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

May 27, 1993
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10CFR50.91
10CFR50.92

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498; 50-499
Proposed Licensing Amendment Concerning
Technical Specifications Based Upon Nuclear Fuel Upgrade,
Plant Safety Evaluation and
Revised Thermal Design Procedure

Pursuant to 10 CFR 50.90, Houston Lighting & Power (HL&P) hereby proposes to amend its Operating Licenses NPF-76 and NPF-80 by incorporating the attached proposed changes to the Technical Specifications and Updated Final Safety Analysis Report (UFSAR) for the South Texas Project Electric Generating Station (STP).

The purpose of this proposed license amendment is to implement revised UFSAR Chapter 15 accident analyses, increasing peaking factor limits, and include as inputs the results of a Revised Thermal Design Procedure analysis on operational setpoints. All of these subjects are described in the STP UFSAR and the Technical Specifications. These changes will accompany a transition to Westinghouse Vantage 5 Hybrid fuel type.

In an effort to improve fuel economy and reduce the cobalt source term, HL&P has upgraded fuel type to the Westinghouse Vantage 5 Hybrid design. The fuel design changes have been evaluated under the STP 10CFR50.59 review process and determined not to constitute an unreviewed safety question. Additionally, STP has performed an extensive reanalysis of the licensing basis, including non-LOCA and LOCA transients, rack criticality and containment building post-accident response. The results of the reanalysis are the bases of this submittal.

A001 Change: NRC PDR 1 w/out Prop
NSIC 1 w/out Prop
1/1
Ltr Encl

The implementation of the revised safety system setpoints would be best accomplished for Unit 1 during the 1994 refueling outage. Implementation for Unit 2 should be accomplished at the same time. If implementation on the two units cannot be accomplished at the same time, HL&P will revise this application accordingly.

HL&P has reviewed the attached proposed amendment pursuant to 10CFR50.92 and determined that it involves no significant hazards considerations. The basis for this determination is provided in the attachments. In addition, based on the information contained in this submittal and the NRC Final Environmental Assessment for STP Unit 1 and 2, HL&P has concluded that, pursuant to 10CFR51, there are no significant radiological or non-radiological impacts associated with the proposed action and the proposed license amendment will not have a significant effect on the quality of the environment.

The following issues will be resolved by the proposed amendment:

a) Technical Specifications changes for Veritrak/OTΔT as committed in HL&P letters ST-HL-AE-4277, dated December 09, 1992 and ST-HL-AE-4384, dated March 31, 1993

b) Justification for Continued Operation #920020, Rev. 3, "Veritrak Transmitters"

c) Justification for Continued Operation #920698, Rev. 0, "Containment System Response DBA"

d) Justification for Continued Operation #910393, Rev. 0, "Pressurizer Safety Relief Valve Loop Seal Purge Time"

e) Justification for Continued Operation #910049, Rev. 2, "Steam Line Break Mass and Energy Releases"

Among the enclosed items are:

1) WCAP-11273, Rev 2 "Westinghouse Setpoint Methodology for Protection Systems - South Texas Units 1 and 2" (Proprietary). [Non-proprietary version WCAP-11488, Rev. 1, provided]

2) WCAP-13441, "Revised Thermal Design Procedure Instruments Uncertainty Methodology for South Texas Units 1 and 2" (Proprietary). [Non-proprietary version WCAP-13442 provided]

Also enclosed are Westinghouse authorization letters, CAW-93-437 and CAW-93-438, accompanying affidavit, Proprietary Information Notice, and Copyright Notice, for the Revised Thermal Design Procedure and Setpoint Methodology, respectively.

As WCAP-11273 and WCAP-13441 contain information proprietary to Westinghouse Electric Corporation, they are supported by affidavits signed by Westinghouse, the owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavits should reference CAW-93-437 or CAW-93-438, respectively, and should be addressed to N. J. Liparulo, Manager of Nuclear Safety & Regulatory Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

The STP Nuclear Safety Review Board has reviewed and approved the proposed changes.

In accordance with 10CFR50.91(b), HL&P is providing the State of Texas with a copy of this proposed amendment. (With non-proprietary WCAP-11488, Rev. 1, and WCAP-13442)

Should you have any questions concerning this matter, please contact Mr. A. W. Harrison at (512) 972-7298 or me at (512) 972-7138.

S L Rosen

S. L. Rosen
Vice President,
Nuclear Engineering

HRP/sr

Attachment(s):

VOLUME 1:

ATTACHMENT 1: No Significant Hazards Evaluation for the Proposed License Amendment

ATTACHMENT 2: Marked-up Current South Texas Project Technical Specifications Reflecting the Proposed License Amendment

ATTACHMENT 3: Marked-up Unit 1 Cycle 6 Core Operating Limits Report Reflecting the Proposed License Amendment

ATTACHMENT 4: Marked-up Unit 2 Cycle 4 Core Operating Limits Report Reflecting the Proposed License Amendment

VOLUME 2:

ATTACHMENT 5: Marked-up Current South Texas Project UFSAR reflecting the Proposed License Amendment attachments (continued)

VOLUME 3:

Reference Material Provided for Information

Reference 1: Westinghouse Plant Safety Evaluation

Reference 2: WCAP-13441, Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology

Reference 3: Revised Containment Analysis

Reference 4: Criticality Analysis of the South Texas Units 1 & 2 Fresh & In-containment Fuel Storage Racks

Reference 5: WCAP-11273, Rev 2, Westinghouse Setpoint Methodology for Protection Systems

c:

Regional Administrator, Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

L. E. Kokajko
Project Manager
U.S. Nuclear Regulatory Commission
Washington, DC 20555

J. I. Tapia
Senior Resident Inspector
c/o U. S. Nuclear Regulatory
Commission
P. O. Box 910
Bay City, TX 77414

J. R. Newman, Esquire
Newman & Holtzinger, P.C., STE 1000
1615 L Street, N.W.
Washington, DC 20036

D. E. Ward/T. M. Puckett
Central Power and Light Company
P. O. Box 2121
Corpus Christi, TX 78403

J. C. Lanier/M. B. Lee
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

K. J. Fiedler/M. T. Hardt
City Public Service
P. O. Box 1771
San Antonio, TX 78296

Rufus S. Scott
Associate General Counsel
Houston Lighting & Power Company
P. O. Box 61867
Houston, TX 77208

Institute of Nuclear Power
Operations - Records Center
1100 Circle 75 Parkway
Atlanta, GA 30339-3064

Dr. Joseph M. Hendrie
50 Bellport Lane
Bellport, NY 11713

D. K. Lacker
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

U.S. Nuclear Regulatory Comm.
Attn: Document Control Desk
Washington, D.C. 20555

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter	)	
	)	
Houston Lighting & Power Company, et al.,	)	Docket Nos. 50-498
	)	50-499
	)	
South Texas Project Units 1 and 2	)	
	)	

AFFIDAVIT

S. L. Rosen being duly sworn, hereby deposes and says that he is Vice President, Nuclear Engineering, of Houston Lighting & Power Company; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the proposed amendment concerning the Nuclear Fuel Upgrade, Plant Safety Evaluation and Revised Thermal Design Procedure; is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge and belief.

S L Rosen

S. L. Rosen
Vice President,
Nuclear Engineering

STATE OF TEXAS)
)
)

Subscribed and sworn to before me, a Notary Public in and for The State of Texas this 27th day of May, 1993.

SAVANNA S. FRANKLIN
Notary Public, State of Texas
My Commission Expires 7-16-94

Savanna S. Franklin

Notary Public in and for the
State of Texas

TSC\93-145.001

NRC SUBMITTAL PACKAGE
for the
Proposed Revision Technical Specifications and UFSAR to
Technical Specifications Based Plant Safety Evaluation and
Revised Thermal Design Procedure (Nuclear Fuel Upgrade)

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

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Safety Evaluation
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ATTACHMENTS/ENCLOSURES

VOLUME 1:

ATTACHMENT 1:	No Significant Hazards Evaluation for the Proposed Fuel Upgrade
ATTACHMENT 2:	Marked-up Current South Texas Project Technical Specifications Reflecting the Proposed Fuel Upgrade
ATTACHMENT 3:	Marked-up Unit 1 Cycle 6 Core Operating Limits Report Reflecting the Proposed Fuel Upgrade (Typical)
ATTACHMENT 4:	Marked-up Unit 2 Cycle 4 Core Operating Limits Report Reflecting the Proposed Fuel Upgrade

VOLUME 2:

ATTACHMENT 5:	Marked-up Current South Texas Project UFSAR reflecting the Proposed Fuel Upgrade

VOLUME 3:

Reference Material Provided for Information

Reference 1:	Westinghouse Plant Safety Evaluation
Reference 2:	Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology
Reference 3:	Revised Containment Analysis
Reference 4:	Criticality Analysis of the South Texas Units 1 & 2 Fresh & In-containment Fuel Storage Racks
Reference 5:	Westinghouse Setpoint Methodology for Protection Systems

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

1.0 Summary

This proposed licensing change is intended to upgrade the fuel used in the South Texas Project to Westinghouse's VANTAGE 5 Hybrid (V5H) design and implement numerous safety analysis and operational margin improvements into the South Texas Project Technical Specifications and Updated Final Safety Analysis Report (UFSAR). It also addresses a number of open issues including Veritrak transmitter uncertainty, post-accident cable insulation resistance, and revised containment volume. It also provides the basis for the T_{hot} reduction program.

Houston Lighting and Power has contracted with Westinghouse to upgrade the fuel used in the South Texas Project to V5H design, and the first reload is planned for Unit 2 Cycle 4 and Unit 1 Cycle 6. The mechanical fuel changes for V5H fuel, and the associated UFSAR changes, to be used in Unit 2 Cycle 4 and Unit 1 Cycle 6 have been approved by Houston Lighting & Power via the internal 10CFR50.59 review process and found not to involve an Unreviewed Safety Question, though descriptions of these changes are included in this submittal package for completeness. It is planned that the safety analysis changes and associated setpoint changes will be implemented on Unit 1 at the beginning of cycle 6 (Spring, 1994) and during mid-cycle 4 on Unit 2.

This submittal provides the Safety Evaluation, No Significant Hazards Evaluation , and the proposed Tech Spec/COLR/UFSAR changes. Analyses used in the development of this package are provided in References 1, 2, 3, 4, and 5.

The effects of the proposed changes do not pose a significant increase in hazards.

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

2.0 Purpose/Scope

The purpose of this proposed licensing change is to upgrade the fuel used in the South Texas Project to Westinghouse's V5H design and implement numerous safety analysis and operational margin improvements into the South Texas Project Technical Specifications and UFSAR.

The following scope of changes is proposed.

- o VANTAGE 5H fuel with zircalloy mid-grids and Integral Fuel Burnable Absorbers (IFBA)
- o Increased Peaking Factor Allowance
- o RCS Average Temperature Range (allow vessel average temperature of 593°F to 582.3°F)
- o Revised Thermal Design Procedure (RTDP)
- o Positive Moderator Temperature Coefficient (PMTC)
- o Shutdown Margin Reduction from 1.75%ΔK/K to 1.3%ΔK/K
- o Modified Overtemperature and Overpower ΔT
- o 10% Steam Generator Tube Plugging
- o Added Tolerance for Pressurizer Safety Relief Valve Drift
- o Added Tolerance for Steam line Safety Relief Valve Drift
- o Steamline Break Mass and Energy Release Inside Containment
- o Increased Fuel Storage Rack Enrichment Limit
- o Reduced Auxiliary Feedwater Flow

Previously submitted revisions to the Technical Specifications (Reference 15):

- o RWST boron concentration increase - the boron concentration maintained in the RWST will increase to 2800 ppm (min) to 3000 ppm (max).
- o Accumulator boron concentration increase - the boron concentration in the accumulators will increase to 2700 ppm (min) to 3000 ppm (max).
- o Boric acid storage tank volume increase - the volume maintained in the boric acid storage tank will be increased from 2900 gallons to 3200 gallons.

In addition, the Veritrak instrumentation issue is addressed as is the incorporation of the revised containment free volume. Numerous editorial changes have been included to improve clarity.

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

3.0 Description of Changes

There are five major modifications proposed to the South Texas Project licensing bases:

1. Mechanical Fuel Upgrade to VANTAGE 5H;
2. Upgraded Safety Analysis;
3. Increased Maximum Allowable Fuel Enrichment;
4. Revised Reactor Containment Building P/T Response; and,
5. Revised Instrumentation Setpoints.

The V5H fuel analysis, safety analysis and documentation detail has been provided by the Plant Safety Evaluation (Reference 1). The PSE document will be referenced within this Safety Evaluation to provide the necessary details of the V5H upgrade. The PSE also provides much of the basis for the revised Technical Specification and UFSAR sections. Analyses related to items 3-5 are provided in the supplied References.

The required changes are noted below and the marked-up sections of the Technical Specifications (Attachment 2), COLR (Attachments 3 & 4) and the Updated Final Safety Analysis Report (Attachment 5) are provided.

3.1 Mechanical Fuel Upgrade to Vantage 5H

The specific features of VANTAGE 5H which represent a change from the current STP fuel are:

- o Zircaloy grids - The Inconel structural grids used in standard fuel are replaced by Zircaloy grids in VANTAGE 5H (except for the top and bottom grids which remain Inconel).

- o Integral Fuel Burnable Absorbers (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 pellets. IFBAs provide peaking factor and moderator temperature coefficient control.

The mechanical changes were adopted under the 10CFR50.59 review process (Ref. 6) and were found not to involve an Unreviewed Safety Question. No changes to Technical Specifications or Core Operating Limits Report (COLR) are necessary for the mechanical design changes. Changes to the UFSAR resulting from the 50.59 review are included in Attachment 5 for completeness.

3.2 Upgraded Safety Analyses

To accommodate the new safety analysis, the following changes to the Technical Specifications are proposed:

Specification 2.1/Figure 2.1-1:

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

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.

Table 2.2-1:
Revised to reflect the change of setpoints for the RTDP and the new setpoint study, primarily affected are OTΔT and OPΔT, and associated constants described in the Notes to the table.

Specification 3.1.1.1/Figure 3.1-1:
Revised to require a 1.3%$\Delta\rho$ ramped shutdown margin in Modes 1, 2, 3, and 4.

Specification 3.1.1.2/Figure 3.1-2:
Revised to require a 1.3%$\Delta\rho$ ramped shutdown margin in Mode 5.

Specification 3.1.1.3/Figure 3.1-2a:
Revised to permit a positive moderator temperature coefficient below 100% rated thermal power.

Specification 3/4.2.5:
Changed to reflect the revised pressurizer pressure, and RCS minimum measured flow rate. Also, removed RCS flow test below 75% and removed calibration standard for Special Test Equipment.

Table 3.3-4:
Revised Engineered Safety Features Actuation System setpoints to reflect RTDP and new Setpoint Study.

Specification 4.7.1.2.1:
Changed to reflect the revised minimum flow of the motor-driven and steam-driven auxiliary feedwater pumps for surveillance requirements to 500 gpm/pump.

Bases for Specification 2.1.1:
Revised to describe the proposed DNB design basis methodology.

Bases for Specification 3.2.2 and 3.2.3, Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor:
Modified to use the revised reactor coolant system flow rate and the revised DNBR value used in the safety analyses and the revised DNBR design limit.

Bases for Specification 3/4.2.5, DNB Parameters:
Modified to use the revised pressurizer pressure, temperature and reactor coolant system flow rate.

Bases for Specification 3/4.7.1.2, Auxiliary Feedwater System:
Modified to reflect the revised flow rate from each auxiliary feedwater pump to 500 gpm.

Bases for Specification 3/4.7.1.2, Auxiliary Feedwater System:
Modified to a flow rate from each auxiliary feedwater pump of 500 gpm and pressure of 1363 psig.

Corresponding to the above changes, the COLR is revised as described below.

The $F_{\Delta H}^{RTP}$ limit of 1.46 remains for Standard fuel and is revised to 1.55 for V5H fuel in Section 2.5.1.

3.3 Increased Maximum Allowable Fuel Enrichment

The following changes are proposed to incorporate the increased maximum fuel enrichment allowance.

Specification 5.3, Reactor Core:
Modified to reflect a higher maximum fuel enrichment of 5.0 weight percent uranium 235.

Specification 5.6.1
A description of new fuel storage and in-containment storage rack requirements is added. Also, the word "nominal" is inserted before the IFBA B^{10} loading in part 5.6.1.2. This was inadvertantly omitted from the previous submittal for this Spec (Reference 13).

