UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 61 AND 50 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated May 27, 1993, as supplemented by letter dated April 18, 1994, Houston Lighting and Power Company (HL&P) proposed to amend the South Texas Project (STP) Units 1 and 2 Technical Specifications (TS) and Updated Final Safety Analysis Report (UFSAR) to upgrade the reload fuel assemblies to Westinghouse (W) VANTAGE 5 Hybrid (V5H) design. Currently, STP Units 1 and 2 utilize the Westinghouse 17x17XL standard (STD) fuel design for core reloads. This fuel is like W STD fuel except that it is longer (14 ft. long versus 12 ft. STD) to accommodate the longer STP core design. Fresh VANTAGE 5H fuel assemblies, manufactured to the STP 14 ft. length, will be used in each future reload until a full core loading of VANTAGE 5H fuel is achieved. The licensee also proposed to implement numerous safety analysis and operational margin improvements into the TS and UFSAR.

The first fuel loadings of VANTAGE 5H fuel are scheduled for South Texas Unit 1 Cycle 6 and South Texas Unit 2 Cycle 4. The safety analysis changes and associated setpoint changes will be implemented for both units during refueling outage 5 for Unit 1.

In addition to the proposed TS changes, HL&P submitted a safety evaluation report for the reload transition from the present XL STD fueled core to an all VANTAGE 5H fueled core. This report provided the results of the fuel, nuclear, thermal-hydraulic, and accident analyses which have been reviewed by the staff.

The April 18, 1994, letter revised the implementation date due to delays imposed by recently completed outages. The amendments will be fully implemented for both units upon completion of the fifth refueling outage for Unit 1.

2.0 EVALUATION

Background

The licensee has upgraded the fuel used in the South Texas Project units to the Westinghouse Vantage 5 Hybrid design in an effort to improve fuel economy and reduce the cobalt source term. In conjunction with the mechanical fuel upgrade, the licensee proposed the following changes which affect the safety analysis: an increased peaking factor allowance, a change to the RCS average temperature range, a revised thermal design procedure, a positive moderator coefficient, shutdown margin reduction, modified overtemperature and overpower delta T, 10 percent steam generator tube plugging, added tolerance for pressurizer and steam line safety relief valve drift, steamline break mass and energy release inside containment, increased fuel storage rack enrichment limit, and reduced auxiliary feedwater flow. By Amendment Nos. 51 (Unit 1); 40 (Unit 2) and Amendment Nos. 54 (Unit 1); 43 (Unit 2), the NRC had previously approved increases in the refueling water storage tank boron concentration, the accumulator boron concentration, and the boric acid storage tank volume to accommodate the change in fuel type.

The proposed amendment resolves the licensee's commitment to technical specification changes for Veritrak/Overtemperature delta T and the following Justifications for Continued Operation (JCO): JCO #920020 "Veritrak Transmitters," JCO #920698 "Containment System Response DBA", JCO #910393 "Pressurizer Safety Relief Valve Loop Seal Purge Time," and JCO #910049 "Steam Line Break Mass and Energy Release."

In addition to the changes to the technical specifications, the UFSAR and the Core Operating Limits Report (COLR) have been revised.

2.1 Fuel Mechanical Design

STP Units 1 and 2 are currently operating with Westinghouse 17x17 XL standard (STD) fuel. Beginning with Unit 1 Cycle 6 and Unit 2 Cycle 4, reload fuel will consist of the Westinghouse VANTAGE 5H fuel design eventually leading to an all VANTAGE 5H fueled core. The V5H fuel design is described in WCAP-10444-P-A, Addendum 2, which was approved for reference in a staff safety evaluation of November 1, 1988. Since then, V5H fuel has been approved for reload applications in numerous plants. The features of the VANTAGE 5H fuel design which differ from those of the current STP STD XL fuel design include the replacement of intermediate inconel structural grids with Zircaloy grids, and the use of integral fuel burnable absorbers (IFBA).

NRC Information Notice 93-82, "Recent Fuel and Core Performance Problems in Operating Reactors," pointed out that VANTAGE 5H fuel can be damaged by vibrational fretting wear caused by a flow condition adjacent to the core baffle. The fuel vendor, Westinghouse, proposed short-term and long-term corrective actions. The licensee informed the staff that the STP fuel design and core loading have adopted the Westinghouse recommendation of short-term corrective action to address the vibrational fretting wear problem. The staff considers that the licensee's corrective action is acceptable for STP.

The licensee analyzed stress, strain, rod internal pressure, fatigue, and rod bowing based on the approved methodologies for steady state and transient conditions. These analyses considered the longer fuel design of the STP core. The results showed that the VANTAGE 5H fuel performed satisfactorily. The staff considers these analyses adequate.

The licensee also analyzed the rod cluster control assemblies (RCCAs), control rod drive mechanisms (CRDMs), neutron source assemblies, burnable absorber assemblies, and thimble plug assemblies. The absorber materials used in the RCCAs are boron carbide pellets plus silver-indium-cadmium alloy. The burnable absorbers used are the Westinghouse designed wet annular burnable absorbers (WABAs). All the RCCAs and WABAs designs have been approved previously. Therefore, the staff concludes that the RCCAs, WABAs, and CRDMs are acceptable for STP.

