs in the thermowells. This change increased the RTD response time from 6.0 seconds to 6.5 seconds. South Texas Project is the first plant where a change in the method of measuring the hot and cold leg reactor coolant temperatures has been implemented. The staff's evaluation of the modifications was discussed in SSER 2, dated January 1987.

In a letter dated November 12, 1987, the licensee indicated that the RTD response time was longer than that stated in SSER 2 and specified in the TS. Therefore, the licensee proposed that the TS be modified to show an increase in the RTD response time from 6.5 seconds to 8.0 seconds. The letter included the proposed TS changes, revised pages of the FSAR and the reanalysis of FSAR Chapter 15 accidents affected by the increase in RTD response time.

By letter dated August 24, 1988, the licensee submitted additional clarifying information concerning two responses given in a December 23, 1987 letter.

The original modification stated that the increased RTD response time was found to meet the design basis departure from nucleate boiling ratio (DNBR) in a plant specific analysis for a steamline rupture at power. The licensee stated in the letter dated August 24, 1988, that the steamline rupture at zero power was more limiting, but was not dependent on RTD response time.

The licensee also revised the December 23, 1987 response by noting that in addition to the required TS response time check for the RTDs, the RTDs are cross-calibrated during heat-up after each refueling. This cross-calibration is a common practice for Westinghouse plants.

In SSER 2 and SSER 5, the staff evaluated and found acceptable the elimination of the RTD bypass system at South Texas Project and the effect on the FSAR Chapter 15 non-LOCA analyses. The licensee has demonstrated that the conclusions in the FSAR remain valid and the DNBR limit value is met. The staff has reviewed the additional information submitted by the licensee and finds that it does not materially change those conclusions. Thus, the changes are acceptable.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Boundary

5.2.1 Compliance with Codes and Code Cases

Background

The pressurizer surge line (PSL) in STP-1 is a 16 inch schedule 160 stainless steel pipe, 80 feet in length, connecting the bottom of the pressurizer vessel to the hot leg of one of the coolant loops. The outflow of pressurizer water is generally warmer than the water in the hot leg. Such a temperature differential (T) varies with the plant operation activities and can be as high as 300°F in STP during its initial plant heat up. Thermal stratification is the separation of hot and cold flow streams in the horizontal portion of the PSL resulting in a temperature difference at the top and bottom of the pipe. The potential for stratification is increased as T increases and as the pressurizer insurge or outsurge flow rate decreases. Stratification in a PSL was found recently and confirmed by data measured from several pressurized water reactor (PWR) plants.

In the STP original PSL design analysis, the insurges or outsurges were assumed to sweep the fluid along the line, resulting in uniform thermal loading at any particular piping location. Such analysis did not reflect PSL actual thermal condition and potentially may overlook undesirable line deflection and its actual high stresses may exceed design limits. In addition, the striping phenomenon, which may induce high cycle fatigue to the inner pipe wall, was not analyzed. Thus, assessment of stratification effects on the PSL is necessary to ensure piping integrity and Code conformance.

Since stratification in the PSL is a generic concern to all PWRs, an NRC Information Notice No. 88-80 was issued on October 7, 1988. An NRC Bulletin (88-11, dated December 20, 1988) was published after the review of STP-1 and 2 was underway. On November 30, 1988, the licensee and Westinghouse presented the staff with results of a bounding analysis of PSL to account for the stratification effects. Subsequently, a report was submitted by HL&P dated December 9, 1988, which consisted of presentation material and explanatory text. The following is the staff's evaluation of the licensee's efforts and information provided in the report and presentation.

Initial Evaluation

The licensee had instrumented the PSL and collected data for verifying stratification conditions in STP. Such data was utilized in conjunction with data collected from PSLs of three other Westinghouse designed PWRs for updating design thermal transients and for developing flow stratification profiles. Conservative enveloping techniques were adopted based on the best available information. The staff found that the licensee's efforts for updating PSL stratification conditions were comprehensive, and generally acceptable. However, due to the high sensitivity of PSL stratification to normal letdown, charging and pressurizer spray activities, the up-to-date data was insufficient to verify a complete list of design thermal transients due to relatively short monitoring duration. The licensee indicated that PSL monitoring would continue until the next refueling outage in STP Unit 1 for more complete data collection. The staff concurred with the licensee's approach.

The staff found that the PSL thermal striping phenomenon was inadequately explained due to the lack of verification in defining the amplitude, duration and oscillation frequency. The staff indicated that more evidence was needed to confirm that the assumptions used are indeed conservative.

The licensee had performed a reanalysis of PSL piping and supports to account for thermal stratification effects. The analysis consisted of three parts: (1) global bending effects on stresses, moments, displacements, and support reaction loads, based on both axial and radial variations in the pipe metal temperature; (2) local stresses due to thermal gradients; and (3) local stresses and effects to fatigue due to thermal striping. The global and local stresses in items (1) and (2) were superimposed to obtain the total stresses.

The staff found that the approaches used for performing PSL reanalysis were generally acceptable. In addition, the licensee indicated that one pipe support was removed to accommodate thermal expansion. The staff noted that the licensee's contention that the PSL stresses after support removal meet the limits of NB-3600 Equation (12) in the ASME Code, Section III, and pipe movements will be reviewed for clearance consideration and verified during the next plant heatup for Unit 2.