3.4 Revised Reactor Containment Building Volume

Previously, HL&P had submitted to the USNRC a proposed licensing change to reflect a revised containment building volume (Reference 30). Since the uncertainties for the mass/energy releases for the containment design are affected by the revised safety analyses, the affected containment response analyses were revised and the previous submittal has been retracted. The following proposed Tech Spec changes incorporate the effects of the revised mass/energy releases and the revised containment volume.

Specification 3/4.6.1 Primary Containment
Except as noted below, change 37.5 psig to 41.2 psig throughout.

Surveillance Requirement 4.6.1.2.g
Delete the 41.25 psig in line five.

Surveillance Requirement 4.6.1.3.a
Replace "of 37.5 psig;" with "not less than Pa;".

Limiting Conditions . . . 3.6.1.5 decrease temperatures to 110°F.

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

Bases for Specification 3/4.6
Same change as first identified under 3/4.6.1

Insert "(41.2 psig)" following "Pa" in 3/4.6.1.2.

Insert "(Pa)" following the first occurrence of "37.5 psig" in paragraph two of 3/4.6.1.4 and in the first paragraph of 3/4.6.1.6

Section 5.0 Design Features
5.2.1.g. Net free volume is $3.38x10^6$ - $3.41x10^6$ ft^3.

3.5 Revised Instrumentation Setpoints

Changes to the Technical Specifications and COLR's due to the instrumentation setpoints are inherently related to changes in these documents due to the revised safety analysis. Therefore, changes due to the revised setpoints are discussed in Section 3.2, above.

3.6 Related Licensing Submittals

These proposed changes are based on the following related submittals by Westinghouse:

1. WCAP-12909-P, Westinghouse ECCS Evaluation Model: Revised Large Break LOCA Power Distribution Methodology," May 1991. (Reference 9)

2. Trich, S.R., "Methodology Clarifications to WCAP-12909-P, ET-NRC-91-3633, November 21, 1991. (Reference 27)

3. WCAP-10484, Addendum 1 (Proprietary), "Spacer Grid Heat Transfer Effects During Reflood," D.L. Shimeck, December 1992. (Reference 10)

The containment building pressure/temperature analyses incorporate the initial containment building temperature assumptions which were submitted for NRC approval in Reference 28. Therefore, the analyses contained in this submittal are predicated on NRC approval of the changes in assumptions proposed in Reference 28.

Safety Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

4.0 Safety Evaluation

This section will discuss the impact of the proposed changes on the design and licensing basis of the plant.

The VANTAGE 5H fuel analysis, safety analysis and documentation detail have been provided in the supplied Plant Safety Evaluation (PSE) (Reference 1). The PSE and other References will be cited within this Safety Evaluation to provide necessary details.

The safety evaluation will be presented in the following sections:

4.1 Mechanical Fuel Upgrade to VANTAGE 5H
4.2 Upgraded Safety Analysis
4.3 Increase in the Maximum Allowable Fuel Enrichment
4.4 Revised Reactor Containment Building P/T Response
4.5 Revised Instrumentation Setpoints
4.6 Miscellaneous Changes to Support Plant Operations

The current licensing basis for the South Texas Project (STP) is reflected in the UFSAR. The changes discussed in Section 3.0 represent changes to the plant as described in the UFSAR, changes in the methodology employed in certain areas of the reference safety analysis and changes to the input and assumptions used in the reference safety analysis.

4.1 Mechanical Fuel Upgrade to VANTAGE 5H

The specific features of VANTAGE 5H which represent a change from standard fuel are:

- o Zircaloy grids - The Inconel structural grids used in standard fuel are replaced by Zircaloy grids in VANTAGE 5H (except for the top and bottom grids which remain Inconel).

- o Integral Fuel Burnable Absorbers (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO2 pellets. IFBAs provide peaking factor and moderator temperature coefficient control.

These features, and the associated UFSAR changes, were reviewed for use at STP via the internal 10CFR50.59 review process for use in Unit 2 Cycle 4 and Unit 1 Cycle 6 (Reference 6). These topics are included in this evaluation for completeness.

Reconstitutable Top Nozzle (RTN) and Debris Filter Bottom Nozzle (DFBN) are features that have been used extensively in Westinghouse designs. Analyses and tests have been performed that confirm that hydraulic compatibility of these particular designs to existing designs so that these components do not impact any

parameters important to the safety analysis. These nozzle designs have been used previously at the South Texas Project units.

Additional information on these features can be found in Section 2.0 of the PSE (Ref. 1) and in the 50.59/USQE (Ref. 6).

4.1.1 Neutron Fluence on the Reactor Vessel

The implementation of VANTAGE 5H fuel is expected to have no adverse impact on the RT_{PTS} values and heatup and cooldown operating limits calculated for South Texas reactor vessel based on the assumptions presented by the fluence assessments. The fluence in the beltline region of the reactor vessel is not expected to increase. Therefore, the current RT_{PTS} and heatup and cooldown limits will be conservative.

4.1.2 Foreign Object Evaluation

Previously, a safety evaluation was performed (References 11 and 12) assessing the impact of South Texas Unit 1 operation with approximately 194 grams (0.43 lbs) of stainless steel screen material (originating from full flow filters utilized during hot functional testing) within the primary system. The screen material is judged to be either in the form of a ball (0.66 inch diameter) or as individual pieces of 0.038 inch diameter wire in 3/8 inch lengths. The purpose of this evaluation is to confirm that the new fuel design does not impact the previous conclusion of Foreign Object Evaluation (References 11 and 12).

Prior to Cycle 6 reload with VANTAGE 5H fuel assemblies, particles of filter debris entrained in the flow through the core larger than 0.25 inch in diameter would become lodged at a fuel assembly bottom nozzle. Therefore, if the screen was in the form of a ball (0.25 inch diameter or greater), the debris would have been trapped by a fuel assembly bottom nozzle and would not have entered into the fuel rod array. The smaller individual pieces of filter debris can pass through the bottom nozzle of a fuel assembly, enter into the fuel rod array, and either become trapped in the fuel assembly grids or pass entirely through the assembly. Any of the debris that would enter the Auxiliary Fluid Systems was not expected to have an adverse affect on their ability to perform their required safety-related functions.

Although none of the debris is known to have been removed from the system, various visual inspections have not identified its presence. It is judged that a significant portion of the 0.4 lbs of filter debris have been either removed from the reactor coolant system (since the original fuel assemblies have been replaced) or the debris have traveled to stagnant flow areas in the reactor coolant or auxiliary systems. It is not anticipated that any remaining filter debris within the reactor coolant system would become entrained in the flow and enter into the VANTAGE 5H fuel assemblies resulting in flow blockage in the free span of the fuel rod array (during all plant conditions). Previous investigations have shown that debris are capable of causing fuel clad breach due to fuel rod fretting wear from debris trapped at fuel assembly grid locations. There has been no observed increase in primary coolant activity suggestive of fuel damage due to rod fretting wear from debris trapped

at fuel assembly grid locations at South Texas Unit 1. Similarly, none would be expected with the VANTAGE 5H fuel upgrade assemblies. For the full flow filter debris which may remain elsewhere within the South Texas Unit 1 primary system, the conclusions of foreign object safety evaluation remain valid, and the presence of the filter debris need not be reanalyzed for the fuel upgrade.

4.2 Upgraded Safety Analysis

4.2.1 Associated Fuel/Upgrade Related Changes

The following are changes which affect the safety analysis.

- o VANTAGE 5H fuel with zircalloy mid-grids and Integral Fuel Burnable Absorbers (IFBA)
- o Increased Peaking Factor Allowance
- o RCS Average Temperature Range (allow vessel average temperature of 593°F to 582.3°F)
- o Revised Thermal Design Procedure (RTDP)
- o Positive Moderator Temperature Coefficient (PMTC)
- o Shutdown Margin Reduction from 1.75%ΔK/K to 1.3%ΔK/K
- o Modified Overtemperature and Overpower ΔT
- o 10% Steam Generator Tube Plugging
- o Added Tolerance for Pressurizer Safety Relief Valve Drift and Loop Seal Purge Time
- o Added Tolerance for Steam line Safety Relief Valve Drift
- o Steamline Break Mass and Energy Release Inside Containment
- o Increased Fuel Storage Rack Enrichment Limit
- o Reduced Auxiliary Feedwater Flow

Previously submitted revisions to the Technical Specifications (Reference 15):

- o RWST boron concentration increase - the boron concentration maintained in the RWST will increase to 2800 ppm (min) to 3000 ppm (max).
- o Accumulator boron concentration increase - the boron concentration in the accumulators will increase to 2700 ppm (min) to 3000 ppm (max).
- o Boric acid storage tank volume increase - the volume maintained in the boric acid storage tank will be increased from 2900 gallons to 3200 gallons.

A detailed discussion on the effects of changes can be found in Section 5 of the PSE (Ref. 1).

The safety analyses and evaluations performed demonstrated the acceptability of all of the above changes. Transition core effects were also addressed in the analysis.

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A number of upgrades have been incorporated into the non-LOCA safety analyses presented in Chapter 15 of the South Texas Project Units 1 and 2 UFSAR in addition to required assumption changes to support the VANTAGE 5H fuel upgrade. The features have various effects on the non-LOCA safety analysis. Therefore, each will be identified separately.

4.2.1.1 VANTAGE 5H fuel with Zircaloy Mid-grids and Integral Fuel Burnable Absorbers (IFBA)

Currently the South Texas plants have Westinghouse 17x17 XL Standard fuel. The VANTAGE 5H fuel has the same rod outer diameter but has zircaloy mid-grids. The only direct effect on the non-LOCA safety analyses coming from use of this fuel type is the potential for a change in the rod drop time from gripper release to dashpot entry. For South Texas, the current licensed rod drop time of 2.8 seconds has been shown to be sufficiently conservative so as to remain bounding even with V5H fuel and zircaloy mid-grids. Therefore, there is no need for a safety analysis rod drop time assumption change.

There may be some indirect effects of the VANTAGE 5H fuel and the use of IFBAs in the core design which can not be separated from the types of changes which typically occur at the time of a fuel reload design. Any indirect impact is incorporated into all of the non-LOCA safety analysis via the assumed physics parameters and verified at the time of the reload safety evaluation. There is no impact to the LOCA analyses.

4.2.1.2 Increased Peaking Factor Allowance

The peaking factors justified by the non-LOCA safety analyses include an $F_{\Delta H}$ of 1.62 and an F_Q of 2.7. The peaking factors are explicitly assumed in the generation of some non-LOCA safety analysis input. A value for $F_{\Delta H}$ is assumed in the calculation of the core thermal limits which in turn are used to generate the Overtemperature ΔT and Overpower ΔT reactor trip setpoints. Also, this parameter is directly used in the calculation of input to the Partial Loss of Flow (UFSAR Section 15.3.1), Complete Loss of Flow (UFSAR Section 15.3.2) and Locked Rotor (UFSAR Sections 15.3.3 and 15.3.4) analyses. F_Q is used explicitly to calculate input for the Rod Ejection (UFSAR Section 15.4.8) analysis.

There may be some indirect effects of the increased peaking factors which can not be separated from the types of changes which are typically seen at the time of a fuel design. Any indirect impact is incorporated into all of the non-LOCA safety analysis via the assumed physics parameters and verified at the time of the reload safety evaluation.

The revised safety analysis predicts that once burned Standard fuel would have a zirc/oxide thickness over the 17% limit at the higher proposed peaking factors of F_Q=2.7 and $F_{\Delta H}$=1.62, and this would reduce the margin to safety. Therefore, the $F_{\Delta H}$ is reduced to 1.55 for the Standard fuel to obtain acceptable results. F_Q remains at 2.7 for both fuel types. STP will, therefore, carry separate limits (presented in the COLR) for $F_{\Delta H}$: one for Standard fuel and one for VANTAGE 5H fuel.

4.2.1.3 RCS Average Temperature Range

A range of RCS Average Temperatures were incorporated into the safety analyses. This range includes a vessel average temperature from 582.3 to 593.0°F at full power. This will give South Texas the flexibility to choose the RCS full power temperature at the beginning of the fuel cycle design and implement the programmed value in that cycle without making further changes to the licensing basis safety analyses.

Table 4.2-1 indicates the vessel average temperature assumption used in the various non-LOCA accident analyses. In some cases, it was necessary to analyze the event at the high and low end of the temperature range to determine a limiting condition. For others, existing sensitivities were cited as justification for choosing one temperature or the other. Events which primarily examine the margin to DNB were done assuming a vessel average temperature of 593.0°F because high temperatures are more conservative for DNB calculations.

The results of all of the non-LOCA safety analyses were acceptable with respect to the event-specific criteria when analyzed at the limiting temperature condition. Therefore, any RCS vessel average temperature between 593.0 and 582.3°F will be acceptable so long as each fuel design considers the chosen operating temperature for the cycle. The LOCA analyses show that the limiting conditions remain with the higher RCS average temperature of 593°F.

4.2.1.4 Revised Thermal Design Procedure

Currently, the thermal design procedure used in the justification of the DNB design basis for South Texas is referred to as Standard Thermal Design Procedure (STDP). With the fuel upgrade, the Revised Thermal Design Procedure (RTDP) methods for calculating the DNB design basis are being used for selected transients (Reference 29). The uncertainty on the initial conditions for non-LOCA transients are handled differently depending on whether RTDP or STDP is used in the determination of the safety analysis limit DNBR.

For events which focus primarily on DNB, the following applies. If STDP is used, then instrument uncertainties are applied to the initial condition assumptions for power, pressure, flow and temperature in the conservative direction with respect to DNBR calculations. An example is the analysis of the Startup of an Inactive Loop (UFSAR Section 15.4.4). While this event is primarily concerned with ensuring that the DNB design basis is met, the initial reactor coolant flow is below the range for which RTDP uncertainties have been defined. Therefore, Standard Thermal Design Procedure is used for this event. If RTDP is used, only instrument biases are applied to the transient initial conditions, in the conservative direction with respect to DNBR calculations. The instrument uncertainties are instead statistically combined during the calculation of the safety analysis limit DNBR. RTDP is not used in analyses at zero power because the power levels reached during the transient are insufficient to apply the DNB correlations used in conjunction with RTDP. Further, the uncertainties on power and RCS average temperature used in the development of the safety analysis limit DNBR are not appropriate at zero power.

For events which focus on criteria other than DNB such as pressurizer overfill or saturation in the hot legs, the instrument uncertainties for pressure, power, temperature and flow were applied to the initial conditions as is currently done in the South Texas UFSAR (i.e. RTDP method was not applied).

DNBR limits are revised to reflect the reallocation of uncertainties. See Table 4.2-1 for a separation of RTDP and STDP transients.

The analyses completed for the upgrade justify the following instrument uncertainties:

Power:	±2% Calorimetric error
Temperature:	±6.25°F [includes a bias of 0.25°F] (See Note 1)
Pressure:	±46 psi [includes a bias of 13 psi] (See Note 2)

Thermal Design Flow: 381,600 gpm
Minimum Measured Flow: 392,300 gpm (See Note 3)

Notes:

1) DNB margin has been allocated to cover an additional 0.55°F bias on temperature

2) The Locked Rotor Pressure transient used an uncertainty of 46 psi. The Feedline Break analysis used an uncertainty of 50 psi. All other analyses used a pressurizer pressure uncertainty of +63 psi.

3) The Minimum Measured Flow requirement includes a measurement uncertainty of 2.7% plus an additional 0.1% for the NRC mandated venturi fouling allowance. Total: 2.8%. The calibration requirements for the Special Test Equipment have been removed from the Tech Specs since their specifications are mandated by the RTDP document and will be controlled administratively. Also, the <75% precision flow test has been deleted since its requirements have been determined to be met by the 12 hour flow surveillance.

4.2.1.5 Positive Moderator Temperature Coefficient (PMTC)

A Positive moderator temperature coefficient (PMTC) specification of +5 pcm/F from 0% rated thermal power (RTP) to 70% RTP and a linear ramp from +5 pcm/F at 70% RTP to 0 pcm/F at 100% RTP is proposed.

The flexibility to use PMTC in future core designs may be needed to improve fuel economy and design flexibility. Table 4.2-1 shows those non-LOCA analyses in which a positive MTC was assumed. Generally, a PMTC is assumed in transients which result in a heatup of the primary system prior to reactor trip. A PMTC accentuates the heatup by adding positive reactivity. Generic sensitivities and PMTC safety evaluations for other Westinghouse PWRs were used as justification to determine in which analyses a PMTC should be assumed.