Based on the approved mechanical methodologies, the staff concludes that the VANTAGE 5H fuel mechanical design for STP is acceptable.

2.2 Nuclear Design

The effects of the VANTAGE 5H fuel on the STP physics parameters as compared to the STD fuel are small and the STP spent fuel pool criticality analysis allows for the storage of VANTAGE 5H fuel assemblies. The nuclear design parameters characterizing the STP transition core have been computed by methods previously used and approved for Westinghouse reactors.

Included in the licensee's submittal is a proposal to increase the allowable fuel enrichment from 4.5 weight percent (w/o) uranium-235 to 5.0 w/o. Storage of spent fuel with the higher enrichment was discussed and approved in a staff safety evaluation of August 25, 1992 (Amendment Nos. 43 and 32). The current submittal provides additional discussion of new fuel racks and in-containment fuel storage racks. The acceptance criteria for criticality require the effective neutron multiplication factor, K_{eff}, in the fresh fuel storage rack to be less than or equal to 0.95 for fully flooded conditions or 0.98 under optimum moderation conditions, including uncertainties. For the in-containment fuel storage rack, K_{eff} must be maintained less than 0.95, including uncertainties, for all conditions.

The licensee's report shows that the acceptance criteria are met for STP fresh and in-containment fuel storage racks for the storage of all Westinghouse 17x17 fuel assemblies (including extra-length assemblies) with the following conditions and enrichment limits:

Fresh rack - Storage of fuel assemblies with nominal enrichments up to 5.0 w/o in any location. There are no requirements on position or IFBA for these assemblies.

In-Containment - Rack: Storage of fuel assemblies with nominal enrichments up to 4.5 w/o in any location. Fuel assemblies with enrichments above 4.5 w/o can also be stored, but each assembly must contain sufficient IFBA to satisfy the requirements shown in Figure 6 of Section 6 of the racks' criticality analysis included in the reference documents volume of the licensee's submittal.

The re-analysis of the reactivity effects of fuel storage in the fresh fuel and in-containment fuel storage racks was performed with the KENO Va Monte Carlo computer code with neutron cross sections generated by the AMPX code package from the 227 energy group ENDF/B-V library. Since the KENO Va code package does not have depletion capability, burnup analyses were performed with the two-dimensional transport theory code, PHOENIX. These codes are widely used for the analysis of fuel rack reactivity and burnup and have been benchmarked against results of numerous critical experiments. The staff concludes that the analysis methods used by the licensee are acceptable, and that the proposed storage rack provisions discussed above are acceptable.

Beginning with Cycle 6 of Unit 1, future cycles of operation for STP will use increased power peaking factors to increase nuclear design flexibility and allow loading patterns with reduced leakage which in turn will allow longer operating cycles without increasing vessel fluence. The maximum heat flux hot channel factor (F_Q) limit at rated thermal power (RTP) will increase from the current value of 2.50 for STD fuel to 2.7 for both fuels. The operational full power nuclear enthalpy rise hot channel factor ($F_{delta\ H}$) will increase from the current STD value of 1.46 to 1.49 for STD fuel and 1.55 for VANTAGE 5H fuel. The peaking factors assumed in the design and safety analyses are 2.7 F_Q, and 1.55 (STD) and 1.62 (V5H) $F_{delta\ H}$. The higher design values of $F_{delta\ H}$ account for analytical and surveillance uncertainties. The lower value for STD fuel is needed to comply with the local oxidation criterion in LOCA analyses. These increased limits on peaking factors continue to ensure that the design limits on peak local power density and minimum departure from nucleate boiling ratio (DNBR) are not exceeded during normal operation and anticipated operational occurrences (AOOs), and that the peak clad temperature will not exceed the emergency core cooling system (ECCS) acceptance criteria in the event of a LOCA, as discussed in Section 2.4.

The reduced thimble size of the VANTAGE 5H fuel design could affect the control rod scram time. The drop time is measured from the beginning of decay of stationary gripper coil voltage to dashpot entry. The effect of this increase on the STP safety analyses has been considered and it was determined that the 2.8 second rod drop time assumed in existing analyses remains bounding. The licensee will verify this conclusion in startup tests. The staff finds this acceptable.

Based on its review, the staff concludes that approved methods have been used and that the nuclear design parameters meet applicable criteria and are supported by design bases safety analyses discussed in Section 2.4 of this safety evaluation. Therefore, the proposed nuclear design and the analytical methods used are acceptable.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic analysis, DNB performance, and hydraulic compatibility during the transition from a mixed VANTAGE 5H-STD fueled core to an all VANTAGE 5H core incorporate the WRB-1 and W-3 DNB correlations, the revised thermal design procedure (RTDP), and an improved THINC IV modeling. Each of these has been reviewed and approved by the NRC. For the WRB-1 DNB correlation, the NRC has approved a 95/95 DNBR limit of 1.17 for the 17x17 STD fuel assemblies. To account for uncertainties associated with rod bow, a flow anomaly associated with reactor coolant system (RCS) and nuclear instrumentation system parameters, safety analysis DNBR limits of 1.43 and 1.38 were used for typical cells and thimble cells, respectively. The W-3 DNB correlation, with a 95/95 DNBR limit of 1.30, and the standard thermal design procedure (STDP) thermal-hydraulic methods are still used when conditions are outside of the range of the WRB-1 DNB correlation and of the RTDP. These correlations are used for both STD and V5H fuel designs because of the thermal and hydraulic compatibility of these fuel types as demonstrated in WCAP-10444-P-A. Also, the W-3 correlation with a 95/95 DNBR limit of 1.45 is used for steam line break analyses in the pressure range of 500 to 1000 psia since this range is below the range of the primary DNB correlations.