The staff's evaluation of information provided by the licensee concluded that the information was comprehensive and generally acceptable. However, additional information was needed for completion of the staff's review prior to issuance of the full-power license. The additional information needed included: (1) confirmation of design transients when adequate PSL data is collected from the on-going monitoring programs in STP or other plants; (2) evidence for quantitative verification of striping phenomenon, including its amplitude, duration, and frequency; (3) a more detailed explanation to verify the credibility of the linear equivalent techniques used in the analysis or global bounding effects; (4) justification for not considering mean stress effects in determining fatigue usage factor contributed by thermal striping; and (5) verification of the sizes of leakage flaws for "leak-before-break" analysis.

HL&P submitted additional information by letters dated January 27, February 1, and February 15, 1989. In addition, the staff reviewed the licensee's response to NRC Bulletin 88-11 regarding PSL thermal stratification. The staff conducted a detailed review of the piping stress calculation packages, which included Westinghouse and Bechtel calculations, to ascertain ASME Code compliance and a walkdown of the pressurizer surge line to observe any evidence of thermal interference and discernible distress. This review was conducted as a followup to the initial review.

Final Evaluation

The staff performed a review of design thermal transients that were based on enveloping the best known data from several Westinghouse plants. The approach is conservative and acceptable for providing input to the bounding analysis performed by HL&P. HL&P has instrumented the surge line in Unit 1. As indicated in a letter dated February 1, 1989, the monitoring program will continue until the first refueling outage in Unit 1. Once the program is completed, HL&P will review the data from South Texas and available industry data to confirm the acceptability of the currently used data on transients.

The staff discussed the basis for defining amplitude, duration, and oscillation frequency of thermal striping with Westinghouse personnel. It also reviewed the results of several flow tests conducted by Westinghouse in its Waltz Mill Laboratory. Westinghouse also provided a detailed description of its flow tests in Revision 1 to WCAP-12067. The amplitude used by Westinghouse was conservative in comparison with amplitudes actually observed in flow tests. The durations were adequately defined to account for the decaying of striping effects on stress and fatigue calculations. The staff found the approaches acceptable.

The stratification induced by global bending of the surge line in the South Texas plant was calculated by Westinghouse using both the WECAN and ANSYS computer codes. For the WECAN computation, a finite element piping structural model with step-change thermal profiles was used. For the ANSYS computation, a conventional pipe element model with linear thermal profiles was used to calculate equivalent nonlinear effects. The staff reviewed sample calculations and discussed the analysis techniques with Westinghouse personnel. The staff's review verified that the results of both computations were similar in regard to calculated surge line displacements, which compared favorably with displacement data obtained by measurements in South Texas Unit 1. Westinghouse indicated that it will continue to compare linearly calculated results with data obtained by measurements for the first few Westinghouse plants when the monitoring program is implemented in these plants. The staff concluded the calculated results and those obtained by measurements are adequate and acceptable for the surge line in the South Texas plant.

The staff found that HL&P's response regarding the effects of mean stress on fatigue calculations for thermal striping was inadequately described in WCAP-12067, Supplement 1 to Revision 1. Additional information was provided in a letter dated February 9, 1989. HL&P indicated that the maximum effect of the mean stress was included in a curve in the ASME Code. Although any value of a mean stress above the curve values was not considered in the striping analysis, it was judged not to be necessary because of the various conservatisms involved in the striping analysis process and the fatigue calculations process. The staff reviewed the additional information and found it acceptable.

The staff reviewed the surge line piping analysis performed by Westinghouse and Bechtel - the nuclear steam supply system (NSSS) supplier and the architect-engineer (AE), respectively, for the South Texas plant. The review included a detailed review of the calculation packages and verification of the proper handling of the interface between the NSSS supplier and the AE during various phases of the analysis. The staff found that all the required loadings had been considered in the calculations. The stresses were properly combined to meet the limits delineated in ASME Code, Section III, Subsection NB-3650. The staff found that the design calculations and piping isometric drawings had been updated to reflect design changes. In addition, it found that the interface between the NSSS and the AE was appropriate.

The staff performed a detailed audit of the pipe support calculations. There is only one support in the surge line. The support was designed by Bechtel with the required stiffness specified by Westinghouse. The review showed that the support stiffness conforms with the requirements and is acceptable.

The staff conducted a walkdown along the entire surge line in Unit 2 and a detailed review of the as-built piping isometric drawings. The staff found that the clearances at the wall penetrations are adequate to accommodate thermal expansion of the surge line. The as-built configuration appeared to be correctly reflected in the isometric drawing. The removal of one support was properly indicated. In addition, although the surge line had experienced heatup transients, the staff found no discernible distress in the piping and pipe supports.

In the letter dated February 15, 1989, HL&P submitted additional information on the reevaluation of the pressurizer surge lines using "leak-before-break" (LBB) technology as permitted by revised General Design Criterion 4 (GDC 4) of Appendix A to 10 CFR Part 50. The additional information was submitted in response to the staff's request for additional information dated January 12, 1989, and was provided in Westinghouse's Supplement 1 to Revision 1 of WCAP-12067.

Previously, the staff had found the pressurizer surge lines in compliance with revised GDC 4 using LBB technology. However, the reported phenomenon of thermal stratification in pressurizer surge lines necessitated a reevaluation of the pressurizer surge lines as discussed in NRC Bulletin 88-11.