For conservatism and to bound the transients at all operating conditions, the PMTC was assumed up to full power. The Locked Rotor (UFSAR Section 15.3.3) pressure transient is an exception. In order to obtain acceptable results, it was necessary to run the analysis at full power assuming a 0 MTC. An additional case was conservatively examined at 90% RTP to justify the PMTC of +5 pcm/°F. The Locked Rotor rods in DNB case was also examined at a full power PMTC of +5 pcm/°F and found to have acceptable results.

All analyses demonstrated that the presence of a PMTC did not cause a violation of the event-specific safety analysis criteria. Therefore, it is acceptable to incorporate this change into future core designs, as needed. At the current time, STP has elected not to incorporate PMTC and this is reflected in the COLR.

4.2.1.6 Shutdown Margin Reduction from 1.75Δk/k to 1.3Δk/k

Table 4.2-1 shows those transients in which shutdown margin is assumed as an explicit input to the non-LOCA analyses. The results of each of these transients met the event-specific criteria therefore, the reduced shutdown margin is acceptable. The post-LOCA shutdown analysis also shows the reduction in shutdown margin to be acceptable.

4.2.1.7 Modified Overtemperature and Overpower ΔT

The Overtemperature ΔT setpoints were revised because the core thermal limits were changed to reflect the use of RTDP. The Overpower ΔT K_6 was changed to reflect the Veritrak component only. The setpoints were determined using the same methods as currently used in the UFSAR. Refer to Table 4.2-2 for those events in which the safety analysis predicted that Overtemperature ΔT actuated for reactor protection. A full power steamline break analysis was completed to ensure that the Overpower ΔT setpoints remained acceptable. The event-specific criteria were met for each of the analyses in Table 4.2-2, therefore, the revised setpoints are acceptable.

4.2.1.8 10% Steam Generator Tube Plugging

The analyses completed for the fuel upgrade considered the most limiting steam generator tube plugging condition. For cooldown events, it is generally more conservative to assume 0% steam generator tube plugging so that the heat transfer area is maximized and the cooldown is accentuated. Conversely, in an event which results in a heatup of the primary system, typically, the highest level of steam generator tube plugging is assumed so that the heat transfer area is minimized and the heatup accentuated. It should be noted that it is expected that the higher level of tube plugging will not cause the reactor coolant flow to decrease below current design values. Therefore, in the analyses which model 10% steam generator tube plugging, it was not necessary to reduce the reactor coolant flow assumption.

Table 4.2-1 identifies the transients which assumed 10% steam generator tube plugging and those which assumed 0%. As with the previous assumptions, all transients met the event-specific criteria, therefore, with

respect to the non-LOCA and LOCA accident analyses, it is acceptable to plug up to 10% of the steam generator tubes.

4.2.1.9 Added Tolerance for Pressurizer Safety Relief Valve Drift and Loop Seal Purge Time

The non-LOCA safety analysis justified a pressurizer safety valve tolerance of $\pm$1% (ASME code setting) and $\pm$1% for drift during operation.

WCAP-12910 (Reference 23) examines the effect of a loop seal on the relief characteristics of the pressurizer safety valves. The pressurizer safety valves, when set with a loop seal as opposed to steam, will have three effects which should be considered in the safety analysis. In addition to the ASME code tolerance, for STP, a shift is assumed to be 1% of the set pressure to account for drift. Also, the loop seal must be purged from the valve before primary relief can occur. A pressurizer loop seal purge time of 1.12 seconds was assumed in the non-LOCA accident analysis.

The non-LOCA analyses considered the effect of the loop seal and the increased tolerance where appropriate. The transients which primarily examine peak RCS pressure include the Loss of Load/Turbine Trip (UFSAR 15.2.2/15.2.3) event and the Locked Rotor (UFSAR 15.3.3) event. In these events, the effects of the loop seal and the increased tolerance were explicitly modeled. The results of these analyses and the evaluations of the remaining non-LOCA transients demonstrate that the pressurizer safety valves will provide sufficient relief even when a loop seal and an analytical tolerance totalling $\pm$2% is assumed.

4.2.1.10 Added Tolerance for Steamline Safety Relief Valve Drift

Secondary pressure must remain below 110% of the steam generator shell design pressure during non-LOCA transients. Steamline safety valves ensure that this limit is met during a non-LOCA transient. Currently, the tolerance on the steam generator safety valves is $\pm$1% of the set pressure (ASME code tolerance only). In the analyses for the V5H fuel upgrade, an additional 2% was added for drift with an additional 3% for accumulation. The safety analyses took credit for the staggered pressure setpoints currently in the Technical Specifications. In previous analyses, the most limiting setpoint was assumed for all valves.

This steamline safety valve model was included in the analysis of the Loss of Normal Feedwater\Loss of Offsite Power analysis as well as in the Feedwater Line Break transient. The effects of reduced Auxiliary Feedwater requirements was evaluated at the higher relief valve settings and found to be acceptable.

4.2.1.11 Steamline Break Mass and Energy Release Inside Containment

In addition to the re-analysis of the Chapter 15 non-LOCA transients, the Steamline Break Mass and Energy releases for use in a containment pressure and temperature analysis were calculated. VANTAGE 5H fuel and

the associated plant changes were included in those calculations. The containment response calculations are presented in Section 4.4

4.2.2 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

Table 5.1.1-1 of the PSE lists the principal power rating values assumed in analyses performed for this report. For most accidents analyzed to demonstrate that the DNB design basis is met, the Revised Thermal Design Procedure (RTDP) is employed in defining the initial conditions. The other accidents obtain initial conditions by adding the maximum steady-state uncertainties to rated values (this procedure is commonly known as the Standard Thermal Design Procedure or STDP).

Accidents employing RTDP assume minimum measured flow (MMF) while accidents analyzed with STDP assume thermal design flow (TDF). Furthermore, the initial pressure, temperature and power is assumed to be at the nominal values.

4.2.2.1 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor coolant system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. Depending upon event specific characteristics, conservatism may dictate the use of either large or small reactivity coefficient values. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis.

4.2.2.2 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the accident analyses, the critical parameter is the time from the start of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. Calculations have demonstrated that even with the reduced thimble size associated with V5H fuel, the rod drop time of 2.8 seconds used in previous accident analyses for South Texas remains a conservative assumption. This will be verified during startup testing.

PSE Section 5.1.1.3 provides a discussion of the reactivity characteristics of the RCCA's assumed in the accident analyses.

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4.2.2.3 Trip Setpoints and Delays Assumed in Accident Analyses

Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in PSE Table 5.1.1-4.

The OTΔT and OPΔT trip setpoints were determined to bound the transition cores as well as a full core of VANTAGE 5H fuel. New axial offset limits that bound the various cores were employed to determine a new F(ΔI) penalty.

The difference between the limiting trip setpoint assumed in the analysis and the nominal trip setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant technical specifications.

4.2.2.4 Pressurizer Water Level

STP has a nominal pressurizer level program which ramps from 25% span at no-load conditions to 60% span at full power. For the Vantage 5H fuel upgrade, this level program will remain appropriate for operating conditions when the Reactor Coolant System hot full power average temperature is 593.0°F. For the reduced T_{avg} operation the level program will change depending on the particular fuel cycle conditions. For the coldest RCS T_{avg}, 582.3°F, the pressurizer level program will ramp linearly from 32.9% at no-load conditions to 47% at full power.

4.2.3 Non-LOCA Accidents

This section addresses the impact on the UFSAR Chapter 15 Non-LOCA accident analyses of the transition from Westinghouse 17x17XL STD fuel to Westinghouse VANTAGE 5H fuel. In most cases, the basic methodologies applied to the non-LOCA analyses presented for VANTAGE 5H fuel in this report are the same as those currently documented in the South Texas Project Units 1 and 2 UFSAR for standard fuel.

The limiting single failures have not changed from those assumed in the current South Texas Project Units 1 and 2 UFSAR for standard fuel.

The non-LOCA methodology is discussed in detail in PSE section 5.1.1.6. The methodology confirms that, if a core configuration is bounded by existing safety analyses, then the applicable safety criteria are satisfied. The methodology systematically identifies parameter changes on a cycle-by-cycle basis which may violate existing safety analysis assumptions and identifies the transients which require evaluation. This methodology is applicable to the evaluation of VANTAGE 5H transition and full cores.

For each of the potentially impacted non-LOCA transients, consideration was given to effects of the VANTAGE 5H fuel design features and modified safety analysis assumptions discussed in Section 5.1 of the PSE. As

dictated by event specific sensitivities, a decision was made for each transient with regard to the need for formal analysis, as opposed to simply evaluating the impact of the subject features and assumptions.

PSE Table 5.1.2-1 documents for each of the potentially impacted non-LOCA transients whether an analysis was performed or an evaluation was sufficient to assess the impact of the VANTAGE 5H transition.

4.2.3.1 Increase in Heat Removal by the Secondary System

A number of postulated events that could result in increased heat removal from the RCS by the secondary system have been identified and the appropriate limiting cases are presented in Section 15.1 of the UFSAR. The analyses and evaluations for these events, necessary to support the introduction of VANTAGE 5H fuel into the South Texas Units, are discussed below, and in PSE Section 5.1.3.

4.2.3.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

As described in UFSAR Section 15.1.1, the cooldown of the RCS resulting from reductions in feedwater temperature produce an increase in core power in the presence of a negative moderator temperature coefficient of reactivity. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The high neutron flux, overtemperature ΔT (OTΔT), and overpower ΔT (OPΔT) reactor trips prevent any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than the limit value.

The net effect on the RCS from the reduced feedwater temperature is similar to that from increased secondary steam flow, i.e. the reactor reaches a new equilibrium condition at a power level corresponding to the new steam generator ΔT. A decrease in normal feedwater temperature is classified as an ANS Condition II event, a fault of moderate frequency.

The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. The method of analysis is discussed in PSE Section 5.1.3.1.2 and the Results are discussed in PSE Section 5.1.3.1.3.

The decrease in feedwater temperature transient is less severe than the increase in secondary steam flow event. Based on results presented in PSE Section 5.1.3.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met and the conclusions of the UFSAR remain valid.

4.2.3.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

As described in UFSAR Section 15.1.2, the addition of excessive feedwater will produce a decrease in reactor coolant temperature which, in the presence of a negative moderator temperature coefficient of reactivity, causes an increase in core power. Such transients are attenuated by the thermal capacity of the secondary

plant and of the RCS. The high neutron flux, OPΔT and OTΔT reactor trips prevent any power increase which could lead to a DNBR less than the safety analysis value. Continuous addition of excessive feedwater would ultimately be limited by the steam generator high-high level trip which closes all feedwater control and isolation valves, trips the main feedwater pumps and trips the turbine. The increase in normal feedwater flow incident is classified as an ANS Condition II event, a fault of moderate frequency.

The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. The method of analysis is discussed in PSE Section 5.1.3.2.2 and the Results are discussed in PSE Section 5.1.3.2.3.

The results of the analyses show that the predicted DNBR for an excessive feedwater addition at power remains above the limit value (see PSE Section 4.2) so that the DNBR design basis is met. Similarly, the results demonstrate that neither the reactor coolant system nor the secondary system are overpressurized during this event. It was also determined that the reactivity insertion for the RCCA bank withdrawal from subcritical event (PSE Section 5.1.6.1) is larger than that for an excessive feedwater addition at zero power. Therefore, the results of the RCCA bank withdrawal from subcritical event bound those for the zero power excessive feedwater addition event. For all the cases of excessive feedwater addition considered, the conclusions of the UFSAR remain valid.

4.2.3.1.3 Excessive Increase in Secondary Steam Flow

As described in UFSAR Section 15.1.3, an excessive load increase incident is a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control. The excessive load increase transient is considered to be an ANS Condition II event, i.e. a fault of moderate frequency.

The reactor control system is designed to accommodate a 10 percent step load increase and a 5 percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. Steam flow increases greater than 10 percent are discussed in PSE Sections 5.1.3.4 and 5.1.3.5.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

- OTΔT
- OPΔT
- Power range high neutron flux
- Low pressurizer pressure

The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. The method of analysis is discussed in PSE Section 5.1.3.3.2 and the Results are discussed in PSE Section 5.1.3.3.3.

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The analysis results show that with a ten percent step load increase, the DNBR remains above the limit value (see PSE Section 4.2), thereby ensuring that the DNB design basis is met. Also, neither the primary nor secondary system pressures exceed 110% of the design values. The results of the analysis also demonstrate that for the modeled transient, the plant reaches a stabilized condition following the load increase without a reactor trip occurring. Therefore the conclusions of the UFSAR remain valid for this event.

4.2.3.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

As described in UFSAR Section 15.1.4, the most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of the largest of any single steam dump, relief, or safety valve. The main steam system depressurization results in an initial steam flow increase which decreases during the accident as the steam pressure falls. The energy removal produces a cooldown and depressurization of the RCS. In the presence of a negative moderator temperature coefficient, the cooldown then results in a positive reactivity insertion.

Accidental depressurization of the main steam system is classified as an ANS Condition II event, i.e. a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. The event is analyzed to demonstrate that the DNB design basis is satisfied assuming hot shutdown conditions with a stuck RCCA, offsite power available, and a single failure in the engineered safety features system.

The primary plant safety features that provide protection against an accidental depressurization of the main steam system due to the opening of a steam generator safety or relief valve are as follows:

1. Safety Injection System actuation from low pressurizer pressure or low steamline pressure signals

2. Reactor trip from high neutron flux, overpower ΔT, or in conjunction with the safety injection signal.

3. Redundant isolation of the main feedwater lines to minimize the additional cooldown that would otherwise occur from continued main feedwater flow.

4. Shutting of the fast-acting steam line stop valves

This list summarizes material presented in greater detail in UFSAR Section 15.1.4.1.

The method of analysis is discussed in PSE Section 5.1.3.4.2 and the Results are discussed in PSE Section 5.1.3.4.3.

The analyses show that the DNB design basis is met for the accidental depressurization of the main steam system event. Additionally, it is noted that this event is less limiting than the rupture of a main steam line presented in PSE Section 5.1.3.5.

4.2.3.1.5 Steam System Piping Failure

As described in UFSAR Section 15.1.5, the steam release arising from a main steam line rupture produces an initial increase in steam flow followed by a gradual decrease during the accident as the steam pressure falls. The energy removal produces a cooldown and depressurization of the RCS. In the presence of a negative moderator temperature coefficient the cooldown then results in a positive reactivity insertion. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core could become critical and return to power. A post-trip return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors induced by assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shutdown by the boric acid injection delivered by the safety injection system. A major steam line rupture is classified as an ANS Condition IV event, i.e. a limiting fault. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.4. As noted in UFSAR Section 15.1.5.1, although DNB and possible clad perforation may be acceptable following a main steam line rupture, the analysis does, in fact, show that the DNB design basis is met for any such rupture even assuming the most reactive RCCA stuck in its fully withdrawn position.

The primary plant safety features that provide protection against the consequences of a main steam line rupture are as follows:

1. Safety Injection System actuation from low pressurizer pressure, low steamline pressure, or Hi-1 containment pressure signals.

2. Reactor trip from high neutron flux, overpower ΔT, or in conjunction with the safety injection signal.

3. Redundant isolation of the main feedwater lines to minimize the additional cooldown that would otherwise occur from continued main feedwater flow.

4. Shutting of the fast-acting steam line stop valves

This list summarizes material presented in greater detail in UFSAR Section 15.1.5.1. For steam line ruptures downstream of the isolation valves, closure of all valves would completely terminate the blowdown. The limiting single failure is loss of a single train of ESF. All MSIVs are assumed to close. For any break, in any

location, no more than one steam generator would blow down even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10 of the UFSAR.

Steam flow is measured by monitoring dynamic head inside the steam pipes. Nozzles that are of considerably smaller diameter than the main steam pipe are located in the steam generators and serve to limit the maximum steam flow for any break at any location.

The method of analysis is discussed in PSE Section 5.1.3.5.2 and the Results are discussed in PSE Section 5.1.3.5.3.

The analyses show that for both with and without offsite power available, the DNB design basis continues to be met for the rupture of a main steam line. Therefore, the conclusions of the UFSAR for this event remain valid.

4.2.3.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents have been postulated which result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system. These events are discussed in this section and in PSE Section 5.1.4.

4.2.3.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

As stated in the UFSAR, this event is not applicable to South Texas Project Units 1 and 2.