The licensee has indicated that a rod bow penalty of less than 1.0 percent is applicable to the 17x17 XL STD and V5H fuel. There is not a DNBR penalty associated with mixed cores for these fuels. The licensee indicated that the DNBR penalty related to the flow anomaly is about 3.6 percent, and would be accommodated by the margin between the design limit for the STD and V5H fuels and the analysis limit. The licensee has indicated that the margin is also intended to accommodate DNBR penalties that may occur in the future, and to provide flexibility in the design and operation of the plant.

The staff concludes that the rod bow, flow anomaly, and transition core penalties are adequately covered by the margin maintained between the design and safety limit DNBR values. Maintenance of adequate DNBR margin to cover DNBR penalties is confirmed by the licensee on a cycle-specific basis during the reload safety evaluation process.

The licensee's submittal included WCAP-11273, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems South Texas Project Units 1 and 2." This STP-specific report describes the application of the Westinghouse setpoint methodology which has been used in applications to other operating plants, including the V. C. Summer plant, which was referenced in the report. In Section 2.5 of this safety evaluation, we conclude that this setpoint methodology is also applicable to the South Texas units.

The thermal-hydraulic evaluation of STP with VANTAGE 5H fuel as well as the evaluation of VANTAGE 5H demonstration assemblies in various operating reactors have shown that 17x17 XL STD and VANTAGE 5H fuel assemblies are hydraulically compatible, and that sufficient DNBR margin exists in the safety limit DNBR to cover any rod bow and transition core penalties.

Approved methodologies were used and all thermal-hydraulic design criteria were satisfied. Therefore, the staff finds the thermal-hydraulic design of the STP transition STD/V5H and final VANTAGE 5H cores acceptable.

2.4 Transient and Accident Analyses

The impact on the plant safety analyses of the transition from Westinghouse STD fuel to Westinghouse VANTAGE 5H fuel as well as other changes which represent a departure from those currently used for STP has been reviewed by the licensee to determine which events need to be re-analyzed. The review was based on event-specific sensitivities, and a decision was made for each transient with regard to the need for a formal analysis as opposed to simply evaluating the impact of the subject features and assumptions. Events were reanalyzed in accordance with methods described in the Westinghouse reload methodology report, WCAP-9272-P-A.

A nominal core thermal power of 3800 MWt was assumed. The safety evaluations also assumed 10 percent steam generator tube plugging, and were performed at a thermal design flow of 95,400 gpm per loop, which conservatively bounds the licensing minimum measured flow of 98,075 gpm per loop. No one steam generator was assumed to exceed 10 percent tube plugging. The analyses also account for added tolerance for pressurizer safety valve setpoint drift and loop seal purge time, and a reduced steam-driven and motor-driven auxiliary feedwater pump surveillance flow requirement of 500 gpm.

The RTDP methodology discussed above was used to define the initial conditions for those re-analyzed accidents which have DNB as a limiting criterion, and are initiated at or near full power conditions to demonstrate that the DNB design basis is met. The other reanalyzed accidents used the standard thermal design procedure (STDP) to obtain initial conditions by adding the maximum steady-state errors to nominal values. The NRC requires a review of the temperature, pressure, power, and flow uncertainties used in the safety evaluations when using the RTDP. For STP, the uncertainties have been calculated based on plant procedures for instrument calibration, heat balance calculations, and RCS flow measurement.

The staff has reviewed the accidents which were re-analyzed or re-evaluated. These re-analyses applied methods which have been previously found acceptable by the staff. The results, which include transition core effects, show changes in the consequences of transients and accidents previously analyzed. However, the results remain within the required acceptance criteria. Specifically, for non-LOCA events, during normal operation and anticipated operational occurrences, there is at least a 95 percent probability at a 95 percent confidence level (95/95 probability/confidence) that DNB will not occur on the limiting fuel rod. During these operational modes, there is also a 95/95 probability/confidence that the peak kw/ft fuel rods will not exceed the melting temperature of UO_2, taken as 4900°F (unirradiated) and 4800°F at end of life. For these events, peak RCS pressure does not exceed the acceptance criterion of 110 percent of the 2500 psia design pressure.