The staff had two concerns relative to HL&P's LBB reevaluation of the pressurizer surge lines. The first concern was that there was a discrepancy between the staff estimates and HL&P's estimates of the size of the leakage flaw in the LBB evaluation (SSER 6). HL&P provided information that showed that the computer code used did provide conservative best-estimate results. This was done by comparing predicted crack sizes with both plant data obtained by measurements and experimental results. The staff found the additional information provided by HL&P adequate for resolving this issue for South Texas Project, Units 1 and 2. The second concern related to the stability of a flaw in the pressurizer surge line during a forced cooldown on the discovery of a leak in the surge line. This concern was raised because the loads on the surge line would increase during depressurization, which would be necessary in order to repair the leak. The staff finds (1) the additional information provided by HL&P and (2) HL&P's commitment to revise plant operating procedures to provide prompt depressurization in the event of leaks adequate for resolving this issue for South Texas Project, Units 1 and 2.

On the basis of the above regarding the LBB reevaluation, the staff finds that the previous conclusion (that the pressurizer surge lines in South Texas Project, Units 1 and 2, were in compliance with revised GDC 4) is still valid.

On the basis of the review and inspection, the staff concludes that HL&P has made acceptable efforts to meet Action Items 1.a and 1.b as delineated in NRC Bulletin 88-11. The efforts demonstrate that, on the basis of the available stratification data, the surge line meets the applicable design codes. Pipe movements of the pressurizer surge line also will be reviewed and verified during the next plant heatup for Unit 2, scheduled to be part of the bottom-mounted instrument inspection outage, to ensure that clearances have been considered. Additionally, HL&P will verify the stress and fatigue analyses to ensure compliance with the ASME Code when the plant-specific data from the Unit 1 monitoring program are completed during its first refueling outage.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2 Specific Findings

7.2.2.2 Design Modification for Automatic Trip Using Shunt Coil Trip Attachment

In the SER, the staff reported on its review of the plant specific submittal for this issue. The staff concluded that the proposed design for the automatic actuation of the reactor trip breakers shunt trip attachments is acceptable except for the breaker response time testing, which should be included in the Technical Specifications.

Upon further review of the licensee submittal dated October 14, 1985, the staff noted that Westinghouse prepared a report of the reactor trip breaker Undervoltage Trip Attachment (UTA) and Shunt Trip Attachment (STA) life cycle test which concluded that periodic testing for STA can be limited to verifying that it can trip the breaker with 70Vdc (minimum design voltage). Since both the UTA and STA trip the same breaker mechanism and response time is determined through the UTA testing periodic testing of the automatic shunt trip feature, response time is not required. This is acceptable.

7.5 Information Systems Important to Safety

7.5.2 Specific Findings

7.5.2.9 Qualified Display Processing System (QDPS)

The licensee provided additional information to the staff in a letter dated April 29, 1988. The information provided was to notify the staff that the Qualified Display Processing System (QDPS) Verification and Validation (V&V) Plan had been modified. The original review and acceptance of the V&V plan was discussed in Section 7.5.2.9 of SSER 4, dated July 1987. The modification to the V&V plan was made because the QDPS is now installed in both Units 1 and 2. Therefore, it is necessary for the validation of the programming modifications to be performed utilizing a test jig. The test jig will be a replica of those parts of the system affected by the group of changes being implemented. Configuration of the test jig will be documented and traceable to the hardware and software actually installed in the QDPS. Section 6.3 has been added to the STP QDPS V&V program.

The staff has reviewed the revised QDPS V&V program and concluded that it follows the guidance of ANSI/IEEE - ANS 7.4.3.2 - 1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations" and R.G. 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plant". Therefore, the staff concludes that the revised QDPS V&V program is acceptable.

8 ELECTRICAL POWER

8.3 Onsite Power System

8.3.3 Compliance with GDC

8.3.3.3 Physical Independence (Compliance with GDC 17)

8.3.3.3.3 Raceway and Cable Separation

In SSER 3, the staff concluded that the licensee's justification for the minimum separation of cables and raceways not meeting the requirements of R.G. 1.75 was acceptable. The basis for the staff's acceptance was the results of tests conducted by Wyle Laboratories for the licensee which demonstrated that the worst case electrical fault for given conditions of electrical separation would not propagate from the fault cable/raceway to a target cable/raceway. The test data from Wyle Laboratories, Report No. 53575, was submitted to the staff for review as an enclosure to a letter dated February 19, 1987.

In the February 19, 1987 letter, the licensee identified six electrical cable/raceway configurations that did not conform to the R.G. 1.75 separation criteria. These were shown to be acceptable by specific configurations and tests described in Report No. 53575. In all cases, the actual configuration was at least equal to, or more conservative than the corresponding configurations in the report. Based on this, the licensee stated that these six cable/raceway configurations would be considered as the "design basis" for STP, and acceptability for each occurrence would be on the basis of results set forth in Report No. 53575. No special documentation or justification would be prepared for these six configurations. All other nonconforming configurations for which the results of Report No. 53575 might be used as justification for acceptability would be documented by Nonconformance Reports (NCR), Field Change Requests (FCR), or an approved engineering document per Site Specific Procedure (SSP) 45.

The staff finds the licensee's methodology for utilizing the results of Report No. 53575 as previously described to be acceptable on the basis that it will eliminate paperwork which would serve no useful purpose. Documenting each occurrence of any of the six identified configurations with NCRs or FCRs would only serve to provide justification for an electrical cable/raceway configuration which the staff has already found acceptable. Thus, the NCRs or FCRs in these instances would be superfluous. The licensee should, however, maintain for audit purposes a record of each instance of a nonconforming configuration which has been justified on the basis of Report No. 53575 test results.