4.2.3.2.2 Loss of External Electrical Load

As discussed in UFSAR Section 15.2.2, this transient is bounded by the analysis for the turbine trip event. The turbine trip event is described in PSE Section 5.1.4.3.

4.2.3.2.3 Turbine Trip

As described in UFSAR Section 15.2.3, for a turbine trip event, the reactor would be tripped directly (unless below approximately 50 percent power) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (typically 0.1 second) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Among the turbine trip initiation signals noted in the UFSAR are generator trip, low condenser vacuum, turbine overspeed and manual trip.

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure, with a resultant increase in primary system temperature and pressure. Due to a more rapid loss of steam flow caused by faster valve closure, the turbine trip event produces a slightly more severe transient than the loss of electrical load event.

The turbine trip event is classified as an ANS Condition II event, a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. The specific acceptance criteria applied to this event are that the DNB design basis be satisfied, while also assuring that the neither the primary nor secondary system pressure limits are exceeded.

The primary plant safety features that provide protection against damage to the RCS or the steam system in the event of a turbine trip event are as follows:

1. High pressurizer pressure reactor trip
2. Overtemperature ΔT reactor trip
3. Overpower ΔT reactor trip
4. Opening of the pressurizer safety valves
5. Opening of the steam generator safety valves
6. Low-low steam generator water level reactor trip

The method of analysis is discussed in PSE Section 5.1.4.3.2 and the Results are discussed in PSE Section 5.1.4.3.3.

Results of the analyses show that the plant design is such that a turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within 110% of the design pressure. The integrity of the core is also maintained since the DNBR remains above the limit value (see PSE Section 4.2). Thus the conclusions presented in the UFSAR remain valid for the turbine trip event.

4.2.3.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves, loss of condenser vacuum, and other events resulting in a turbine trip are all bounded by the analysis Section 4.2.3.2.3, "Turbine Trip", and PSE Section 5.1.4.5. Additional information for these event categories is provided in UFSAR Sections 15.2.4 and 15.2.5.

4.2.3.2.5 Loss of Condenser Vacuum and Other Events Resulting in a Turbine Trip

See discussion in Section 4.2.3.2.3 and PSE Section 5.1.4.3 .

4.2.3.2.6 Loss of Non-Emergency AC Power to the Plant Auxiliaries

A complete loss of non-emergency AC power to the plant auxiliaries accident is described in Section 15.2.6 of the UFSAR. This event may result in the loss of all power to the station auxiliaries; i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

Upon the loss of power to the reactor coolant pumps, the RCS flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. A loss of non-emergency AC power to the station auxiliaries is classified as an ANS Condition II event, a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. This is a long-term heat removal event which is specifically analyzed to determine if the auxiliary feedwater (AFW) system is capable of removing the stored and residual heat, thereby preventing either RCS overpressurization or water relief from the pressurizer (assuring that the core remains covered).

As discussed in UFSAR Section 15.2.6.1, a loss of AC power to the station auxiliaries as described above could also result in a loss of normal feedwater if the condensate pumps lose power to operate. A loss of normal feedwater is the most limiting Condition II event in the decrease in secondary heat removal category and, in PSE Section 5.1.4.7, is analyzed for a case which includes the loss of AC power. Since these results of PSE Section 5.1.4.7 are bounding, just as in the UFSAR, no detailed analytical results are presented here for the loss of non-emergency AC power to the station auxiliaries.

4.2.3.2.7 Loss of Normal Feedwater Flow

The loss of normal feedwater flow accident is described in Section 15.2.7 of the UFSAR. This event, which may arise from pump failures, valve malfunctions, or the loss of AC power, results in a reduction in the capability of the secondary system to remove the heat generated in the core.

If the reactor were not tripped during this accident, core damage could possibly occur from the sudden loss of heat sink. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur. A significant loss of water from the RCS could conceivably lead to core damage. It should be noted that for this event, the plant is tripped well before the steam generator heat transfer capability is reduced and the primary system conditions never approach a violation of the DNB design basis.

The loss of normal feedwater event is classified as an ANS Condition II event, a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. This is a long-term heat removal event which is specifically analyzed to determine if the auxiliary feedwater (AFW) system is capable of removing the stored and residual heat, thereby preventing either RCS overpressurization or water relief from the pressurizer, thus assuring that the core remains covered.

The analysis assumes that the reactor is protected for the loss of normal feedwater event by reactor trip on a low low narrow range water level signal in any steam generator. The AFW system is started automatically by the signals identified in UFSAR Section 15.2.6.1.

The method of analysis is discussed in PSE Section 5.1.4.7.2 and the Results are discussed in PSE Section 5.1.4.7.3.

The results of explicit analysis demonstrate that a loss of normal feedwater event does not adversely affect the core, the RCS, or the steam system. Auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves, and the water level in all steam generators receiving auxiliary feedwater is maintained above the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves. The results of this report confirm that the conclusions of the UFSAR remain valid.

4.2.3.2.8 Feedwater System Pipe Break

The major feedwater line rupture event is described in Section 15.2.8 of the UFSAR. This event is defined as a break in the feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. A break upstream of the feedline check valve would affect the nuclear steam supply system only as a loss of feedwater and that case is covered by the evaluation in section 4.2.3.2.7, "Loss of Normal Feedwater Flow," and in PSE Section 5.1.4.7.

As noted in the UFSAR, depending upon the size of the break and the plant operating conditions, a major feedline rupture could cause either a RCS cooldown (by excessive energy discharge through the break) or a RCS heatup. The potential cooldown resulting from a secondary pipe rupture is evaluated in Section 5.1.3.5, so that only the RCS heatup effects are evaluated for a feedline rupture.

A feedwater line rupture event reduces the ability to remove heat generated by the core from the RCS. As discussed in UFSAR Section 15.2.8.1, the analysis must show that the auxiliary feedwater system can assure that even in the event of a major feedline rupture, no substantial overpressurization of the RCS shall occur and decay heat is removed in order to maintain sufficient liquid in the RCS to keep the reactor core covered.

A major feedwater line rupture is classified as an ANS Condition IV event. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.4. To conservatively assure that the general Condition IV criteria are met, the specific criterion applied to this event is that no bulk boiling occurs in the primary coolant system following a feedline rupture, prior to the time that the heat removal capability of the steam generators receiving auxiliary feedwater flow exceeds nuclear steam supply system heat generation.

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In addition to the actuation of the auxiliary feedwater system, the following plant protection system functions can act to mitigate the effects of a major feedwater line rupture:

1. Reactor trip on high pressurizer pressure, overtemperature ΔT, and low-low steam generator water level in any steam generator.

2. Safety injection signal from low steam line pressure or high containment pressure (Hi-1).

To prevent the pressurizer from going water solid, the analysis assumes the operator cross connects AFW to a second steam generator or opens the PORV to the intact steam generator 30 minutes after initiation of the event.

The method of analysis is discussed in PSE Section 5.1.4.8.2 and the Results are discussed in PSE Section 5.1.4.8.3.

Results of the analyses for both an RCS average temperature of 582.3°F or 593.0°F show that for the postulated feedwater line rupture, the assumed auxiliary feedwater system is adequate to remove decay heat, prevent overpressurizing the RCS, and prevent bulk boiling/uncovering of the reactor core. Therefore, the conclusions of the UFSAR remain valid.

4.2.3.3 Decrease in Reactor Coolant System Flowrate

A number of faults which could result in a decrease in the reactor coolant system flowrate are postulated. These events are discussed in this section.

4.2.3.3.1 Partial Loss of Forced Reactor Coolant Flow

The partial loss-of-coolant flow accident is described in Section 15.3.1 of the UFSAR. This event may arise following a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by the a reactor coolant bus. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

The partial loss of forced reactor coolant flow event is classified as an ANS Condition II event, i.e. a fault of moderate frequency. The general criteria for this event category are discussed in UFSAR Section 15.0.1.2. The actual limiting criterion which the analysis for this event primarily addresses is to demonstrate that the DNB design basis is met.

Protection against the partial loss-of-coolant event is provided by the low primary coolant flow trip signal which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive-8, low flow in

any loop actuates a reactor trip. Between approximately 10% rated thermal power (Permissive-7) and the power level corresponding to Permissive-8, low flow in any two loops actuates a reactor trip. Above Permissive-7, two or more reactor coolant pump circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip.

The method of analysis is discussed in PSE Section 5.1.5.1.2 and the Results are discussed in PSE Section 5.1.5.1.3.

The analysis shows that the minimum DNBR remains above the limit value (see PSE Table 4-2) at all times during the transient. Thus, no adverse fuel effects or clad rupture is predicted, and all applicable acceptance criteria are met. Therefore, the conclusions of the UFSAR remain valid for this event.

4.2.3.3.2 Complete Loss of Forced Reactor Coolant Flow

UFSAR Section 15.3.2 describes a complete loss of forced reactor coolant flow, which may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent adverse effects to the fuel if the reactor were not tripped promptly.

The complete loss of forced reactor coolant flow event is classified as an ANS Condition III event, i.e. an infrequent fault. The general criteria for this event category are discussed in UFSAR Section 15.0.1.3. The actual limiting criterion that is conservatively applied to this event by Westinghouse is to demonstrate that the DNB design basis is met.

The signals which provide the necessary plant protection for this event at the South Texas Units 1 and 2 are:

1. Reactor coolant pump undervoltage or underfrequency
2. Low reactor coolant loop flow

These trip functions are described in the UFSAR. For the analysis of the complete loss-of-coolant flow event, the reactor trip actually assumed in the analysis is the pump undervoltage trip.

The method of analysis is discussed in PSE Section 5.1.5.2.2 and the Results are discussed in PSE Section 5.1.5.2.3.

The analysis shows that the minimum DNBR remains above the limit value at all times during the transient. Thus, no adverse fuel effects or clad rupture is predicted, and all applicable acceptance criteria are met. Therefore, the conclusions of the UFSAR for this event continue to be valid.

4.2.3.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

The reactor coolant pump shaft seizure event is discussed in UFSAR Section 15.3.3. For the instantaneous seizure of a reactor coolant pump rotor, flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the Reactor Coolant System.

The insurge into the pressurizer causes a pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves in a sequence dependent on the rate of insurge and pressure increase. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident; however, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray are not included in this analysis.

The locked rotor event is classified as an ANS Condition IV incident, i.e. a limiting fault. The general acceptance criteria applicable to this event category are discussed in UFSAR Section 15.0.1.4. The specific criteria considered in the analysis are as noted in the UFSAR. That is, at no time during the transient can the RCS pressure exceed that which corresponds to the faulted allowable stress limit. Also, the peak clad surface temperature calculated for the hot spot must remain below 2700°F. Finally, the analysis for this event includes the calculation of the percentage of fuel rods postulated to experience DNB. This value is used as input to the consideration of the radiological consequences associated with this event.

The method of analysis is discussed in PSE Section 5.1.5.3.2 and the Results are discussed in PSE Section 5.1.5.3.3.

For the locked rotor analysis, the peak RCS pressure reached is less than that which would cause stresses to exceed the faulted allowable stress limits (typically 116% design), thereby assuring that the integrity of the primary coolant system is maintained. This represents a change in the current acceptance limit of 110% design pressure. Since the peak clad surface temperature calculated for the hot spot during the transient remains considerably less than 2700°F (the temperature at which clad embrittlement may be expected), the core will remain in place and intact with no loss of core cooling capability.

The percentage of fuel rods in the core postulated to undergo DNB during the locked rotor event is less than 10%. Therefore, the conclusions of the UFSAR with respect to the locked rotor event continue to be valid.

4.2.3.3.4 Reactor Coolant Pump Shaft Break

The evaluation for this event remains consistent with that presented in UFSAR Section 15.3.4. That is, the model used for analyzing the locked rotor event of PSE Section 5.1.5.3 bounds the most limiting conditions with respect to either a locked rotor or pump shaft break event. With a failed shaft, the impeller could conceivably be free to spin in the reverse direction, as opposed to being fixed in position (as would occur for a locked rotor). The effect of such reverse spinning is a slight decrease in the end point core flow when compared to the locked rotor. The model for the analysis of PSE Section 5.1.5.3 explicitly includes this effect, so that the results bound the most limiting conditions for the locked rotor and shaft break; therefore, a separate analysis for the shaft break is not presented.

4.2.3.4 Reactivity and Power Distribution Anomalies

As discussed in Section 15.4 of the UFSAR, a number of faults have been postulated which could result in reactivity and power distribution anomalies. Included among the possible causes of reactivity changes are RCCA motion or ejection. Possible sources of power distribution changes include RCCA motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section.

4.2.3.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

As described in UFSAR Section 15.4.1, an RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is highly unlikely, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical, at hot zero power, or at power. The "at power" case is discussed in PSE Section 5.1.6.2.

This event is classified as an ANS Condition II event, i.e. a fault of moderate frequency. The general acceptance criteria for this event category are presented in UFSAR Section 15.0.1.2. The specific acceptance criterion applied is to demonstrate that the DNB design basis is satisfied for this event.

As discussed more fully in UFSAR Section 15.4.1.1, should a continuous RCCA withdrawal accident occur, the automatic features of the reactor protection system available to terminate the transient are as follows:

1. Source range high neutron flux reactor trip
2. Intermediate range high neutron flux reactor trip
3. Power range high neutron flux reactor trip (low setting)
4. Power range high neutron flux reactor trip (high setting)
5. High nuclear flux rate reactor trip

In addition, control rod stops on high intermediate range flux and high power range flux serve to discontinue rod withdrawal and may prevent the need to actuate the intermediate range flux trip and the power range flux trip, respectively.

The method of analysis is discussed in PSE Section 5.1.6.1.2 and the Results are discussed in PSE Section 5.1.6.1.3.

In the event of an RCCA withdrawal from subcritical accident taking place, the core and the RCS are not adversely affected since the fuel and DNB design basis continues to be met throughout the transient. Therefore, the conclusions presented in the UFSAR for this event remain valid.

4.2.3.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power

The uncontrolled RCCA bank withdrawal at power event, as described in UFSAR Section 15.4.2, results in an increase in core heat flux. Since the heat extraction from the steam generator lags behind the power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB.

This event is classified as an ANS Condition II, i.e. a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1. The primary acceptance criterion explicitly considered in the analysis of this event is to verify that the DNB design basis is satisfied for the transition to VANTAGE 5H fuel.

As discussed more fully in the UFSAR, the following reactor trips are intended to provide the required protection against an uncontrolled RCCA bank withdrawal at power event:

1. Power range high neutron flux
2. Overtemperature ΔT
3. Overpower ΔT
4. High pressurizer pressure
5. High pressurizer water level

In addition to these reactor trips, there are RCCA withdrawal blocks available on high neutron flux, overpower ΔT, and overtemperature ΔT.

The manner in which the combination of overpower ΔT and overtemperature ΔT trips provide protection over the full range of RCS conditions is illustrated in PSE Figure 5.1.1-6. This figure presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT and overtemperature ΔT trip functions are represented as "protection lines" on this diagram. The protection lines include all adverse

instrumentation and setpoint errors so that under nominal conditions a trip would occur wel within the area bounded by these lines.

The method of analysis is discussed in PSE Section 5.1.6.2.2 and the Results are discussed in PSE Section 5.1.6.2.3. The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates to ensure that the minimum value of DNBR is always larger than the limit value discussed in PSE Section 4.0.

4.2.3.4.3 Rod Cluster Control Assembly Misalignment

UFSAR Section 15.4.3.1 contains a general discussion of the RCCA misalignment accidents that identifies causes and provides descriptions of the specific events involved. As presented there, the RCCA misalignment accidents include:

1. One or more dropped RCCAs within the same group
2. A dropped RCCA bank
3. Statically misaligned RCCA
4. Withdrawal of a single RCCA

The current licensing basis analysis reports that an upper bound on the number of fuel rods predicted to experience DNB is 5% of the total fuel rods in the core. The single RCCA withdrawal analysis for the VANTAGE 5H transition uses the same 5% limit as a measure of acceptability for this event.

The transition to VANTAGE 5H fuel, itself, has almost no impact on the discussion in UFSAR Section 15.4.3 with regard to the various RCCA misalignment accidents. For example, the means of detecting a dropped RCCA bank, misaligned RCCA, or single RCCA withdrawal remain just as described in UFSAR Section 15.4.3.1. However, certain of the modified safety analysis assumptions, while not impacting the general methodology, do alter the explicit DNB related calculations for a given case. Among the assumptions falling into this category are the use of the Revised Thermal Design Procedure and the increased $F_{\Delta H}$ power distribution peaking factor.