The submitted discussion proposes that the STP design basis limiting RCS peak pressure criterion for locked rotor events be changed from the current basis of 110 percent of design pressure (2750 psi) to faulted stress limits (about 2900 psi). The locked rotor event analyses submitted in support of proposed TS changes acceptably meet the current locked rotor RCS pressure criterion of 110 percent of design pressure, and take credit for delayed loss-of-offsite power. This indicates to the staff that there is no need for the proposed change in design basis pressure criterion. Furthermore, the staff considers acceptance criteria for accident analyses to be generic positions which the staff has historically supported, and are not within the scope of this review. The licensee's submittal does not provide justification for a plant specific exception to the staff generic position. Consequently, the staff does not accept this plant specific proposal and continues to evaluate STP locked rotor event analyses by its current RCS pressure criterion.

The maximum average fuel pellet enthalpy was less than 225 cal/gm (unirradiated) and 200 cal/gm (irradiated) for all control rod ejection events, thus meeting the NRC criterion of less than 280 cal/gm.

The radiological consequences of those accidents reanalyzed to reflect an increase in fuel burnup to 60,000 MWD/MTU associated with the use of VANTAGE 5H fuel, and the other changes being implemented in STP, such as the increased peaking factors and increased BOC MTC, are not significantly changed by the current reference analyses.

The large break LOCA analysis for STP 1 and 2, applicable to a full core of VANTAGE 5 fuel assemblies, was performed to develop specific peaking factor limits. The large break LOCA analyses assumed that the reactor was running at 3876 MWt (102 percent of rated power) with a total peaking factor (F_Q) of 2.7, a hot channel enthalpy rise factor ($F_{delta-H}$) of 1.62 (1.55 for once-burned STD), RCS flow of 95,400 gpm per loop, and hot leg temperature (T_{hot}) of 625.6 °F. The approved Westinghouse 1981 ECCS evaluation model with BASH was used and a spectrum of cold leg breaks was analyzed. The worst case peak clad temperature (PCT) was 2177 °F for a double-ended cold leg guillotine (DECLG) break with a discharge coefficient (C_d) of 0.6. The analysis assumed both maximum containment safeguards (lowest containment pressure) and maximum ECCS safeguards ("no failure" single failure). The maximum local Zirconium/water reaction of 15.39 percent was calculated for a different case, assuming DECLG break with a C_d of 0.8 and minimum low pressure safety injection (failure of one safety injection train). This was explained by differences in the limiting assumptions (single failure, etc.) between the cases, the high peaking factors, and the long STP core, resulting in differing calculations of burst node location versus the PCT node. The calculated maximum core-wide Zirconium/water reaction rate was less than 1 percent for all cases analyzed.

Therefore, the results demonstrated that the PCT acceptance criterion of 2200°F as well as the criteria related to clad oxidation and maximum hydrogen generation contained in 10 CFR 50.46 continue to be met. In addition, the core remains amenable to cooling during and after the LOCA, and will be maintained in a shutdown condition with borated water with no credit for control rod insertion.

The small break (SB) LOCA analyses were performed with the approved Westinghouse ECCS small break evaluation model using the NOTRUMP and LOCTA-IV codes. The analysis assumed a full core of VANTAGE 5H fuel to determine PCT for a spectrum of cold leg breaks. The small break LOCA analyses make the same assumptions regarding plant condition (RCS flow, reactor power, peaking factors, etc.) as listed above for large break LOCA analyses, with the failure of an emergency power train which results in loss of one complete train of ECCS components (including 2 auxiliary feedwater trains) identified as the most limiting single failure. The minimum delivered flow available to the RCS is based on this single failure. The results demonstrate that the remaining ECCS with 2 auxiliary feedwater trains provide sufficient core cooling to meet the acceptance criteria limits of 10 CFR 50.46 for the limiting SBLOCA, a 1.5-inch break. The calculated PCT for this case is 1816°F, the maximum calculated local Zirconium/water reaction is 5.96 percent (also for this case), and the calculated core-wide Zirconium/water reaction is less than 1 percent. These results also demonstrate that SB LOCA events meet the performance requirements of 10 CFR 50.46(b), and are not limiting.

Because the methodologies used to perform the transient and accident analyses supporting the proposed changes are applicable and approved methodologies, with clarifications as discussed above, the analyses are acceptable.

2.5 Setpoint Methodology

The staff reviewed the Westinghouse methodology for calculating instrument loop uncertainties and instrument trip setpoint as presented in Westinghouse documents, WCAP-11273 "Westinghouse Setpoint Methodology for Protection Systems," and WCAP-13411 "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology." On a continual basis, under contracts from the licensee, Westinghouse performs all periodic calculations and analyses for the revised thermal design procedures (RTDP) using this methodology for both units of STP. In addition, Westinghouse was contracted by the licensee to revise setpoint-calculations for all protection systems of both units of STP by incorporating this methodology.