By letters dated October 29 and December 21, 1987, the licensee identified two more nonconforming electrical cable/raceway configurations. The acceptability of these configurations has been demonstrated by the results of Configuration 3, Test 3 of Report No. 53575. However, the licensee is required to generate a NCR or a FCR for each nonconforming condition. This requirement is a result of the licensee's commitment in the letter dated February 19, 1987, which was discussed earlier. In the letter dated December 21, 1987, the licensee requested relief from its commitment to document nonconforming conditions for the two configurations described in the letters dated October 29 and December 21, 1987. These configurations would also be considered as part of the STP design basis and treated the same way as the six previously identified and reported configurations.

The staff has reviewed the licensee's proposal and concluded that it is acceptable for the same reasons as stated above. As with the original six configurations, the licensee should maintain, for audit purposes, a record of each instance of a nonconforming configuration covered by the above two configurations and which have justified on the basis of Report No. 53575 test results.

9 AUXILIARY SYSTEMS

9.2 Water Systems

9.2.1 Service Water Systems (Auxiliary Cooling Water System and Essential Cooling Water System)

On April 1, 1988, HL&P noticed that several small bore socket connections (two inch and under) in the aluminum-bronze Essential Cooling Water (ECW) system at STP-1 were leaking. The leaks were categorized as seepage with leak rates less than 10 milliliters per day. Based on the first three samples removed and examined, HL&P concluded the leakage was caused by dealuminization, i.e., selective corrosion of a phase in the alloy structure of the valves and fittings. On April 21, 1988, a letter forwarding the plan to deal with the problem was submitted for STP-1.

By letter dated May 12, 1988, HL&P provided more details on the dealuminization phenomena. The following conclusions were reached:

- The nature of the corrosion was "dealloying", a phenomenon in which the aluminum in one of the microstructural phases selectively corroded, leaving the balance of the matrix intact.

- The material of the cast valves (ASME SB148 Grade CA954) and fittings (typically ASME SB148 Grade CA952) contained the Gamma-2 phase. This condition lent itself to selective corrosion of the Gamma-2 phase, causing dealloying, in severe corrosive environments.

- The attack was significant at crevices, tapering off in areas away from the crevice.

- The chemistry of the water in the socket crevices was significantly more acidic than the bulk water chemistry, thus causing the severe condition which, in combination with the metallurgical condition of the materials, resulted in the selective corrosion.

- Piping and weld metal had suffered no corrosion, demonstrating that alloy CA614 was not subject to the observed phenomenon.

The worst case dealloyed cross section of a fitting was evaluated for structural integrity. Data from the metallographic examinations were combined with stress analysis, structural evaluations, and estimates of the rate of dealloying, and showed that the components would not fail as a result of postulated load combinations. The results of failure analyses showed that due to its ductile behavior and low design stresses, the components would not undergo brittle failure.

In searching for corrosion resistant materials, HL&P concluded that both small bore fittings and valves could be fabricated from wrought aluminum-bronze grade CA614. It is a single phase alloy used in the ECW piping and has proven resistant to dealloying after substantial exposure to the operating environment at STP. In a letter dated May 12, 1988, the licensee committed to implement the permanent corrective action prior to the return to service after the first refueling outage. By letter dated October 11, 1989, HL&P stated that the small bore fittings and valves have been replaced with material which is not suscentable to dealuminization. The staff finds the proposed action and commitment acceptable.

9.5 Other Auxiliary Systems

9.5.5 Emergency Diesel Cooling Water System

Background

During prerequisite testing of the Emergency Diesel Generators (EDG) for Unit 2, it was discovered that several cylinder liner expansion seals (made of 321 stainless steel) were leaking. The purpose of the expansion seals is to allow the cylinder liners to expand and contract while providing a jacket water pressure boundary secondary seal between the cylinder liners and the rest of the diesel engine. The primary seal is the metal to metal contact between the cylinder liner and the cylinder block. There is one expansion seal for each cylinder.

Two expansion seals were removed and analyzed to determine the cause of failure. The results of the analysis showed that the seals had experienced Microbiologically Induced Corrosion (MIC) and transgranular stress corrosion cracking.

In addition to analysis, each seal was pressure tested at 35 pounds per square inch (psi), approximately twice normal operating pressure. A total of nine of the 39 expansion seals were found to be leaking, and were replaced. The licensee has provided a justification for not replacing all expansion seals. The justification is discussed below.

Staff Concern

As stated above, the expansion seals provide a secondary seal between the engine jacket water system and the remainder of the engine. Should the seals fail, any coolant which gets by the primary seal (cylinder liner to cylinder block contact) would go directly to the diesel engine sump where it would contaminate the engine lubricating oil. Should enough contamination occur, there would be a loss of lubrication and subsequent engine failure. It is acknowledged that all expansion seals have been subject to MIC to some degree. Therefore, a potential exists for common mode failure of all EDGs during operation.

Licensee Analysis and Corrective Actions

The licensee analyzed the expansion seals using fracture mechanics and the 35 psi pressure test of the seals. Based on these parameters, the licensee has reached the following conclusions:

1. The majority of "partially through-wall" cracks, if they exist in the expansion seals are categorized as non-propagating and will not grow over the life of the diesels.

2. For any "partially through-wall" cracks that border on being through-wall, a leak may develop, but will be small and will not undergo any noticeable growth over the life of the diesels. Any undetected through-wall cracks will react the same and will not undergo any noticeable growth over the life of the diesels.

3. Corrosion pits, although not specifically analyzed, are bounded by the crack analysis. Consequently it can be concluded that pits will not develop into fatigue cracks during service.