The method of analysis is discussed in PSE Section 5.1.6.3.2 and the Results are discussed in PSE Section 5.1.6.3.3.

For cases of dropped RCCAs or dropped RCCA banks, the DNBR remains greater than the limit value. Therefore, the DNB design basis is met and the conclusions of the UFSAR for this event remain valid.

For the limiting cases associated with the statically misaligned RCCA event the DNB remains above the safety analysis limit value. Therefore, the DNB design basis is met and the conclusions of the UFSAR remain valid for this event.

For the accidental withdrawal of a single RCCA event, with the reactor in either automatic or manual control mode and initially operating at full power, an upper bound on the number of fuel rods experiencing DNBR is 5 percent of the total fuel rods in the core.

4.2.3.4.4 Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature

The startup of an inactive reactor coolant loop at an incorrect temperature event, as described in UFSAR Section 15.4.4 results in an increase in core heat flux. If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the SG in the inactive loop is not isolated, there is a temperature drop across the SG in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Operational procedures require that the unit be brought to a load of less than 25 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle RCP without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II, i.e., a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0. 1. The primary acceptance criterion explicitly considered in the analysis of this event is to verify that DNB design basis is satisfied for the transition to VANTAGE 5H fuel.

Should the startup of an inactive RCL accident occur, the transient will be terminated automatically by a reactor trip on low reactor coolant flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three-loop operation.

The method of analysis is discussed in PSE Section 5.1.6.4.2 and the Results are discussed in PSE Section 5.1.6.4.3.

The transient results of the startup of an inactive reactor coolant loop at an incorrect temperature event show that the core is not adversely affected. The DNBR remains above the safety analysis limit value throughout the transient; thus, the DNBR design basis as described in PSE Section 4.0 is met.

4.2.3.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

As described in UFSAR Section 15.4.6, reactivity can be added to the core by feeding unborated water into the RCS via the CVCS. Boron dilution is a manual operation under strict administrative controls, with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The CVCS is designed to limit the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve provides makeup to the RCS which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. A block diagram summarizing the various protection sequences for safety actions required to mitigate the consequences of the event is provided on UFSAR Figure 15.0-19. Additionally, UFSAR Section 15.0.8 and UFSAR Table 15.0-6 briefly address the plant systems and equipment available to mitigate the consequences of the event. This event is classified as an ANS Condition II incident, a fault of moderate frequency, as defined in UFSAR Section 15.0.1.

The method of analysis is discussed in PSE Section 5.1.6.5.2 and the Results are discussed in PSE Section 5.1.6.5.3.

The results presented above, show that there is adequate time (at least 15 minutes) for the operator to manually terminate the source of the dilution flow in the full power, startup, hot standby, hot shutdown, and cold shutdown (with the RCS not drained) modes of operation. Following termination of the dilution flow, the reactor will be in a stable condition. The operator can then initiate reboration to recover shutdown margin. Uncontrolled boron dilution in the cold shutdown (with the RCS drained down) and refueling modes is administratively precluded.

4.2.3.4.6 Spectrum of Rod Cluster Control Assembly Ejection Accidents

This accident, as discussed in UFSAR Section 15.4.8, is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

This event is classified as an ANS Condition IV event, i.e. a limiting fault. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.4. The specific limiting criteria evaluated in the analysis for this event are summarized as follows:

1. Average fuel pellet enthalpy at hot spot below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.

2. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

3. Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

It should be noted that the current UFSAR includes an additional criterion that the average clad temperature at the hot spot must remain below 2700°F. The elimination of the clad temperature criterion as a basis for evaluating RCCA ejection is consistent with the revised Westinghouse acceptance criteria for this event, as defined in Reference 31.

The only reactor trip function assumed in the analysis for this event is power range high neutron flux, both high and low setting.

The method of analysis is discussed in PSE Section 5.1.6.6.2 and the Results are discussed in PSE Section 5.1.6.6.3.

The analyses indicate that the described fuel limits are not exceeded. It is concluded that there is no likelihood of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no likelihood of further consequence to the RCS. The analyses have also demonstrated that, for fission product release calculations, the number of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core.

4.2.3.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

The in-core instrumentation's ability to detect gross differences between measured and predicted thimble reaction rates is unaffected by fuel type; therefore, the conclusions of the UFSAR remain valid.

4.2.3.5 Increase in Reactor Coolant Inventory

The events which could result in an increase in the reactor coolant system inventory are discussed in this section.

4.2.3.5.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

As stated in the UFSAR, Spurious Safety Injection System operation at power could be caused by error or a false electrical actuation signal. However, spurious SI without an immediate reactor trip has no effect on the RCS. If a reactor trip is generated by the spurious SI signal, a normal shutdown can be commenced with without boration from the SI pumps because of the shutoff head of about 1600 psi.

4.2.3.5.2 Chemical and Volume Control Systems Malfunction That Increases Reactor Coolant Inventory

A malfunction of the CVCS could result in the inadvertent injection of borated water, which could lead to filling the pressurizer to a water-solid condition.

The most limiting case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow to be isolated. This case also conservatively assumes that the charging pumps draw suction from the volume control tank (VCT) for the duration of the transient, precluding the introduction of higher borated water from the Refueling Water Storage Tank (RWST) to the RCS. To prevent filling the pressurizer water-solid, the operator must terminate charging.

The method of analysis is discussed in PSE Section 5.1.7.1.2 and the Results are discussed in PSE Section 5.1.7.1.3.

The results show that none of the operating conditions during the transient approach core limits. To prevent water relief from the pressurizer, the operator must terminate charging. The sequence of events presented in PSE Table 5.1.7-1 show that the operator has sufficient time (10 minutes) to take corrective action, and the response time is discussed in Section 5.1.7.1.4 of the PSE.

4.2.3.6 Decrease in Reactor Coolant Inventory

The inadvertent opening of a pressurizer safety or relief valve, which results in a decrease in RCS inventory, and the steam generator tube rupture accident are discussed in this section.

4.2.3.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

An accidental depressurization of the RCS, as described in UFSAR Section 15.6.1, could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a pressurizer safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, its opening produces a much more rapid RCS depressurization. Therefore, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results

in rapidly decreasing RCS pressure, and the effect of the pressure decrease is to increase power via the moderator density feedback (assuming a positive moderator temperature coefficient). The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

This event is classified as an ANS Condition II accident, i.e. a fault of moderate frequency. The general acceptance criteria for this event category are discussed in UFSAR Section 15.0.1.2. The specific acceptance criterion applied in the analysis of this event is that the DNB design basis be satisfied. The reactor protection system signals that would be expected to trip the reactor for this event are overtemperature ΔT and pressurizer low pressure.

The method of analysis is discussed in PSE Section 5.1.8.1.2 and the Results are discussed in PSE Section 5.1.8.1.3.

The results of the analysis shows that the low pressurizer pressure and overtemperature ΔT reactor protection system signal provides adequate protection against the RCS depressurization event.

4.2.3.6.2 Steam Generator Tube Rupture

The licensing basis Steam Generator Tube Rupture (SGTR) analysis for South Texas Units 1 and 2 is presented in WCAP-12369 (Reference 24) and the results are included in Section 15.6.3 of the UFSAR. The SGTR analysis includes an analysis to demonstrate margin to steam generator overfill and an analysis to ensure that the offsite radiation doses resulting from the event are less than the allowable guideline values specified in Standard Review Plan 15.6.3 and 10CFR100. An analysis has also been performed to determine the effect of a 10°F T-Hot reduction on the SGTR analysis for South Texas, and the results of the T-Hot reduction analysis are presented in WCAP-12833 (Reference 25). The SGTR analysis for the T-Hot reduction indicated that margin to overfill will be maintained.

The evaluation to determine the effect of the V5H fuel upgrade and associated changes on the results of the SGTR analysis for South Texas Units 1 and 2 addresses the UFSAR analysis performed for design RCS temperatures and the analysis performed for the T-Hot reduction. The criteria used in the UFSAR SGTR analysis and the T-Hot reduction analysis were also used as the basis for this evaluation to establish the continued applicability of the analyses by demonstrating that the conclusions in the UFSAR and T-Hot reduction analyses remain valid.

The plant response to the SGTR event was modeled using the LOFTTR2 computer program using conservative initial conditions and assumptions. The analysis methodology includes the simulation of the operator actions for recovery from an SGTR event based on the South Texas Emergency Operating Procedures, with operator action times developed from plant simulator studies. The SGTR analysis for the T-Hot reduction was also performed using the same methodology and assumptions used for the UFSAR analysis, with the exception of the differences in the initial reactor coolant and secondary parameters associated with the T-Hot reduction.

An evaluation was performed to determine the effect of the VANTAGE 5H fuel upgrade and associated changes on the margin to overfill and offsite dose analyses for both the UFSAR SGTR analysis and the T-Hot reduction analysis. It was determined that the changes either do not impact the SGTR analysis or are already incorporated in the SGTR analysis, with the exception of the changes to the OTΔT reactor trip setpoint. Since reactor trip occurs on an OTΔT signal for the SGTR analysis, an evaluation was performed to assess the impact of the setpoint changes on the analysis. The changes to the OTΔT trip setpoint are designed to provide margin to reactor trip, and thus the changes will result in a delay in reactor trip for an SGTR. Since sensitivity studies have demonstrated that earlier reactor trip is conservative for both the margin to overfill and offsite dose analyses, the changes to the OTΔT trip setpoint will not adversely affect the SGTR analysis. On this basis, it is concluded that the results of the SGTR analysis in the UFSAR and the SGTR analysis for the T-Hot reduction in WCAP-12833 remain valid for operation with the VANTAGE 5H fuel upgrade and associated changes.

The effect of the VANTAGE 5H fuel upgrade and associated changes has been evaluated for the SGTR analysis for South Texas Units 1 and 2. Based on this evaluation, it is concluded that the results of the SGTR analysis in the UFSAR and the SGTR analysis for the T-Hot reduction in WCAP-12833 remain valid.

4.2.4 LOCA Evaluations

4.2.4.1 Large Break LOCA (LBLOCA) Evaluations

Large break LOCA (LBLOCA) analyses have been performed for the South Texas Project which assess the affect of the VANTAGE 5H upgrade and additional changes specified in Section 3.0 and PSE Section 5.0 on the results presented in Chapter 15 of the UFSAR. A detailed discussion of the analysis is provided in PSE Section 5.2.1. The Westinghouse 1981 Evaluation Model + BASH (Previously accepted by the NRC) was utilized and a spectrum of cold leg breaks were analyzed. This model was modified to consider the special characteristics of 14ft cores and resulted in several adjustments. Zirc grid cooling is discussed in an earlier W letter to the NRC. SI Accumulator temperatures above 90°F are discussed in an HL&P letter to the NRC (Ref. 28). And the power shapes used in these analyses are consistent with the methodology of WCAP 12909-P (Ref. 27). These additions represent changes in methodology from the current licensing basis.

The large break LOCA analysis performed for the South Texas Project Units 1 and 2 has demonstrated that for breaks up to a double-ended severance of the reactor coolant piping, the Emergency Core Cooling System (ECCS) will meet the acceptance criteria of Title 10 CFR Part 50 Section 46. That is:

1. The calculated peak cladding temperature will remain below the required 2200°F.

2. The amount of fuel cladding that reacts chemically with the water or steam does not exceed 1% of the hypothetical amount that would be generated if all the zirconium metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the LOCA.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat produced by the long-lived radioactivity remaining in the core.

The time sequence of events for all breaks analyzed is shown in PSE Table 5.2.1-3. The large break LOCA analysis assuming a full core of VANTAGE 5H fuel utilizing the 1981 EM + BASH calculational model, resulted in a peak cladding temperature of 2165°F for the limiting DECLG break at a total peaking factor of 2.70. The maximum local metal-water reaction was less than 17%, and the total core wide metal-water reaction was less than 1.0% for all cases analyzed. Further, the clad temperature transients turned around at a time when the core geometry was still amenable to cooling.

4.2.4.2 Small Break LOCA (SBLOCA) Analysis

The SBLOCA was analyzed to determine the affect on the UFSAR Chapter 15 analysis resulting from the upgrade to VANTAGE 5H fuel and other changes specified in Section 3.0 and PSE Section 5.0. The approved NOTRUMP Small Break ECCS Evaluation Model was utilized for a spectrum of cold leg breaks. Discussion of the analysis performed is found in PSE Section 5.2.2.

The small break VANTAGE 5H LOCA analysis, utilizing the currently approved NOTRUMP Evaluation Model, resulted in a peak cladding temperature (PCT) of 1836°F for the 1.5-inch diameter cold leg break. The analysis assumed a limiting small break power shape consistent with a LOCA $F_Q(z)$ envelope of 2.70 at the core midplane elevation linearly decreasing to 2.49 near the top of the core. The maximum local metal-water reaction is 5.61 percent, and the total core metal-water reaction is less than 1.0 percent for all cases analyzed. The clad temperature transients turn around at a time when the core geometry is still amenable to cooling.

The analyses described in the preceding sections demonstrate that the high head safety injection pumps, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperature well below the required limits of 10 CFR 50.46. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

4.2.4.3 Long Term Cooling - Post LOCA Shutdown

Long-term cooling is discussed in UFSAR Section 15.6.5. The Westinghouse licensing position for satisfying the requirements of 10CFR 50.46 Paragraph (b) Item (5) "Long-term cooling" is defined in WCAP-8339 (Reference 22). The Westinghouse evaluation model commitment is that the reactor remain shutdown by the borated ECCS water. A discussion of this topic is provided in PSE Section 5.2. Confirmation that the proposed increase in the boron concentration of the RWST and accumulators will provide enough margin to

keep the core subcritical for long term cooling requirements will be verified through the normal reload evaluation process.

4.2.4.4 Long Term Cooling - Boron Precipitation

The following discussion and change to the hot leg switchover time were previously submitted to the NRC (Reference 15) and this discussion is presented here for continuity only.

A discussion of the hot leg switchover time is presented in UFSAR Section 15.6.5. An analysis has been performed to determine the maximum boron concentration in the reactor vessel following a hypothetical LOCA. This analysis considered a proposed maximum boric acid concentration of 3000 ppm in the RWST and accumulators and 2800 ppm in the RCS. A discussion of this topic is provided in PSE Section 5.2. The results of the analysis show that the maximum allowable boric acid concentration of 23.53 weight percent (the boric acid solubility limit less 4 weight percent) will not be exceeded in the vessel if hot leg injection is initiated 10.5 hours after the inception of a LOCA. This is a reduction from the current licensing basis time of 13 hours. The typical time interval between the accident inception and reactor trip/SI actuation signal is negligible when compared to the switchover time.

4.2.4.5 LOCA Reactor Vessel and Loop Forces

Section 5.2 of the PSE provides a discussion of the effect of the upgrade to VANTAGE 5H and associated changes on the LOCA reactor vessel and loop forces.

The blowdown forces created by a hypothesized rupture in the reactor coolant system (RCS) are caused principally by the propagation of an associated decompression wave. The strength of the decompression wave is a function of the assumed break opening time, the pipe break area and relative location of the break, and the RCS steady state operating conditions of power, temperature and pressure. These parameters are not significantly affected by a change to VANTAGE 5 Hybrid fuel at South Texas Project Units 1 and 2. The magnitude of the LOCA hydraulic forces in the fuel region is principally a function of the steady state pressure distribution associated with the loss coefficients, loop flow, and density. Previous studies performed for these calculations have demonstrated that there will be no adverse change in the in the LOCA forces calculated for a hypothesized LOCA as a result of this fuel change. Those LOCA hydraulic forces acting upon the RCS loop piping are also not significantly affected by the changes in the fuel assembly design. Analyses presented in WCAP 13051 for T_{hot} reduction show that the limits continue to be met at all RCS temperatures considered.

4.2.5 Plant and Systems Evaluation

This safety evaluation assesses the impact of the VANTAGE 5H fuel upgrade on the auxiliary fluid systems and the Instrumentation and Controls System Performance and Class 1E Equipment Qualification.

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4.2.5.1 Auxiliary Fluid Systems

The impact of the VANTAGE 5H fuel upgrade safety functions of the auxiliary fluid systems have been reviewed and found to be unaffected by the fuel upgrade. However, a change to the boric acid storage tank volume is required to meet Modes 5 and 6 reactor coolant system concentration requirements. This is based on a reasonable value of 2800 ppm, which is expected to bound future cores. The current boration requirement is 1600 gallons (2900 gallons contained, 1300 unusable), whereas the revised requirement is 1880 gallons. Therefore, Reference 15 proposes to revise the Technical Specification value of 2900 gallons to 3200 gallons to account for this increase. The volume change will have no adverse affect on the operability of auxiliary fluid systems. There is no tank level setpoint associated with this volume.