For determining monitoring instrumentation errors, Westinghouse has taken an approach that the uncertainties can be described with random, normal, and two-sided probability distributions, and the sum of both sides is equal to the range for the parameter being monitored. The individual instrument error components for a channel-uncertainty are combined using the square root of the sum of the squares of those groups of components which are statistically independent. Those errors that are dependent are combined arithmetically into independent groups, which are then systematically combined. Channel uncertainties and uncertainties in trip settings, indications, and computer readouts were computed addressing the following attributes as applicable. These attributes included: process measurement accuracy (PMA), primary element accuracy (PEA), sensor calibration accuracy (SCA), sensor measurement and test equipment accuracy (SMTE), sensor pressure effects (SPE), sensor drift (SD), rack calibration accuracy (RCA), rack measurement and test equipment accuracy (RMTE), rack temperature effects (RTE), rack drift (RD), readout device accuracy (RDOUT), computer isolator drift (ID), analog to digital conversion accuracy (A/D), controller accuracy (CA), and environmental effects (EA). Also considered were: biases, allowances for process variable overshoot and/or undershoot, thermal inertia, deadbands, compensation for excessive thermal drift in Veritrack transmitters, and various coefficients and constants used for certain process conditions (the environment, material conditions and for the geometrical configurations used for instrument connections to the process.) Setpoint margin was calculated using Margin = TA - CSA equation; where TA is Total allowance (safety analysis limit - nominal trip setpoint), and CSA is channel statistical error allowance (Total of component uncertainties). Channel component uncertainties were grouped in groups such as: process allowances, sensor allowances, and rack allowances.

The relationship between the error components and the total error for a instrument channel was computed using following equation.

$$CSA = EA + \{(PMA)^2 + (PEA)^2 + (SCA+SMTE+SD)^2 + (STE)^2 + (SPE)^2 + (RCA+RMTE+RCSA+RD)^2 + (RTE)^2\}^{1/2} \pm \text{ Bias if any}$$

Process allowances are PMA and PEA are both considered independent. PMA includes the non-instrument related effects such as neutron flux, calorimetric power error assumptions, fluid density changes, and temperature stratification assumptions. PEA accounts for errors due to metering devices, such as elbows, venturi, and orifices.

Sensor allowances are SCA, SMTE, SD, STE, and SPE. SCA, SMTE and SD are considered interactive, and STE and SPE are considered independent. The procedures used for calibration and for determining instrument drift compare the instrument output to its known input. Thus, unless "AS LEFT/AS FOUND" data is recorded and tracked for some significant length of time for each component, it is impossible to determine differences between calibration errors, and the drift when the sensor is checked during calibration.

Rack allowances are RCA, RMTE, RCSA, RD and RTE. RCA, RMTE, RCSA and RD are considered interactive, and RTE is independent. Therefore, unless "AS LEFT/AS FOUND" data is recorded and tracked for a period of time for each component, it is impossible to determine differences between calibration errors, and the drift when the rack instrumentation is checked during calibration.

The Westinghouse methodology also considered the following:

a. Synergistic effects of aging, exposure of components to environment such as temperature, humidity, background radiation for all applicable uncertainty attributes.

b. Except for process measurement accuracy, rack drift, and sensor drift, all uncertainties assumed were extremes of the ranges. Therefore, the results were more conservative than using two sigma values. Rack drift and sensor drift were based on a survey of reported plant LERs.

c. During the life of the plant, insulation resistance (IR) of cable(s), terminations, and junctions will be degraded continuously due to synergistic effects of normal environment and/or due to an accident environment, and will introduce error in instruments. In the case that the value of such error was less than 0.1 percent of the span, Westinghouse considered it negligible, and it was omitted. Where its value exceeded 0.1 percent of span, it was considered as an environmental error. Westinghouse confirmed that for quantifying the error introduced due to IR changes, simulated aged cables and related terminations were used during testing. HL&P is aware of effects of IR degradation, and will keep track of value(s) of IR by performing periodic testing. In case the error due to IR degradation is found to exceed 0.1 percent, the affected calculation would be revised to account for the error.

d. Westinghouse calculations were based on a 1:1 ratio between sensor accuracy and accuracy of measurement and test equipment (M&TE), and on a 4:1 ratio between M&TE accuracy and accuracy of electronics (i.e., rack equipment). Westinghouse stated that these accuracy ratios were given to them by HL&P. The staff assumes that HL&P will keep track of the M&TE accuracies, and will maintain them throughout the life of the plant, or revise the setpoint calculations as necessary to address new accuracies of the M&TE.

As a result of the re-evaluation of uncertainties in instrument channels of safety systems, changes were proposed to: values of total allowance(s), Z(s), allowable value(s), sensor error(s), trip setpoint(s); terms used in the OTDT equation including time constants and constants K_1, K_2, and K_6; values of q_t and q_b (percent rated thermal power in the top and bottom halves of the core respectively); value of P_a used in containment leakage rate tests; primary containment average temperature for LCO; and the minimum flow value for surveillance tests of the auxiliary feedwater pump. The proposed changes also include a revision to TS Bases 2.2.1 for the minimum value of DNBR during steady state, and the value of shutdown margin in Bases 3/4.1.1, "Boration Control."

Based upon our review, the staff finds the Westinghouse Methodology used for determining the instrument channel uncertainties, trip setpoints and setpoint margins at STP acceptable.

2.6 Revised Maximum Containment Pressure and Temperature Response

The licensee has performed maximum containment pressure and temperature response reevaluation with the following changes to the safety analysis of that reviewed in the staff's safety evaluation that supported the issuance of the operating licenses (NUREG-0781):

a. Reduced containment free volume,
b. Reduced containment initial temperature,
c. Mass and energy release changes due to fuel upgrade, and effect of T_{hot} reduction.