The above conclusions are based on there being no additional expansion seal degradation due to MIC. To preclude further degradation, the licensee has taken the following actions:

1. The diesel engine jacket water systems, including the expansion seals, were sterilized with hydrogen peroxide to kill any existing bacteria. All flush water used during startup activities has been treated to prevent MIC recontamination of the system.

2. During operations, Low Halogen Nitrite-Borate-Tolytriazole is added to the jacket water coolant to prevent general corrosion. The high pH of this fluid (greater than 10) will prevent recurrence of MIC. The nitrite in the corrosion inhibitor will reduce the oxygen content and create nitrate which acts to inhibit stress corrosion cracking.

In addition to the fracture mechanics analysis and jacket water treatment, the licensee has stated that the diesel engine lubricating oil will be analyzed for the presence of water on a monthly basis. The diesel engine vendor has stated that the maximum permissable contamination of oil with water is 1.0%. By analysis, water in oil can be detected at levels of 0.05% and above. Contamination of 0.05% water in oil represents about 1.05 gallons of water in the volume of oil normally maintained in the diesel engine sump. By periodic analysis, then, the licensee will be able to detect an increase in expansion seal leakage provided the total leakage still remains small. Large inleakage of jacket water can be identified by a change in oil sump level (increase) or jacket water level (decrease), or both. However, the volume of inleakage of jacket water that must occur before it can be detected by these means is more than can be tolerated for EDG operation and would cause catastrophic engine failure.

The licensee has proposed to utilize the EDGs in their present state without replacing any more expansion seals. The rationale for this position is as follows:

1. The expansion seals perform a secondary function. The primary seal between the jacket water system and the remainder of the diesel engine is the metal to metal fit between the cylinder liner and the cylinder block. During EDG operation, the cylinder liner heats up considerably and expands against the cylinder block, thereby enhancing the primary seal. Based on this, the licensee has taken the position that jacket water leakage into the lube oil during EDG operation is highly unlikely.

2. Results of the failure mechanics analysis of the expansion seals show that if the seals withstood 35 psi without leaking or rupturing, they can withstand indefinite service at normal jacket water pressure of about 14 psi without rupturing or leaking.

3. There are adequate means of identifying inleakage of jacket water; i.e., monthly lube oil analysis for small leaks, and jacket water and lube oil sump levels to indicate catastrophic introduction of jacket water.

Evaluation

The licensee's fracture mechanics analysis was submitted as an attachment to a letter dated May 11, 1988. The staff has reviewed both the letter and the attached analysis and has reached the following conclusions:

- The staff agrees with the results of the fracture mechanics analysis to the extent that the expansion seals that have been subject to MIC will not fail catastrophically. The staff, however, does not agree with the conclusion that existing cracks will not experience growth over time. To the contrary, the staff is of the opinion that existing cracks will grow and will eventually leak, albeit slow leakage.

- The staff agrees with the licensee that lube oil analysis on a monthly basis will be adequate to identify slow leakage from the expansion seals. The staff further agrees with the licensee that expansion seal failure, even catastrophic failure, during EDG operation will not result in any significant lube oil degradation.

- The staff concludes that the most likely time for expansion seal failure to cause significant problems is when the EDGs are in standby. In such a case, there are adequate provisions for detecting slow leakage (discussed earlier), and the jacket water and lube oil level alarms represent an acceptable means of detecting major leakage due to catastrophic expansion seal failure.

Conclusions

In order to have unacceptable consequences from expansion seal leakage, there would have to be a catastrophic failure of expansion seals on at least two EDGs immediately prior to a complete loss of offsite power. The staff does not consider this to be a likely event. Therefore, the staff concludes that the licensee's proposal to not replace expansion seals which have been pressure tested is acceptable. Further by letter dated December 9, 1988, the licensee committed to the following items:

- The lube oil for all EDGs at STP will be analyzed for water on a monthly basis.

- The EDG jacket water will be tested weekly to ensure the pH is maintained between 9.5 and 10.5 and corrosion inhibitor concentration is maintained. Additionally, biological samples will be taken every six months to ensure MIC is not present.

- The expansion seals for the Unit 2 EDGs were pressure tested to 35 psi to confirm the accuracy of the fracture mechanics analysis.

  Hydrostatic testing will be conducted at both the first and second refueling. Following the second pressure test, the staff will review the results and determine whether subsequent testing will be required.

- The Unit 1 EDG expansion seals will be pressure tested at the first refueling to ensure that the results of the fracture mechanics analysis are applicable.

- The lube oil sump and jacket water level alarms for all EDGs at STP will be maintained operable.

15 ACCIDENT ANALYSIS

15.4 <u>Reactivity and Power Distribution Anomalies</u>

15.4.6 Inadvertent Boron Dilution

In SSER 3, the staff evaluated additional information provided regarding this type of accident, including analyses for modes 3, 4, and 5, additional information on the analytical model, and a revised list of alarms and modifications for a dilution event. In its evaluation, the staff concluded that for modes 5B (cold shutdown--RCS not completely filled) and 6 (refueling), dilution would be prevented administratively by locking closed valves FCV-110B (in the normal reactor makeup water [RMW] line to charging pump suction), FCV-111B (in the RMW line to top of volume control tank), CV-021A (chemical mixing tank isolation valve), CV-0215 (emergency boration flush line isolation valve), and CV-0221 (alternate emergency boration isolation valve). Further the staff required that the valves be included in the technical specifications.