In addition, the Refueling Water Storage Tank (RWST) boron concentration was changed to a range of 2800 to 3000 ppm based on 2800 ppm minimum and adding 200 ppm to the minimum to get the maximum value. Also, the accumulator boron concentration was revised based on the RWST concentration. Minimum accumulator concentration is set equal to 100 ppm less than the RWST minimum and the maximum is equal to the RWST. Thus, accumulator concentration is set in the range of 2700 to 3000 ppm.

The above modifications to the Technical Specifications and UFSAR were previously submitted to the USNRC (Reference 15) and these discussions are presented here for continuity only.

No changes in accumulator or RWST volumes or setpoints were made. Also, no change was made to the operating temperature range of the RWST.

The performance of the Chemical and Volume Control System, Emergency Core Cooling System, and Containment Spray System are not adversely affected by the fuel upgrade, since there are no changes to the flow resistance characteristics of the pumps or injection paths. The capability of the Residual Heat Removal System is sufficient based on the decay heat values used in the past for RHR cooldown calculations. These values are more conservative for 18 month fuel cycles out to 150 hours after plant shutdown than the ANSI/ANS-5.1-1979 decay heat standard.

4.2 5.2 Instrumentation and Controls System Performance and Class 1E Equipment Qualification

The Reactor Coolant System (RCS) instrumentation is part of the reactor protection system which must conform with criteria of the UFSAR Chapter 7 and IEEE Std. 279-1971. 10CFR50.55a(h) established the compliance with the criteria of IEEE Std.279-1971 as a licensing requirement for plants issued a construction permit after January 1, 1971. IEEE Std. 279-1971, Section 4.4, and 10CFR50.49; the NRC final rule on environmental qualification of electrical equipment important to safety, impose requirements for the environmental qualification of Class 1E electrical equipment. Class 1E equipment is identified in the South Texas Project Equipment Qualification (EQ) program and in the master equipment list. Included among other EQ parameters and requirements are chemistry conditions, which typically refer to the break/blowdown, containment spray, and sump chemistry during accident conditions. It is necessary and sufficient to

demonstrate that changes in RCS chemistry, and the consequent changes in accident chemistry, do not invalidate the current qualification levels.

Class 1E and non-safety-related instrumentation components that have wetted contact with the RCS fluid are part of the primary system pressure boundary. It must be demonstrated that no new failure mode is introduced that could lead to an accident scenario not covered in the plant accident analysis documented in the UFSAR.

As a clarification of the licensing basis applicable to plant I&C and protection systems, it should be noted that only the sensors having wetted contact with the RCS fluid or with the contents of the Accumulator or RWST can be affected by the change in Accumulator and RWST boron concentration associated with the upgrade to VANTAGE 5H fuel under normal plant conditions. This is limited to temperature sensors and pressure and differential pressure transmitters.

HL&P has submitted a request to change the RWST boron concentration to a range of 2800 to 3000 ppm boron and to revise the accumulator boron concentration to a range of 2700 ppm to 3000 ppm boron (Reference 15). No change to the volume or setpoints of the accumulators or RWST were made. The RCS boron concentration is assumed to be 1865 ppm for the post-LOCA sump solution pH analysis. The increase in RWST and accumulator boron concentrations resulted in a post-LOCA sump solution pH between the current Technical Specification limits of 7.0 and 9.5.

Since the chemistry change remains within the limits specified in the STP Technical Specifications, it is concluded that the increase to 3000 ppm boron concentration for the RWST and Accumulators, and 1865 ppm for the RCS, no adverse impact to the corrosion of temperature or transmitter sensing elements during normal plant operations. In addition, the accident chemistry conditions expected in the containment during and following a design basis LOCA or Steamline Break fall within acceptable ranges. Thus, the assumptions and conclusions of the EQ/chemistry evaluations documented in WCAP-12861 (Reference 8) are unaffected. Similar evaluations performed by HL&P based on WCAP-12861 for BOP safety-related electrical and mechanical equipment located in containment are likewise unaffected.

The evaluations documented in WCAP-12861, and others performed by HL&P on the same bases, addressed the significance of corrosion impact within the context of expected failure modes for each item of environmentally qualified Class 1E and safety-related equipment. These evaluations were based on individual consideration of corrosion impact to each of the materials utilized in the construction of the equipment which is exposed to the accident chemistry conditions.

The chemistry change will not impact the operability of any item of Class 1E or safety-related equipment, nor is there any change in the failure effects of this equipment under normal, abnormal or accident conditions.

4.2.5.3 Control System Operation and Margin to Trip

Some of the more important parameters in defining the Nuclear Steam Supply System (NSSS) response to a temperature or power transient are the core reactivity coefficients. With the VANTAGE 5H fuel upgrade, the response of the NSSS and associated control systems will be affected by the change in core reactivity coefficients. The plant response to these changes are evaluated for control systems stability and margin to trip. As part of the upgrade, new setpoints are implemented in the ΔT trip setpoint equations and their effects are included in the analysis.

If the response of the NSSS and the control systems to normal operating transients is degraded, the available margin to various reactor trips could be reduced beyond acceptable limits. If these postulated conditions occur, control systems setpoints can be modified to restore proper NSSS response. Control systems do not perform any protective function in the UFSAR accident analysis. Control system operation is assumed only in cases where their action aggravated the consequences of an event, or as required to establish initial plant conditions for an analysis.

The Method of Analysis and the Results are discussed in Sections 5.5.2 and 5.5.3 of the PSE, respectively.

The VANTAGE 5H fuel and changes in ΔT setpoint equations should not introduce any significant differences in control systems response and only minor changes to the margin to reactor trips and ESF actuations. The only actuations that have any restrictions are the OTΔT turbine runback and OTΔT reactor trip. This is due to the possible OTΔT setpoint penalty contribution from axial offset exceeding the positive deadband. This restriction should not be interpreted as loss of operating margin, but to define a boundary for normal operations and ensure that steady state fluctuations will not result in actuation.

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TABLE 4.2-1

SAFETY ANALYSIS ASSUMPTIONS

UFSAR SECTION		Full Power Vessel Avg. Temp.	PMTC	RTDP	% SGTP	SDM
15.1.2	Feedwater Malfunction	593.0	No	Yes	0%	NA
15.1.3	Excess Increase in Steam Flow	593.0	No	Yes	0%	NA
15.1.4	Main Steam Depressurization	593.0	No	No	0%	1.3
15.1.5	Steamline Break (Core Response)	593.0	No	No	0%	1.3
15.2.2 15.2.3	Loss of Load/ Turbine Trip	593.0	Yes	Yes	10%	NA
15.2.6 15.2.7	Loss of Offsite Power Loss of Normal Feedwater/LOOP	582.3/ 593.0	Yes	No	10%	NA
15.2.8	Feedwater System Pipe Break	582.3/ 593.0	No	No	10%	1.3
15.3.1	Complete Loss of Flow	593.0	Yes	Yes	10%	NA
15.3.2	Partial Loss of Flow	593.0	Yes	Yes	10%	NA
15.3.3 15.3.4	Locked Rotor Shaft Break	593.0	No[1]	No	10%	NA*
15.4.1	Rod Withdrawal from Subcritical	NA	Yes	No	NA	NA
15.4.2	Rod Withdrawal at Power	593.0	Yes	Yes	10%	NA
15.4.4	Startup of an Inactive Coolant Loop	593.0	Yes	No	10%	NA
15.4.6	Boron Dilution	593.0	NA	NA	NA	1.3

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UFSAR SECTION		Full Power Vessel Avg. Temp.	PMTC	RTDP	% SGTP	SDM
15.4.8	RCCA Ejection	NA	Yes	No	NA	NA
15.5.2	CVCS Malfunction	582.3/ 593.0	Yes	No	10%	NA
15.6.1	RCS Depressurization	593.0	Yes	Yes	10%	NA
15.6.3	Steam Generator Tube Rupture	582.3/ 593.0	Yes	NA	15%[2]	NA
15.6.5	Large Break and Small Break LOCA	593.0	NA	NA	10%	NA
15.6.5	Long Term Cooling Post-LOCA Shutdown	NA	NA	NA	NA	NA
15.6.5	Long Term Cooling Boron Precipitation	Na	NA	NA	NA	NA

* Table 1 reflects pressure transient assumptions.
[1] PMTC = 0 at 100% RTP, PMTC=+5 pcm/°F at 90% RTP. (PMTC was utilized for Rods-In-DNB analysis.)
[2] 15% tube plugging was assumed for margin ot overfill analysis and 0% tube plugging was assumed for offsite does analysis.

TABLE 4.2-2

NON-LOCA TRANSIENTS IN WHICH
OVERTEMPERATURE DELTA-T AND OVERPOWER DELTA-T
REACTOR TRIP SETPOINTS MAY ACTUATE

UFSAR SECTION	TITLE
15.2.3	Loss of External Electrical Load
15.4.2	Uncontrolled RCCA Bank Withdrawal at Power
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve
NA	Steamline Break at Power

NOTE: The Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (UFSAR Section 15.4.6) event uses the results from a case of the Uncontrolled RCCA Bank Withdrawal at Power. That case may trip on Overtemperature ΔT.

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4.3 Increase in the Maximum Allowable Fuel Enrichment

The proposed changes include an increase in the maximum nominal enrichment for fuel assemblies from 4.5 weight percent (w/o) uranium-235 to 5.0 w/o.

4.3.1 Criticality Analyses for Fuel Storage Racks

The New Fuel Racks and In-Containment Fuel Storage Racks were reanalyzed to allow storage of Westinghouse 17x17 XL fuel assemblies with enrichments up to 5.0 w/o uranium-235. The Spent Fuel Storage Racks were previously approved for storage of assemblies with enrichments up to 5.0 w/o uranium-235 (References 13 and 14). The New Fuel and In-Containment Fuel Storage Rack criticality analyses are based on maintaining K_{eff} less than or equal to 0.95 for storage of Westinghouse 17x17 fuel assemblies under full water density conditions and less than or equal to 0.98 under low water density (optimum moderation) conditions. The optimum moderation condition applies only to the fresh fuel rack, since this rack is used to store fuel in a dry configuration.

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. For the New Fuel and In-Containment racks, this is accomplished by fixing the minimum separation between fuel assemblies. The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95% probability at a 95% confidence level that the effective neutron multiplication factor, K_{eff}, of the fuel assembly array will be less than 0.95, as recommended by ANSI 57.2-1983, ANSI 57.3-1983, and the NRC letter to power reactor licensees, "Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978. The 0.95 K_{eff} limit applies to both the New Fuel and In-Containment fuel racks under all conditions, except for the New Fuel Storage Rack under low water density (optimum moderation) conditions, where the K_{eff} limit is 0.98, as recommended by NUREG-0800 (Reference 21).

The criticality analyses, including IFBA Credit Reactivity Equivalencing and a review of postulated accidents that would increase reactivity, concluded that the acceptance criteria for criticality is met for the New Fuel and In-Containment Fuel Storage Racks for the storage of all Westinghouse 17x17 fuel assemblies with the following configurations and enrichment limits:

New Fuel Rack — Storage of fuel assemblies with nominal enrichments up to 5.0 w/o in any location. There are no requirements on position or minimum IFBA for these assemblies.

In-Containment Rack — Storage of fuel assemblies with nominal enrichments up to 4.5 w/o in any rack location. Fuel assemblies with enrichments above 4.5 w/o can also be stored, but each assembly must contain sufficient Integral Fuel Burnable Absorbers (IFBA) to satisfy the requirements specified in the criticality report.

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The criticality analysis report for the New Fuel and In-Containment Storage Racks is found in Reference 3. The reanalysis of the New and In-Containment fuel racks criticality analysis does not compromise the performance of any safety-related components or system.

4.3.2 Increased Radiological Source Terms

The impact of the increased fuel maximum enrichment on the radiological consequences of accidents has been evaluated taking into account the changes discussed in Section 3.0 and PSE Section 5.0 (Reference 1). A discussion of this evaluation is presented in PSE Section 5.4. The evaluation concludes that although the doses reported in the UFSAR increase slightly, they remain well within the acceptance limits. An evaluation of the impact of increased discharge burnup on radiological source terms was previously submitted to the USNRC, and subsequently approved, in References 16 and 17, respectively.

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4.4 Revised Reactor Containment Building P/T Response

The containment is designed to limit the release of radioactive materials to the environment after postulated accidents, such that the resulting calculated offsite doses are less than the guideline values of 10 CFR 100. The capability of the containment to perform this function is ensured by comprehensive design, analysis, and testing that includes consideration of the calculated free containment volume as well as the calculated peak containment internal pressure and temperature. The peak containment internal pressure results from the mass and energy releases of a double-ended pump suction guillotine (DEPSG) loss of coolant accident (LOCA). For maximum temperature the design basis accident is a double-ended main steam line break (MSLB) coincident with main steam isolation valve (MSIV) failure.

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 the mathematical error, the original calculation overestimated the containment free volume. The containment free volume is reported in the UFSAR and the Technical Specifications. The containment free volume is also an input to other design calculations and analyses whose results are used to establish both the design and licensing bases of the plant. The impact of the error is discussed in this section.

Based upon the proposed change to VANTAGE 5H fuel and associated uncertainty calculations for the fuel upgrade, there is some impact on the LOCA and MSLB mass and energy release rates. Changes to the uncertainty criteria affect the RCS initial conditions assumed in the mass and energy release accident analyses. In addition, the MSLB mass and energy release rates were analyzed using Westinghouse's LOFTRAN (Ref. 33) computer code instead of MARVEL (Ref. 32). As a result of the analysis, a non-conservative error in the MARVEL mass and energy release rates were found. The effects of the changes are discussed in this section.
The impact of T_{hot} reduction has been similiarly evaluated and the results show that the current analyses remain bounding.

This Safety Evaluation identifies and evaluates the proposed changes resulting from the following conditions:

Containment free volume reduction,
Containment initial temperature reduction, and
Mass and energy release changes due to fuel upgrade.
T_{hot} Reduction

4.4.1 Revised Reactor Containment Building Volume and Initial Temperatures

The containment free volume has decreased from 3.56×10^6 ft^3 to 3.41×10^6 ft^3, with a margin of error of +0.1% and -0.85%. Therefore, the revised containment volume, including the -0.85% margin of error, is 3.38×10^6 ft^3, a reduction of 5.1%. The -0.85% margin of error is applied to the containment pressure/temperature analyses

under guidelines presented in ANSI/ANS 56.4-1983, "Pressure and Temperature Analysis for Light Water Reactor Containments" (Reference 26).

A reevaluation was performed with the following changes to the safety analysis of record in the SER:

(a) Reduced containment free volume,
(b) Worst case pipe break mass and energy releases due to the V5H fuel upgrade analysis, and
(c) Reduced containment initial temperature to 110°F.

The results of the analyses show an increase in containment internal peak pressure from 37.5 psig to 41.2 psig. The peak containment internal atmospheric region temperature increases from 323°F to 327°F.

These values are presented and used both in the UFSAR as well as the Technical Specifications. In addition, these values are inputs to numerous other analyses whose results are also found in the UFSAR and Technical Specifications.

The proposed changes involve revisions to applicable portions of Technical Specification Sections 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 3/4.6.1.5, B3/4.6.1.2, B3/4.6.1.4, B3/4.6.1.6, and 5.2.1, and UFSAR Sections 3.8, 3.11, 5.4, 6.2, 6.4, 6.5, 7.A, 10.4, 10.4A, 12.2, 15.4, 15.6, 15.7, with the above revised values as well as revised calculated margins, setpoints, radiological doses, and editorial changes.

4.4.2 Safety Evaluation

This safety evaluation is based on review of numerous calculations and other documents. The effect on those calculations or documents, which were substantially impacted by the change in containment free volume, reduced containment initial temperature, the new mass and energy releases for the fuel upgrade, and T_{hot} Reduction are described in this safety evaluation under the following topics:

1) Containment Maximum Temperature
2) Containment Maximum Pressure
3) Containment Minimum Pressure
4) Containment Subcompartment Analysis
5) Containment Safety-Related Equipment Qualification
6) Containment Leakage
7) Containment Minimum Backpressure for LOCA ECCS Analysis
8) Containment Hydrogen Generation
9) Safety Injection/Containment Spray Pump Operation
10) Radiological Dose Analysis

The discussions presented in this Safety Evaluation are summaries. Refer to Reference 3 for additional detail.