The licensee indicated that as part of a probabilistic risk assessment model development effort, a review of pertinent calculations identified a mathematical error in the containment free volume calculation. Due to this error, the original calculation overestimated the containment free volume. The reduced containment free volume is 3.38×10^6 ft^3, including a -0.85 percent margin of error, a reduction of 5.1 percent. The original free volume was calculated as 3.56×10^6 ft^3.

The licensee has proposed to change the technical specification limit for the initial containment temperature from 120°F to 110°F.

The effect of the reduced containment free volume, reduced containment initial temperature, the mass and energy releases for the new fuel, and the T_{hot} reduction are evaluated below. The effects of these changes on the containment maximum temperature, containment maximum and minimum pressures, containment subcompartment analysis, containment safety-related equipment, containment leakage, and hydrogen generation were considered.

2.6.1 Containment Maximum Temperature

The licensee indicated that the containment maximum temperature occurs due to a design basis main steam line break (MSLB). The original MSLB mass and energy releases were calculated using the Westinghouse MARVEL code, and the containment temperature and pressure using the Bechtel COPATTA code. For the V5H fuel upgrade effort, the MSLB mass and energy release rates were re-calculated using the updated Westinghouse LOFTRAN code, and containment temperature and pressure were re-calculated using the Brookhaven CONTEMPT4 code. The LOFTRAN and CONTEMPT4 codes have been used at other plants for the above analyses, and the staff has found the use of these codes acceptable.

For the fuel upgrade effort, the MSLB analyses were reanalyzed using a reduced containment free volume of 3.38×10^6 ft^3, revised MSLB mass and energy releases, reduced containment initial temperature of 110°F, and updated passive heat sinks. The maximum containment temperature results from the design basis double-ended main steam line break coincident with one main steam isolation valve (MSIV) single active failure. The re-analysis of the MSLB increases the peak containment temperature from 323°F to 327°F.

The licensee stated that the containment structure is designed for 286°F. The revised design basis MSLB predicts that the peak temperature remains above 286°F for the first 110 seconds of the transient. During this brief period the heat transfer coefficient is not sufficiently high to result in heating the containment structures to the vapor temperature (327°F), and the structure design temperature of 286°F will remain bounding. The licensee also indicated that the containment safety-related equipment is qualified to operate in an accident environment with pressure and temperature equal to 57 psig and 340°F. Since the containment structure and safety-related equipment design temperatures remain bounding, the staff finds the proposed change in peak containment temperature acceptable.

2.6.2 Containment Maximum Pressure

The licensee indicated that the mass and energy release analyses for the V5H fuel upgrade were performed to conservatively maximize the mass and energy release available to the containment following a LOCA. The licensee has applied a multiplier of 1.0025 associated with the V5H fuel to the LOCA mass and energy release rates at time zero to the end of the blowdown portion of the transient. For the post-blowdown phase, this penalty is not required, since the currently listed releases remain bounding.

The net effect of the T_{hot} reduction is to increase the LOCA blowdown phase mass flowrate (during the first 25 seconds) by 2 percent, and decrease the energy releases by 0.6 percent. For the post-blowdown phase, the LOCA mass and energy releases remain unchanged. The licensee stated that based on the current Westinghouse models it is expected that these changes will have negligible effect on the long-term pressure transient results, and therefore, the long-term LOCA mass and energy releases due to T_{hot} will remain bounded by the existing design basis.

The maximum calculated peak containment pressure results from the current case of mass and energy releases of a double-ended pump suction guillotine (DEPSG) loss-of-coolant accident with maximum safety injection and minimum containment heat removal systems in operation. The re-evaluation increases peak containment pressure from 37.5 to 41.2 psig. Based on its review, the staff finds the proposed change acceptable since the peak containment pressure of 41.2 psig, calculated with approved methods, remains bounded by the containment design pressure of 56.5 psig.

2.6.3 Containment Minimum Pressure

The licensee indicated that the calculation for containment minimum pressure is not affected by either the containment free volume reduction, or the mass and energy release change due to the fuel upgrade. However, using an initial temperature of 110°F instead of 120°F, changes the minimum pressure from -3.5 psig to -2.9 psig. Therefore, the containment minimum design pressure of -3.5 psig (11.2 psia) is still applicable, and remains bounding. The staff finds the proposed change acceptable as the containment minimum design pressure of -3.5 psig remains bounding.

2.6.4 Containment Subcompartment Analysis

The licensee has indicated that it has analyzed the effects of reduced containment volume, reduced containment initial temperature, and the short term mass and energy releases due to fuel upgrade for the containment subcompartment analysis such as the pressurizer subcompartment, radioactive pipe chase subcompartment, regenerative heat exchanger subcompartment, RHR system valve room subcompartment, and steam generator loop compartments. The results of the analyses show that the original subcompartment design differential pressure remains bounding, and that the negligible change in peak differential pressure does not significantly affect the design margins or impact the structure design calculations. Based on the above results, the staff finds the proposed change acceptable, since it will not affect the subcompartment designs or the equipment located in them.