During the review of the combined TS, it was noted that closure verification of valve CV-0215 was not included in the technical specifications. By letter dated December 5, 1988, HL&P indicated that the valve had the handwheel removed and a mechanical locking device welded over the valve operator to prevent the valve from being manipulated. Also, a clamp used as part of the arrangement was locked in place with a padlock. Further HL&P committed to eliminate the flow path in question during the first refueling outage. By letter dated October 11, 1989, HL&P stated the dilution flow path was removed. The staff finds this to be acceptable.

15.8 Anticipated Transients Without Scram (ATWS)

15.8.1 ATWS Rule - ATWS Mitigation System

Introduction

On July 26, 1984, the Code of Federal Regulations (CFR) was amended to include Section 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants" (known as the ATWS Rule). The requirements of Section 10 CFR 50.62 apply to all commercial light-water-cooled nuclear power plants.

An ATWS is an anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the Reactor Trip System (RTS) to shut down the reactor. The ATWS Rule requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the probability of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

Paragraph (c)(1) of 10 CFR 50.62 specifies the basic ATWS mitigation system requirements for Westinghouse plants. Equipment, diverse from the RTS, is required to initiate the auxiliary feedwater (AFW) system and a turbine trip for ATWS events. In response to paragraph (c)(1), the Westinghouse Owners Group (WOG) developed a set of conceptual ATWS mitigating system actuation circuitry (AMSAC) designs generic to Westinghouse plants. WOG issued Westinghouse Topical Report WCAP-10858, "AMSAC Generic Design Package," which provided information on the various Westinghouse designs.

The staff reviewed WCAP-10858 and issued a safety evaluation of the subject topical report on July 7, 1986. In the safety evaluation, the staff concluded that the generic designs presented in WCAP-10858 adequately meet the requirements of 10 CFR 50.62. The approved version of the WCAP is labeled WCAP-10858-P-A.

During the course of the staff's review of the proposed AMSAC design, the WOG issued Addendum 1 to WCAP-10858-P-A by letter dated February 26, 1987. This Addendum changed the setpoint of the C-20 AMSAC permissive signal from 70% reactor power to 40% power. On August 3, 1987, the WOG issued Revision 1 to WCAP-10858-P-A which incorporated Addendum 1 changes and provided details on changes associated with a new variable timer and the C-20 time delay. For those plants selecting either the feedwater flow or the feedwater pump/valve status logic options, a variable delay timer is to be incorporated into the AMSAC actuation logics. The variable time delay will be inverse to reactor power and will aproximate the time that the steam generator takes to boil down to the low-low level setpoint upon a loss of main feedwater (MFW) from any given reactor power level between 40% and 100% power. The time delay on the C-20 permissive signal for all logics will be lengthened to incorporate the maximum time that the steam generator takes to boil down to the low-low level setpoint upon a loss of MFW with the reactor operating at 40% power. The staff considers the Revision 1 changes to be acceptable.

Paragraph (c)(6) of the ATWS Rule requires that detailed information to demonstrate compliance with the requirements be submitted to the NRC. In accordance with paragraph (c)(6) of the ATWS Rule, the licensee provided information by letters dated October 20 and November 13, 1986. The letters forwarded the detailed design description of the ATWS mitigating system actuation circuitry proposed for installation at the South Texas Project.

The staff held a conference call with the licensee on September 3, 1987, to discuss their AMSAC design. As a result of the conference call, the licensee responded to staff concerns by letter dated December 22, 1987. The response raised additional questions with respect to the proposed AMSAC design and a second conference call was held with the licensee on January 21, 1988. The licensee responded by letter dated April 29, 1988. A June 15, 1988, conference call clarified information provided by the submittals.

Review Criteria

The systems and equipment required by 10 CFR 50.62 do not have to meet all of the stringent requirements normally applied to safety-related equipment. However, the equipment required by the ATWS Rule should be of sufficient quality and reliability to perform its intended function while minimizing the potential for transients that may challenge the safety systems, e.g., inadvertent scrams.

The following review criteria were used to evaluate the licensee's submittals: (1) the ATWS Rule, 10 CFR 50.62, (2) "Consideration Regarding Systems and Equipment Criteria," published in the Federal Register, Volume 49, No. 124, dated June 26, 1984, (3) Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related," (4) Safety Evaluation of WCAP-10858, and (5) WCAP-10858-P-A, Revision 1.

Discussion and Evaluation

To determine that conditions indicative of an ATWS event are present, the licensee has elected to implement the WCAP-P-A AMSAC design associated with monitoring the MFW flow and activating the AMSAC when the MFW flow is below the low flow setpoint. Also, as addressed in the introduction section, the licensee will implement the new time delay (described in the introduction section) associated with the C-20 permissive consistent with the requirements of Revision 1 to the WCAP.

Many details and interfaces associated with the implementation of the final AMSAC design are of a plant-specific nature. In the safety evaluation of WCAP-10858, the staff identified 14 key elements that require resolution for each plant design. The following paragraphs provide a discussion on the licensee's compliance with respect to each of the plant-specific elements.

1. Diversity

   The plant design should include adequate diversity between the AMSAC equipment and the existing Reactor Protection System (RPS) equipment. Reasonable equipment diversity, to the extent practicable, is required to minimize the potential for common-cause failures.

The licensee will utilize MFW flow sensing instrumentation as input to AMSAC. The licensee has provided information to confirm that the AMSAC equipment will be diverse from equipment used in the RPS in the areas of design, equipment, and manufacturing. The AMSAC output signals will interface with existing AFW pump and turbine trip circuitry. This interface will use equipment that will be diverse from the RPS actuation equipment. This interface will be made through the use of relays which will be of a different make and manufacturer than those used in the RPS.