4.4.2.1 Containment Maximum Temperature

The containment maximum temperature occurs due to a DBA MSLB. The original MSLB mass and energy releases were calculated using the MARVEL code (Ref. 32). For the V5H Fuel Upgrade effort, the MSLB mass and energy release rates were recalculated using the LOFTRAN code (Ref. 33).

Reanalysis for the fuel upgrade effort changed the worst case MSLB with minimum CHRS to MSLB with one MSIV single active failure. Previous MSLB analyses used a containment free volume of $3.56x10^6$ ft^3. For the fuel upgrade effort, the MSLB analyses were reanalyzed with the following major changes:

(1) Reduced free volume of $3.30x10^6$ ft^3.
(2) Revised MSLB mass and energy releases.
(3) Reduced containment initial temperature to 110°F.
(4) Addition of new passive heat sinks.

The results of the double-ended guillotine rupture MSLB analyses bound the results of the MSLB split breaks for peak containment temperature.

The revised UFSAR Figure 6.2.1.1-27 shows the behavior of the containment vapor temperature. The reanalysis of the DBA MSLB increases the peak containment temperature from 323°F to 327°F. UFSAR Table 6.2.1.1-3 states that the containment structure is designed for 286°F based on the peak containment vapor temperature of 323°F. The revised UFSAR Figure 6.2.1.1-27 that the DBA MSLB the temperature above 286°F exists for the first 110 seconds of the transient. During this brief period (110 seconds), the heat transfer coefficient is not sufficiently high enough to heat the containment structures to the vapor temperature. Therefore, the containment structural temperature will not exceed 286°F and the design structural temperature of 286°F remains bounding.

Refer to Reference 3 for additional detail.

4.4.2.2 Containment Maximum Pressure

The design basis mass and energy releases to the containment as a result of LOCA are described in UFSAR Section 6.2. The long term mass and energy releases and containment pressure responses following LOCA consider the effects of long term depressurization and secondary side heat transfer. The analyses consider the total energy available to the containment from both the primary and secondary side sources at all times of the transient.

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. Based upon the results of an assessment of the current limiting LOCA mass and energy release analysis, the revised uncertainty criteria does reflect changes as compared to the assumptions for the DBA LOCA.

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The results of the evaluation for the V5H fuel upgrade demonstrates that an increase in the blowdown mass and energy release of 0.25% bounds the uncertainty criteria. A multiplier of 1.0025 has been applied to the LOCA mass and energy release rates starting at time zero to the end of the blowdown portion of the transient. The calculation was conservatively performed based upon initial condition effects on the calculated mass and energy releases. For the post-blowdown phase this penalty is not required, since the currently listed releases remain bounding.

The net effect of Thot reduction is to increase the LOCA blowdown phase mass flowrate (during the first 25 seconds) by 2% and decrease the energy releases by 0.6%. For the post-blowdown phase the LOCA mass and energy releases remain unchanged.

The net effect of the 2% blowdown phase mass flowrate increase coupled with a 0.6% decrease in the energy flow rate due to Thot reduction will be minimal. Based on current Westinghouse models it is expected that these changes will have negligible effect on the long-term pressure transient results. Therefore, it is concluded that the Thot reduction long-term LOCA mass and energy releases are bounded by the existing design basis.

The maximum calculated peak containment pressure, as described in UFSAR Section 6.2.1.1.3, results from a DEPSG LOCA with maximum safety injection and minimum containment heat removal systems in operation. As identified in UFSAR Table 6.2.1.1-3 the containment design internal maximum pressure is 56.5 psig. The revised peak pressure of 41.2 psig is well below this design pressure. The increase from 37.5 to 41.2 psig reduces the design pressure margin (shown in UFSAR Table 6.2.1.1-2) from 33.6% to 27.1%.

The 27.1% margin is well above the 10% minimum margin stipulated by the acceptance criteria of the Standard Review Plan (SRP) Section 6.2.1.1.A. In addition, the revised long-term containment pressure will still be less than 50% of the peak pressure within 24 hours. This condition meets the requirements specified in Safety Evaluation Report (SER) Section 6.2.1.1.1.

4.4.2.3 Containment Minimum Pressure

The containment volume is not used in the equation for determining the minimum pressure. This calculation 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 does affect the calculation. The new minimum pressure is -2.9 psig (11.8 psia). Hence, the containment minimum design pressure of -3.5 psig (11.2 psia), reported in UFSAR Table 6.2.1.1-2, and Technical Specification B3/4.6.1.4 is still applicable and bounding.

4.4.2.4 Containment Subcompartment Analysis

Containment subcompartment analyses are performed to calculate the maximum differential pressures in subcompartments. The results of the analyses are used to determine forces and moments on equipment and structures in the compartments. The effects of reduced containment volume, reduced containment initial temperature, and the short term mass and energy releases due to fuel upgrade are addressed in the subsections below.

In each case, Reference 3 should be consulted for additional detail.

4.4.2.4.1 Main Steam Line Break Subcompartment Analysis

There is no adverse impact on the short term mass and energy releases for the fuel upgrade effort. Therefore, only the effects of containment volume reduction need to be addressed.

For subcompartments which are not in direct contact with the main containment volume there is no impact on the calculated pressures. For those subcompartments which are in direct contact with the main containment volume, the peak differential pressures are expected to increase by no more than 6%. The results of the analysis are presented in UFSAR Table 6.2.1.2-13. Table 6.2.1.2-13 incorporates the highest calculated differential pressures from either MSLB or FWLB analyses. The results of the MSLB analysis bound the results of the FWLB analysis for all subcompartments. Table 6.2.1.2-13 shows that sufficient margin exists. SER Section 6.2.1.2 states that substantial design margins exist in the Main Steam and Feedwater Line Subcompartments. Therefore, the original subcompartment design differential pressures remain bounding.

4.4.2.4.2 Pressurizer Subcompartment

For STP, the surge line break produces the most severe pressure pulse in the subcompartments. Therefore, the surge line break dictated the design of the adjacent compartments. In the pipe break evaluation it was concluded that the guillotine breaks in major RCS piping would not occur given an initial critical flaw and then subjected to maximum loading conditions. This is called leak-before-break (LBB) methodology. The technical basis and supporting documentation for excluding the major RCS piping are contained in Section 3.6.2 of the SER, Supplement No. 4 (SSER 4). The large bore lines eliminated by the LBB methodology are the pressurizer surge line, the 8-inch, 10-inch, and 12-inch accumulator lines, which include a portion of the residual heat removal (RHR) lines. Therefore, they will not be considered as a part of the structural design basis of the South Texas plant. The application of the LBB methodology leaves only the spray line break as the design basis for the containment subcompartment analysis.

The reanalysis performed for the V5H Fuel Upgrade demonstrates that the spray line break mass and energy release rates are less than the existing releases given in UFSAR Table 6.2.1.2-1 G. Therefore, the original spray line break results bound the those obtained for the V5H Fuel Upgrade.

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The results of the original spray line break subcompartment analysis are presented in UFSAR Table 6.2.1.2-9. Table 6.2.1.2-9 states the design pressures are governed by the surge line break. The surge line break subcompartment analysis results are presented in UFSAR Table 6.2.1.2-11. Although the surge line break is no longer the design basis break, UFSAR Table 6.2.1.2-11 will be retained since it lists the subcompartment design differential pressures.

SER Section 6.2.1.2 addresses the Pressurizer Subcompartment and
Surge Line Subcompartment Analyses. Supplement 4 to SER Section 3.6.2 documents the acceptance of STP's leak-before-break lines described in this section. As indicated above, the original subcompartment design differential pressures remain bounding.

4.4.2.4.3 Radioactive Pipe Chase Subcompartment

The thermodynamic properties of the CVCS letdown line does not change due to the fuel upgrade effects. Therefore, the letdown line break mass and energy releases do not change. By incorporating the effects of the reduced containment volume, reduced containment initial temperature, and changes due to V5H fuel upgrade, the new calculated differential pressure is 1.42 psi. This differential pressure is used in structural load calculations. The results are presented in UFSAR Table 6.2.1.2-17.

The calculation which determines structural loads in the Radioactive Pipe Chase considered the differential pressure to be negligible. The slight increase calculated by using the revised containment volume does not impact that assumption. Therefore, increasing the differential pressure from 1.32 psi to 1.42 psi is still insignificant and will not impact the structural design calculations.

SER Section 6.2.1.2 states that substantial design margins exist for the Radioactive Pipe Chase Subcompartments. Since the design calculation is not impacted, the proposed change will not cause a reduction in margin of safety.

4.4.2.4.4 Regenerative Heat Exchanger Subcompartment

An estimate was made of the peak differential pressures in the regenerative heat exchanger subcompartment due to a CVCS letdown line break. As described in Section 4.4.2.4.3 above, the mass and energy releases are not affected due to fuel upgrade effects. By accounting for the effects of reduced containment volume, reduced initial temperature, and changes due to V5H fuel upgrade, the new peak differential pressure is expected to increase to 5.0 psi. However, the original design subcompartment differential pressure of 5.52 psi remains bounding. The results are presented in UFSAR Table 6.2.1.2-15.

SER Section 6.2.1.2 states that substantial design margins exist for the Regenerative Heat Exchanger Subcompartments. As indicated above, the original subcompartment design differential pressure is bounding.

4.4.2.4.5 RHR System Valve Room Subcompartment

As described in Section 4.4.2.4.3 above, the mass and energy release rates do not increase due to the fuel upgrade effort. After considering the effects of the reduced containment volume, reduced containment initial temperature, and changes due to V5H fuel upgrade, the calculated peak differential pressure increases from 1.50 psid to 1.58 psid. This negligible change in peak differential pressure does not significantly affect the design margin.

SER Section 6.2.1.2 states that substantial design margins exist for the RHR Valve Room Subcompartments. As indicated above, the original design differential pressures remain bounding. The results are presented in UFSAR Table 6.2.1.2-19.

4.4.2.4.6 Steam Generator Loop Compartments

Due to the effects of the reduced containment volume, reduced containment initial temperature, and changes due to the fuel upgrade, the peak differential pressures for subcompartments in direct contact with the containment are expected to increase by about 7.3%. UFSAR Table 6.2.1.2-5 shows that substantial margin exists. Therefore, the original design pressures remain bounding.

The peak differential pressures are used to determine forces and moments on the steam generators and the reactor coolant pumps due to pipe breaks inside the SG Loop Compartments. After considering the effects of the changes, the peak forces and moments remain bounding.

SER Section 6.2.1.2 states that substantial design margins exist for the Steam Generator Compartment Analysis. As indicated above, the original subcompartment design differential pressures remain bounding.

4.4.2.5 Containment Safety-Related Equipment Qualification

The effects of the increased accident pressures and temperatures are addressed in this section. Radiation effects on qualified equipment were addressed in STP's UFSAR change submittal for the extension of fuel burnup (Reference 16). This submittal addressed the increase in radiation doses on equipment as a result of revised source terms (due to increased fuel burnup) as well as the reduced containment volume. This UFSAR change was approved by the NRC in Reference 17.

By incorporating the reduced containment free volume, reduced containment initial temperature, and changes due to V5H fuel upgrade, the post-accident peak containment pressure and temperature have changed to 41.2 psig and 327°F. UFSAR Section 3.11 addresses equipment qualification. UFSAR Table 3.11-1 lists the environmental conditions, including containment temperatures and pressures which must be considered for equipment qualification.

The containment safety-related equipment are qualified to operate in an accident environment with pressures and temperatures equal to or higher than 57 psig* and 340°F*. The equipment qualification temperature given in Table-1 of the Equipment Qualification Design Criteria 4E019-NQ-1009 (Ref. 18) is 328°F which allows a 12°F margin to the IEEE 323-1974 (Ref. 34) testing requirements.

SSER4 Section 3.11.3 specifies that the containment safety-related equipment temperature acceptance limit is 323°F. Although the design limit decreases from 17°F to 13°F as a result of the increase in containment peak temperature from 323°F to 327°F, the time-temperature equivalency of the qualification tests adequately exceed the duration of the increased peak containment temperature. Therefore, the safety-related equipment will continue to perform their intended function with the proposed change in containment peak accident temperature from 323°F to 327°F.

SSER4 Section 3.11.3 indicates that the containment pressure acceptance limit for safety-related equipment is 48.4 psig. However, the minimum equipment qualification pressure is 57.0 psig. Therefore, the proposed change in containment peak pressure from 48.4 psig to 41.2 psig is not a reduction in the margin of safety because the margin of safety increases from 8.6 psi (57.0 - 48.4) to 15.8 psi (57.0 - 41.2). Hence, adequate margin exists.

* Note: The hydrogen recombiners, RCFC motors, Marathon terminal blocks, radiation monitors, and thermocouple junction boxes are qualified to lower values because they have no thermal or radition sensitive materials. The effects of the accident environment will not impede the safety-related performance of this equipment.

4.4.2.5.1 Electrical Cable Temperature

The maximum temperature of electrical cables inside containment occurs during the DBA MSLB. An evaluation was performed for the temperature effects associated with the reduction in containment free volume, reduction in containment initial temperature to 110°F, and changes due to the fuel upgrade. The peak cable temperature is below the environmental qualification temperature of 340°F. Therefore, the effects of the reduced containment volume and fuel upgrade do not adversely affect the electrical cable qualification temperatures.

4.4.2.5.2 Polar Crane Beam Box Pressurization

The effect of the reduced containment volume, reduced containment initial temperature, and the changes in mass and energy releases due to the fuel upgrade effort on the pressurization of the polar crane beam box have been evaluated. The containment pressurization rate used in the original calculation is more conservative than those obtained in both the revised MSLB and LOCA analyses. Therefore, the results of the original calculation remain bounding.

4.4.2.5.3 Polar Crane Temperature

To account for the reduced containment volume, reduced containment initial temperature, and changes in the mass and energy releases for the V5H fuel upgrade, the DBA LOCA and MSLB cases were reevaluated. For the revised LOCA case, the peak containment equilibrium vapor temperature after initiation of the containment sprays, is below 260°F. For the revised MSLB case, the peak containment equilibrium vapor temperature after initiation of the sprays is below 245°F. Therefore, the polar crane structural temperature will not exceed 260°F and the design structural temperature of 286°F remains bounding.

4.4.2.5.4 Containment Air-Lock Seal Temperature

The personnel air-lock inflatable seal equipment qualification (EQ) temperature was examined. After incorporating the effects of the reduced containment volume, reduced containment initial temperature, and changes in mass and energy releases due to fuel upgrade, the peak temperature for the 0.205 inch thick seal will still be lower than the 283°F determined in the original calculation. Therefore, the inflatable seal original EQ temperature of 315°F remains bounding.

4.4.2.5.5 Containment Abnormal Temperature

The containment abnormal temperature following a loss of offsite power is analyzed. For containment heatup following a fire outside the RCB, the effects of reduced containment free volume and a 10°F reduction in initial temperature result in a new peak temperature below 151°F. The changes in mass and energy releases due to the fuel upgrade do not affect this calculation. Therefore, the offsetting effects of the reduced containment volume and reduced initial containment temperature do not adversely affect the containment building design criteria or the abnormal conditions for internal equipment qualification pressure and temperature.

The maximum containment abnormal temperatures are given in UFSAR Table 3.11-1. The design abnormal temperatures presented in this table remain bounding.

4.4.2.6 Containment Leakage

After considering the effects of the reduced containment volume, a 10°F reduction in containment initial temperature, and changes due to the fuel upgrade, the peak calculated containment pressure is 41.2 psig. The Unit 1 containment was tested at 40.0 psig. Subsequent calculations indicate that if the ILRT had been conducted at 42.5 psig, the measured leak rate would have been significantly below the acceptance criteria. The Unit 2 containment was tested at 44.6 psig and the leakage rate was also significantly below the ILRT acceptance criteria. Therefore, there is no reduction in margin of safety.

Refer to Reference 3 for additional detail.

4.4.2.7 Containment Minimum Backpressure for LOCA ECCS Analysis

The reduction in containment free volume affects containment LOCA reflood rate. The reflood rate impacts the fuel peak clad temperature (PCT). For this analysis, a low containment pressure is conservative since it reduces the core reflood rate and increases PCT. The analysis assumes a containment volume of 3.56×10^6 ft^3, which is 4.4% larger than the new containment volume of 3.41×10^6 ft^3, and a containment initial temperature of 90°F. This analysis gives a conservatively low containment pressure and a higher PCT. Therefore, the original calculation is still conservative and bounding.