2.6.5 Containment Leakage

The licensee indicated that the Unit 1 containment was tested at 40.0 psig, and met the leakage criterion of the technical specifications. The Unit 2 containment was tested at 44.6 psig, and the leakage rate was also below the acceptance criterion of the technical specifications. The licensee is proposing to increase the peak containment pressure from 37.5 to 41.2 psig while maintaining the same leak rate in the technical specifications. Since Appendix J to 10 CFR Part 50 requires the licensee to perform leak testing at the peak accident pressure, 41.2 psig, and the technical specifications require the containment leakage limit to be satisfied, the staff finds the higher containment pressure with the present containment leakage limit to be acceptable.

2.6.6 Containment Hydrogen Generation

The licensee indicated that the revised analysis with reduction in containment free volume, reduced containment initial temperature, and changes due to fuel upgrade demonstrates that the requirements listed in Standard Review Plan Section 6.2.5 (including 10 CFR 50.44 and 10 CFR 50.46) continue to be met. The staff has reviewed this information and finds the proposed change for fuel upgrade acceptable.

2.6.7 Safety Injection/Containment Spray Operation

The licensee indicated that it has evaluated the effects of the reduced containment volume on safety injection and containment spray pump operation, and that the results indicate that the pumps are capable of providing required flow rates under increased containment pressure conditions. The staff has reviewed this information and finds the proposed change for fuel upgrade acceptable.

2.7 Technical Specification Changes

The specific changes proposed for the STP Technical Specifications are evaluated below.

(1) Figure 2.1-1, Reactor Core Safety Limits

The figure was revised to reflect the change of the limiting safety limits on the combination of the reactor thermal power, pressurizer pressure, and the highest operating loop coolant temperature. The changes reflect the DNB margin gained through use of the VANTAGE 5H IFM grid feature, the use of the improved THINC IV code, the WRB-1 DNB correlation, and RTDP. The limits are also reflected in the revised LOCA analyses for STP. Therefore, the limits given in TS Figure 2.1-1 are acceptable.

The Bases for TS 2.1.1 was also revised to describe the new DNB design basis methodology, and are acceptable.

(2) Table 2.2-1, Overtemperature Delta T (OTdT) and Overpower Delta T (OPdT) Trip Setpoints

The implementation of VANTAGE 5H fuel, the use of the RTDP, and the inclusion of parameters as determined by the Westinghouse setpoint methodology whose application is described in WCAP-11273, Rev. 2, cause the DNB core limits to change. These core limit changes result in OTdT and OPdT reactor trip setpoint changes. These setpoint changes are reflected in the STP safety analyses, which resulted in acceptable consequences, and are acceptable.

(3) Figure 3.1.2a, Moderator Temperature Coefficient

The licensee has proposed that the moderator temperature coefficient (MTC) limit specified in the STP TS 3.1.1.2/Figure 3.1.2a be revised for future core designs to permit a positive MTC. Several related safety analyses included in the submittal assumed a MTC of +5 pcm/F. The locked rotor analysis assumed a more limiting MTC of +5 pcm/F for powers up to 70 percent of rated thermal power and a linear ramp value from 70 percent power to 0 pcm/F at 100 percent power. The licensee proposes to incorporate the ramped MTC function assumed in the locked rotor analysis into the STP TS. The licensee is not proposing to implement this MTC revision at this time and the COLR MTC value remains 0 pcm/F. The staff finds the TS limit change acceptable because it is supported by related analyses.

(4) Figures 3.1-1 and 3.1-2, Required Shutdown Margin

The licensee has proposed to reduce the shutdown margin specified in TS 3.1.1.1/Figure 3.1-1 (covering operating Modes 1-4) and 3.1.1.2/Figure 3.1-2 (Mode 5) from 1.75 delta-K/K to 1.3 percent delta-K/K. These changes are supported by the results of affected analyses: main steam system depressurization; steamline break; feedline break; boron dilution events; and, post-LOCA shutdown. The results of these analyses indicate that design and acceptance criteria will continue to be met assuming the reduced margin. The staff finds the proposed reduced shutdown margin is acceptable. TS Bases 3/4.1.1 and 3/4.1.2 have been revised to reflect the reduced shutdown margin, and are acceptable.

(5) TS 3.2.5, DNB Parameters and associated Bases

The DNB-related parameters (RCS T_{avg}, pressurizer pressure, and RCS flow) and flow measurement uncertainty specified in TS 3.2.5 will be modified. The revised measured RCS average temperature range is 582.3 to 593.0 °F. Although the numerical value for maximum RCS T_{avg} (598°F) has not changed, its TS meaning has changed from an indicated value to an analytical value, the difference accounted for by measurement uncertainties. The revised pressurizer pressure is greater than 2189 psig. The revised RCS flow is greater than or equal to 392,300 gpm accounting for a flow measurement uncertainty of 2.8 percent with 10 percent steam generator tube plugging.

The DNB parameter changes reflect the use of the RTDP and implementation of the WCAP-11273, Rev. 2 setpoint methodology, and are supported by analyses in the submittal, as discussed above. The values used in the RTDP and transient and accident analyses conservatively bound these values. The staff finds the proposed changes, are acceptable.

The Bases associated with the above TS changes will be revised to reflect the proposed changes and are acceptable.