2. Logic Power Supplies

Logic power supplies need not be Class 1E, but must be capable of performing the required design functions upon a loss of offsite power. The logic power must come from a power source that is independent from the RPS power supplies.

The licensee has provided information verifying that the logic power supplies selected for the AMSAC logic circuits will provide the maximum available independence from the RPS power supplies. The AMSAC will be powered from nonsafety-related power supplies which will be independent of the RPS and capable of operating upon a loss of offsite power.

3. Safety-Related Interface

The implementation of the ATWS Rule shall be such that the existing RPS continues to meet all applicable safety criteria.

The proposed AMSAC design interfaces at its input with the existing Class 1E circuits of the turbine first-stage impulse pressure channels within the reactor protection system. At its output, the AMSAC will interface with the Class 1E circuits of the AFW pumps. Connections with the AFW control circuits will be made downstream of approved Class 1E isolation devices. The licensee has confirmed to the staff that the existing safety-related criteria that are in effect at STP will continue to be met subsequent to the implementation of ATWS (i.e., the RPS will continue to perform its safety functions within interference from AMSAC). Refer to Item 9 for further discussion.

4. Quality Assurance

The licensee is required to provide information regarding compliance with Generic Letter (GL) 85-06, "Quality Assurance for ATWS Equipment That Is Not Safety-Related."

The licensee has stated that the AMSAC equipment will be handled, stored, installed, calibrated, tested, operated, and maintained in accordance with approved plant procedures. These will be Quality-Related procedures consistent with the requirements of Generic Letter 85-06 for the nonsafety-related AMSAC equipment.

5. Maintenance Bypasses

Information showing how maintenance at power is accomplished should be provided. In addition, maintenance bypass indications should be incorporated into the continuous indication of bypass status in the control room.

The licensee provided information showing how maintenance will be accomplished at power. The staff was informed that maintenance at power will be performed by inhibiting the operation of AMSAC's output relays which will block the output signal and, thus, prevent it from reaching the final actuation devices. The continuous indication of bypass status will be provided in the main control room through the use of the ERFDADS computer. It is the staff's understanding that the licensee will conduct a human-factors review of the subject indication consistent with the plant's control room design process.

6. Operating Bypasses

The operating bypasses should be indicated continuously in the control room. Diversity and independence of the C-20 permissive signal should be addressed.

The licensee has provided information stating that the AMSAC operating bypass (C-20) will be used to enable the operators to bring the plant up in power during startup and to avoid spurious AMSAC actuations at power levels below 40% reactor power (the C-20 setpoint). Above 40% reactor power, the C-20 will automatically arm the AMSAC logics. The C-20 time delay on de-energization (activated on decrease in reactor power below 40%) value will be determined by the licensee. It is the staff's understanding that the C-20 time delay will be implemented consistent with Revision 1 to WCAP 10858-P-A to ensure that AMSAC will perform its required function in the event of a turbine trip (loss of load ATWS). The C-20 permissive signal will originate from existing first-stage turbine impulse chamber pressure sensors. This signal will be taken downstream from qualified isolators and, thus, will not interfere with the RPS. The operating bypass will be indicated continuously in the control room via the ERFDADS computer whenever it arms or enables the AMSAC. It is the staff's understanding that the licensee will conduct a human-factors review of the subject indication consistent with the plant's detailed control room design process.

7. Means for Bypasses

The means of bypassing shall be accomplished by using a permanently installed, human-factored, bypass switch or similar device. Disallowed methods for bypassing mentioned in the guidance should not be utilized.

The licensee stated that bypassing AMSAC during testing and maintenance will be accomplished with a bypass switch permanently installed on the QDPS RPV-N cabinet. The disallowed methods for bypassing, such as lifting leads, pulling fuses, blocking relays, or tripping breakers, are not to be used.

It is the staff's understanding that the licensee will conduct a human-factors review of the AMSAC bypass controls consistent with the plant's detailed control room design.

8. Manual Initiation

Manual initiation capability of the AMSAC function must be provided.

In the plant-specific submittal, the licensee discussed how manual turbine trip and AFW actuation are accomplished by the operator. The operator can use existing manual controls located in the control room to perform a turbine trip and to start AFW flow. Thus, no additional manual initiation capability is required as a result of installing the AMSAC equipment.

9. Electrical Independence From Existing Reactor Protection System

Independence is required from the sensor output to the final actuation device, at which point nonsafety-related circuits must be isolated from safety-related circuits by qualified Class 1E isolators.

The licensee discussed how electrical independence is to be achieved. The proposed AMSAC design requires isolation between the non-Class 1E AMSAC and the Class 1E input circuits associated with the turbine first stage impulse chamber pressure signals and the AMSAC output signals to the AFW system. The licensee stated that the Class 1E inputs to the AMSAC will be isolated from the AMSAC using Westinghouse 7300 Series isolation devices. The AMSAC output signal to the Class 1E AFWS circuits will be isolated using MDR isolation relays. The subject isolation devices are acceptable for use at STP as fully qualified isolators (i.e., satisfactorily tested in accordance with Appendix A of the WCAP Safety Evaluation). Also, subsequent to the AMSAC implementation, the entire RPS design will remain consistent with the electrical separation criteria established for the STP during original plant licensing.

10. Physical Separation From Existing Reactor Protection System

The implementation of the ATWS mitigating system must be such that the separation criteria applied to the existing RPS are not violated.