(6) Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints

Various trip setpoints and allowances have been changed based on the results of a revised Westinghouse reactor protection system setpoint study. The setpoint study implements the TS revisions to optimize trip setpoints, within the bounds of the safety analysis limits. The TS revisions to total allowance are direct results of uncertainty modifications. Changes to "Z" values are a direct result of PMA modifications. Allowable values and setpoints were changed to accommodate the modification in overall channel statistical allowance. The staff approved the methodology used in the setpoint study, as discussed in Section 2.5, and concluded that the revised values are acceptable.

(7) TS 3/4.6.1, Primary Containment

The containment maximum pressure specified in containment leak rate limits (TS 3.6.1.2.a), containment air lock leak rate limits (TS 3.6.1.2.b and TS 3.6.1.3.b), containment leak rate testing criteria (Surveillance Requirements 4.6.1.1.c and 4.6.1.2), and containment air lock leak rate testing criteria (Surveillance Requirement 4.6.1.3.b) is increased from 37.5 psig to 41.2 psig. The increase is based on a re-calculation of containment free volume, mass and energy release changes due to the fuel upgrade, containment initial temperature reduction, and T_{hot} reduction. The staff reviewed the licensee's analysis of the change in containment pressure as discussed in Section 2.6.2, and found it acceptable. Therefore, the related technical specification changes are acceptable.

(8) TS 3.6.1.5, Primary Containment Average Air Temperature

The maximum average containment air temperature specified in TS 3.6.1.5 is decreased from 120 degrees F to 110 degrees F. The limit was changed due to a reanalysis of main steam line break mass and energy releases for the V5H fuel. As discussed in Section 2.6, the licensee evaluated the effects of decreased initial temperature on containment maximum temperature and pressure, containment minimum pressure, hydrogen generation, and containment subcompartment analysis, and found that the results are bounded by the design. The staff found the licensee's evaluation acceptable, and this TS change is acceptable.

(9) TS 4.7.1.2.1, Auxiliary Feedwater System and associated Bases

The minimum flow of the motor-driven and steam-driven auxiliary feedwater pumps was reduced for surveillance requirements to 500 gpm. The new minimum flow value is reflected in the STP safety analyses, which resulted in acceptable consequences, and is, therefore, acceptable.

(10) TS 5.2.1.g, Containment Net Free Volume

The containment net free volume specified in TS 5.2.1.g is decreased from 3.56×10^6 ft^3 to 3.38×10^6 ft^3 due to a correction of the original calculation. Containment pressure and temperature response were re-analyzed using the corrected containment volume, and were found to remain bounded by the design. This change is acceptable.

(11) TS 5.3, Reactor Core

TS 5.3 was modified to reflect an increase in the maximum nominal enrichment for fuel assemblies from 4.5 weight percent (w/o) uranium-235 to 5.0 w/o. This change affects the criticality analyses for fuel storage racks, and increases the radiological source terms. As discussed in Section 2.2, the storage racks were re-analyzed, and the staff concluded that the acceptance criteria for criticality is met. The impact of the increased maximum enrichment on the radiological consequences of accidents is a slight increase in the doses reported in the UFSAR. However, the doses remain well within the acceptance limits. As discussed in Section 2.4, the radiological consequences of accidents were reanalyzed to consider the increased discharge burnup and were found not to be significantly changed. Therefore, the change to a maximum nominal enrichment of 5.0 w/o is acceptable.

(12) TS 5.6.1, Fuel Storage

TS 5.6.1.7, TS 5.6.1.8, and Figure 5.6-7 are added to describe the new fuel storage and in-containment storage rack requirements. The staff has concluded that the acceptance criteria are met, and storage provisions are acceptable for STP fresh and in-containment fuel storage racks for the storage of all Westinghouse 17x17 fuel assemblies as discussed in Section 2.2. The proposed change is acceptable.

3.0 CONCLUSIONS

The staff has reviewed the reports submitted to support the proposed STP TS changes for VANTAGE 5 fuel and concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design, instrument uncertainty and setpoint methodologies, containment building response, and transient and accident analyses are acceptable. The proposed TS changes suitably reflect the necessary modifications for operation of STP, and are adequately justified.

The staff will continue to evaluate RCS pressure response to a locked rotor event calculated for STP according to the criterion of 110 percent design pressure. Therefore, the licensee's proposal to amend the STP design basis by using faulted stress limits as a criterion for evaluation of locked rotor events is not accepted in this safety evaluation. As noted above, the analyses supporting the proposed TS changes - including locked rotor analyses - acceptably meet current criteria. Therefore, our present finding that the proposed RCS pressure criterion change for locked rotor events is not accepted in this safety evaluation does not alter our conclusions regarding the acceptability of the proposed TS changes.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.33, an Environmental Assessment and finding of no significant impact was published in the Federal Register on May 25, 1994 (59 FR 27074). Accordingly, based upon the Environmental Assessment, the Commission has determined that issuance of these amendments will not have a significant effect on the quality of the human environment.

6.0 <u>CONCLUSION</u>

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Contributors: F. Orr, SRXB/NRR
S.V. Athavale, HICB/NRR
R. Goel, SCSB/NRR

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