The licensee stated that the AMSAC circuitry will be physically separated from the RPS circuitry. The licensee has further stated that the cable routing will be independent of protection system cable routing and that the ATWS equipment cabinets will be located so that there will be no interaction with the protection system cabinets. The existing separation criteria (FSAR Section 8.3) associated with the RPS will not be compromised as a result of the AMSAC installation and implementation.

11. Environmental Qualification

The plant-specific submittal should address the environmental qualification of ATWS equipment for anticipated operational occurrences.

The licensee stated that AMSAC mitigation equipment will be located in areas of the plant that are considered to be a mild environment. The licensee also stated that the equipment will be environmentally qualified for anticipated operational occurrences that might occur associated with the respective equipment locations.

12. Testability at Power

Measures to test the ATWS mitigating system before installation, as well as periodicially, are to be established. Testing of the system may be performed with the system in the bypass mode. Testing from the input sensor through to the final actuation device should be performed with the plant shut down.

The licensee stated that a complete end-to-end test of the AMSAC system, including the AMSAC outputs through to the final actuation devices, will be performed during each refueling outage. With the plant at power, the system will be tested with the AMSAC output actuation devices bypassed. The testing capability consists of a series of overlapping tests. These tests will verify analog channel accuracy, setpoint (bistable trip) accuracy, coincidence logic operation, and operation and accuracy of all timers.

This bypass of the AMSAC output actuation devices will be accomplished through a permanently installed bypass switch which negates the need to lift leads, pull fuses, trip breakers, or physically block relays. Status outputs to the plant computer and main control board, indicating that a general warning condition exists for AMSAC, will be initiated when the system's outputs are bypassed. Plant procedures will be used to test the AMSAC circuitry and outputs. These procedures will ensure that AMSAC is returned to service when testing is complete.

It is the staff's understanding that the licensee will conduct a human-factors review of the controls and indications used for testing purposes that is consistent with the station's detailed control room design process.

13. Completion of Mitigative Action

The licensee is required to verify that (1) the protective action, once initiated, goes to completion and (2) the subsequent return to operation requires deliberate operator action.

The licensee responded that the system design will be such that AMSAC is consistent with the circuitry of the AFW and turbine trip control systems, as well as the blowdown and sampling systems. Once initiated, the design will ensure that the protective action goes to completion. Deliberate operator action will then be necessary to terminate AFW flow, clear the turbine trip signal, and re-open the turbine stop valves.

14. Technical Specifications

The plant specific submittal should address technical specification requirements for AMSAC.

The licensee responded that no technical specification action is proposed with respect to the AMSAC and that normal administrative procedures are sufficient to control AMSAC.

The equipment required by the ATWS Rule to reduce the risk associated with an ATWS event must be designed to perform its functions in a reliable manner. A method acceptable to the staff for demonstrating that the equipment satisfies the reliability requirements of the ATWS Rule is to provide limiting conditions for operation and surveillance requirements in the TS.

In its Interim Commission Policy Statement of Technical Specification Improvements for Nuclear Power Plants (52 FR 3788, February 6, 1987), the Commission established a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in TS. The staff is currently reviewing, for all plants, ATWS requirements to criteria in this Policy Statement to determine whether and to what extent TS are appropriate. The staff will provide guidance regarding the technical specification requirements for AMSAC at a later date.

Conclusion

The staff concludes, based on the above discussion that the AMSAC design proposed by the Houston Lighting & Power Company for the South Texas Project, Unit 1 is acceptable and is in compliance with the ATWS Rule, 10 CFR 50.62, paragraph (c)(1). In addition, HL&P in its letter of December 22, 1987 committed to have all modifications completed prior to startup from the first refueling outage. By letter dated October 11, 1989, HL&P informed the staff that all actions scheduled for the first refueling outage have been completed. The staff finds this acceptable. The staff will verify the successful completion of certain noted human-factors engineering reviews to which the licensee has committed during the NRC's post-implementation inspection. Until staff review is completed regarding the use of technical specifications for ATWS requirements, the licensee should continue with the scheduled installation and implementation (planned operation) of the ATWS design and provide testing utilizing administratively controlled procedures. The staff will provide guidance regarding the technical specification requirements for AMSAC at a later date.

16 TECHNICAL SPECIFICATIONS

During the development of the Combined TS, a discrepancy between the TS reviewed for Unit 1 and the SER and its supplements other than Section 16 was identified. Resolution of this issue follows.

° Monitoring of Component Cyclic or Transient Limits (Table 5.7-1)

The cyclic and transient conditions listed in the South Texas Units 1 and 2 Technical Specification Table 5.7-1 Component Cyclic or Transient Limits is much less comprehensive than the conditions listed in FSAR Table 3.9-8 Summary of Reactor Coolant System Design Transients. The purpose of Table 4.7-1 is to require the licensee to maintain the cyclic and transient limits of the reactor coolant system within the design bases in accordance with ASME Code requirements. If the numbers assigned to the transients listed in FSAR Table 3.9-8 are exceeded, the plant could eventually be operating under a condition that is beyond design bases. Specifying and tracking an incomplete list of transients, as would be the case if only those in Table 5.7-1 were tracked, would not provide adequate information. The licensee was requested to justify the deviation between the Technical Specification Table 5.7-1 and the limits specified in FSAR Table 3.9-8. During a meeting with the staff, the licensee committed to develop procedures to adequately track the transients listed in FSAR Table 3.9-8. The commitment was confirmed in HL&P letter of December 7, 1988.

Dated: October 19, 1989

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