Refer to Reference 3 for additional detail.

4.4.2.8 Containment Hydrogen Generation

The revised analysis incorporates the reduction in containment free volume, reduced containment initial temperature, and changes due to the fuel upgrade. Results from the revised analysis show the recombiners will reach full efficiency before the hydrogen concentration reaches 3.5 volume percent. Therefore, this meets the requirements of the SRP and there is no reduction in the margin of safety.

4.4.2.9 Safety Injection/Containment Spray Operations

The effects of the reduced containment volume on pump operation have been evaluated. The results indicate that the pumps are capable of providing required flow rates under increased containment pressure conditions. Also, the effects of the reduced containment volume on net positive suction head ($NPSH_a$) have been evaluated. The results indicate that the required $NPSH_a$ is not affected by the decrease in containment free volume.

4.4.2.10 RCB Volume Change Impact on Radiological Dose Analyses

The effects of reduced containment volume on radiological doses are addressed in the subsections that follow and are based on conservative assumptions. The effects of extended burnup on source terms were previously submitted to, and approved by, the NRC (References 16 and 17).

4.4.2.10.1 Fuel Handling Accident Inside Containment

The onsite and offsite dose consequences due to a fuel handling accident inside containment are presented in UFSAR Section 15.7.4 and Table 15.7-10. The effects of the reduced containment volume have been evaluated. Per SRP 15.7.4, dated July 1981, and SER (Supplement 6) Appendix-Z, the offsite dose

consequences are acceptable within the limits of 10CFR100, or 75 rem thyroid and 6 rem whole body. The revised doses are all below these limits.

4.4.2.10.2 Control Room (CR), Technical Support Center (TSC) and Offsite LOCA Radiation Doses

The onsite and offsite dose consequences due to a LOCA are presented in UFSAR Section 15.6.5 and Tables 6.4-2, II.B.2-2, and 15.6-11. The effects of the reduced containment volume have been evaluated. The revised doses all fall below the limits set by 10CFR100 and SER Sections 15.6.5.2.5 and 6.4 (dated April, 1986).

4.4.2.10.3 Rod Ejection Accident

The onsite and offsite dose consequences due to a rod ejection accident are presented in UFSAR Table 15.4-5. The effects of the reduced containment volume have been evaluated. Per SRP 15.4.8, dated July 1981, and SER Section 15.4.8.2 (dated April, 1986), the offsite dose consequences are acceptable within the limits of 10CFR100, or 75 rem thyroid and 6 rem whole body. The revised doses all fall below these limits.

4.4.2.10.4 Plant Building Airborne Concentrations

The onsite radiological dose consequences due to airborne concentrations are addressed in UFSAR Table 12.2.2-2. The effects of the reduced containment volume have been evaluated. All concentrations are within the limits of 10CFR20.103. Therefore, the concentrations calculated are acceptable and there is no impact on the SER.

4.4.2.10.5 Reactor Coolant System Vacuum Degas System (RCSVDS) Releases to Containment

An analysis was performed to confirm that removal of the reactor head will not cause 10CFR20 limits to be exceeded. The analysis considered the reduced containment volume and the results indicated that the fraction of the maximum permissible concentration increased from 0.021 to 0.022. Thus, the change is insignificant. Therefore, the concentrations calculated are acceptable and there is no impact on the UFSAR or SER.

4.4.3 Conclusion

In summary, based upon the results of the evaluation of the current licensing basis LOCA mass and energy release analyses it was determined that for the revised uncertainty calculation conditions the current limiting mass and energy releases must be modified (see Section 4.4).

In the South Texas Safety Evaluation Report, Supplement 4, Section 3.6 (pages 3-6 and 3-8), the Commission approved the surge line and portions of the accumulator and RHR lines to use the leak-before break methodology. These lines also meet the GDC 4 criteria as specified in SSER 4, Section 3.6 (p. 3-4).

Since leak before break methodology has been applied to South Texas, the mass and energy releases contained in UFSAR remain bounding. If the analysis were performed accounting for leak before break, the calculated mass and energy release rates for the surge line break would be less than those contained in the UFSAR due to the reduced break area.

The implementation of 17x17 VANTAGE 5H fuel will have no effect on the current LOCA Short Term or Long Term Mass and Energy Release Analyses. The current licensing basis mass and energy release analyses considered a core stored energy value which bounds the proposed 17x17 VANTAGE 5H fuel design. Therefore, with respect to the subject fuel change there is no effect and the current licensing basis analyses remain valid.

4.5 Revised Instrumentation Setpoints

The Westinghouse Reactor Protection System Setpoint Study (Reference 5) provides a basis for the Reactor Protection System and Engineered Safeguards Actuation System values contained in the Technical Specifications. The proposed setpoint revision will provide adequate margin, preserving the safety analysis limit for the analyzed functions.

The methodology to determine instrument setpoints combines the independent error components for a channel both statistically and, for those errors that are dependent, using a conservative arithmetic summation technique. This methodology is consistent with earlier WCAP submittals and does not constitute a change or revision in technique. The Westinghouse standard methodology has been endorsed by the NRC. Also, various industry standards approve the use of probabilistic and statistical techniques in determining safety-related setpoints. The Westinghouse setpoint methodology has been determined to provide results in a value with 95% probability. As such, use of this methodology will not result in operation or setting of parameter outside of any previously evaluated safety limit.

Westinghouse performed formal uncertainty calculations incorporating revisions to process measurement allowance and biases attributed to insulation resistance and Veritrak transmitters. This revision of the setpoint study provides the results of the calculations and recommended Trip Setpoints. Changes to Trip Setpoints, prompted by revision to the error values used in uncertainty calculations, are identified on the marked-up Technical Specifications. The Technical Specification safety analysis limits are preserved by this revision as discussed below.

The setpoint study implements the Technical Specification revisions to optimize Trip Setpoints, within the bounds of the safety analysis limits. The Technical Specification revisions to Total Allowance are direct results of uncertainty modifications and inclusion of the Veritrak Final Report results creating new Channel Statistical Allowances for the related functions. Referring to the Technical Specification markup, "Z" values change as

a direct result of PMA modifications. Allowable values and setpoints change to accommodate the modification in overall channel statistical allowance.

The proposed Technical Specification revisions from the Westinghouse Setpoint Study are appropriate and selected to ensure adequate margin to the safety analysis limit.

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5.0 Conclusion

The proposed changes to the UFSAR and Technical Specifications, as described above, are acceptable because they do not pose a significant increase in hazard or involve a significant reduction in a margin of safety. HL&P requests approval of the proposed changes.

In addition, these proposed changes resolve the following outstanding items:

JCO-91-0049, "Containment Systems Response to DBA," Rev 2., 3/5/91.

JCO-91-0393, "Pressurizer Safety Relief Valve Loop Seal Purge Time, STPEGS Units 1 and 2," Rev 0, 11/5/91.

JCO-92-0698, "Steam Line Break Mass and Energy Releases," Rev 0, 10/2/92.

JCO-92-0020, "Veritrak Transmitters," Rev 3, 5/12/92

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6.0 References

1. "Plant Safety Evaluation for South Texas Project Units 1 and 2 Fuel Upgrade". February, 1993, FAL-93-116.

2. WCAP-13441, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology, South Texas Project Units 1 & 2," February 1993.

3. HL&P Plant Analysis, "Effects of Reduced Containment Free Volume, Reduced Initial Temperature and V5H Fuel Upgrade", February 1993.

4. Fecteau, M.W., *et al*, "Criticality Analysis of the South Texas Units 1 & 2 Fresh and In-Containment Fuel Storage Racks", Westinghouse Commercial Nuclear Fuel Division, July 1992. Attachment to Letter ST-UB-HL-1132, R.C. Cobb, WCNFD, to D.F. Hoppes, HL&P, dated July 15, 1992.

5. WCAP-11273, "Westinghouse Setpoint Methodology for Protection Systems South Texas Project Units 1 and 2," Rev. 2, February 1993.

6. LCN-1834, "Inclusion of Zirc Grids and IFBA, for Unit 2 cycle 4, and Unit 1 cycle 6 in Support of V5H Fuel Upgrade," February 23, 1993.

7. WCAP-8312-A, Rev. 2, "Westinghouse Mass and Energy Release for Containment Design", August 1975.

8. WCAP-12861, "Spray Additive Deletion Chemistry Impact to Qualification of Westinghouse Supplied Class 1E Equipment", March 1991.

9. WCAP-12909-P, Westinghouse ECCS Evaluation Model: Revised Large Break LOCA Power Distribution Methodology," May 1991.

10. WCAP 10484, Addendum 1 (Proprietary), "Spacer Grid Heat Transfer Effects During Reflood," D.L. Shimeck, December 1992.

11. SECL-87-152, "Plant Operation with Unrecovered Full Flow Filter Debris in the RCS," June 1987.

12. SECL-87-233, "Plant Operation with Unrecovered Full Flow Filter Debris in the Auxiliary Systems," June 1987.

13. "South Texas Project Electric Generating Station, Units 1 and 2 Docket Nos. STN 50-498, STN 50-499, Proposed Amendment to the Unit 1 and Unit 2 Technical Specifications", Letter ST-HL-AE-4093, from W.H. Kinsey, dated May 26, 1992, to USNRC Document Control Desk.

14. "Issuance of Amendment Nos. 43 and 32 to Facility Operating License Nos. NPF-76 and NPF-80 - South Texas Project, Units 1 and 2 (TAC Nos. M83416 and M83417)", Letter from G.F. Dick, Jr., USNRC, to D.P. Hall, HL&P, dated August 25, 1992.

15. "South Texas Project Units 1 and 2, Docket Nos. STN 50-498; 50-499, Proposed Licensing Amendment Concerning the RWST and SI Accumulators Allowable Boron Concentration Ranges", Letter ST-HL-AE-4281 from S.L. Rosen, dated January 14, 1993, to USNRC Document Control Desk.

16. "South Texas Project Electric Generating Station, Units 1 and 2 Docket Nos. STN 50-498, STN 50-499, Proposed Revision to Updated Safety Analysis Report for Extended Burnup Fuel", Letter ST-HL-AE-3906, from S.L. Rosen, dated October 30, 1991, to USNRC Document Control Desk.

17. "Issuance of Amendment Nos. 38 and 29 to Facility Operating License Nos. NPF-76 and NPF-80 - South Texas Project, Units 1 and 2 (TAC Nos. M82128 and M82129)", Letter from G.F. Dick, Jr., USNRC, to D.P. Hall, HL&P, dated June 8, 1992.

18. Design Criteria 4E019-NQ-1009 "Equipment Qualification," Revision 10.

19. NUREG-0781, Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2, including Supplements 1 through 7; Sections 3.11, and 6.2.

20. NUREG-1346, Technical Specifications, South Texas Project, Unit Nos. 1 and 2, Docket Nos. 50-498 and 50-499, Appendix "A" to License Nos. NPF-76 and NPF-80, March 1989.

21. NUREG-0800 Standard Review Plan, Section 6.2., Revision 2, July 1981.

22. WCAP-8339, "Westinghouse ECCS Evaluation Model - Summary," F.M. Bordelon, H.W. Massie, and T.A. Zordan, July 1974.

23. WCAP-12910, "Pressurizer Safety Valve Set Pressure Shift," Barret, G.O., *et al*, March 1991.

24. WCAP-12369, "LOFTR2 Analysis for a Steam Generator Tube Rupture for the South Texas Project Units 1 and 2," D.F. Holderbaum, R.N. Lewis, K. Rubin, September 1989.

25. WCAP-12833, "South Texas Units 1 & 2 Thot Reduction Phase 1 Licensing Report," July 1991.

26. ANSI/ANS 56.4-1983, "Pressure and Temperature Analysis for Light Water Reactor Containments"

27. Trich, S.R., "Methodology Clarifications to WCAP-12909-P, ET-NRC-91-3633, November 21, 1991.

28. "South Texas Project Electric Generating Station, Units 1 & 2 Docket Nos. STN 50-498, STN 50-499, Proposed Revision to Updated Safety Analysis Report to Reallocate Margin to Address Large Break LOCA Analysis, Initial Containment Temperature Assumptions," Letter ST-HL-AE-4211, from S.L. Rosen, dated October 12, 1992, to USNRC Document Control Desk.

29. [not used]

30. Letter ST-HL-AE-4066, "Safety Analysis Report Concerning Containment Volume", and attached License Change Amendment, August 4, 1992.

31. Letter NS-NRC-89-3466, from W. J. Johnson (W) to R. C. Jones (NRC), October 1989, Subject: Use of 2700°F PCT acceptance limit in non-LOCA accidents.

32. Hargrove, H. G., "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909 (non-proprietary), October 1972.

33. LOFTRAN, WCAP-7907-P-A

34. IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

ATTACHMENT 1

No Significant Hazards Evaluation
for the
Proposed Fuel Upgrade

No Significant Hazards Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

Summary

The purpose of this proposed license amendment is to upgrade the fuel used in the South Texas Project to Westinghouse's VANTAGE 5 Hybrid (V5H) design and implement several analytical and operational upgrades into the South Texas Project Updated Final Safety Analysis Report (UFSAR) and the Technical Specifications.

The following constitute major modifications proposed to the South Texas Project licensing bases:

1. Mechanical Fuel Upgrade to V5H;
2. Upgraded Safety Analysis;
3. Increase in the Maximum Allowable Fuel Enrichment;
4. Revised Reactor Containment Building Volume.

This opportunity is also being taken to address miscellaneous editorial changes to the UFSAR.

Description of Changes

The first V5H upgrade fuel is planned for Unit 1 Cycle 6 and Unit 2 Cycle 4. The mechanical fuel changes for V5H fuel, and the associated UFSAR changes, to be used in Unit 2 Cycle 4 have been approved by Houston Lighting & Power via the internal 10CFR50.59 review process. Descriptions of these changes are included in this document for completeness. It is planned that the safety analysis changes, and associated setpoint changes, will be effective on Unit 1 at the beginning of Cycle 6 and during mid-cycle 4 on Unit 2.

Approval of this submittal package is sought to (a) improve fuel utilization and economy; (b) resolve the Veritrak instrumentation issue; and, (c) incorporate the revised containment building volume.

The specific features of V5H which represent a change from standard fuel are:

- Zircaloy grids - The Inconel structural grids used in standard fuel are replaced by Zircaloy grids in V5H (except for the top and bottom grids which remain Inconel).

- Integral Fuel Burnable Absorbers (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 pellets. IFBAs provide peaking factor and moderator temperature coefficient control.

Reconstitutable Top Nozzle (RTN) and Debris Filter Bottom Nozzle (DFBN) are features that have been used extensively in Westinghouse designs and are currently in use at STP.

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The following are changes which affect the safety analysis.

- o V5H fuel with zircaloy mid-grids and Integral Fuel Burnable Absorbers (IFBA)
- o Increased Peaking Factor Allowances
- o RCS Average Temperature Range (allow vessel average temperature of 593°F to 582.3°F)
- o Revised Thermal Design Procedure (RTDP)
- o Positive Moderator Temperature Coefficient (PMTC)
- o Shutdown Margin Reduction from 1.75%ΔK/K to 1.3%ΔK/K
- o Modified Overtemperature and Overpower ΔT
- o 10% Steam Generator Tube Plugging
- o Added Tolerance for Pressurizer Safety Relief Valve Drift and Loop Seal Purge Time
- o Added Tolerance for Steamline Safety Relief Valve Drift
- o Steamline Break Mass and Energy Release Inside Containment
- o Increased Maximum Allowable Fuel Enrichment
- o Reduced Auxiliary Feedwater Flow

Previously submitted revisions to the Technical Specifications (Reference 13):

- o RWST boron concentration increase - the boron concentration maintained in the PWST will increase to 2800 ppm (min) to 3000 ppm (max).
- o Safety Injection Accumulator boron concentration increase - the boron concentration in the accumulators will increase to 2700 ppm (min) to 3000 ppm (max).
- o Boric acid storage tank volume increase - the volume maintained in the boric acid storage tank will be increased from 2900 gallons to 3200 gallons.

A detailed discussion on the effects of these changes can be found in Section 5 of the Plant Safety Evaluation by Westinghouse (Reference 1). A summary of each change is presented below.

V5H Fuel with Zircaloy Mid-grids and Integral Fuel Burnable Absorbers (IFBA)

Currently, the South Texas units use Westinghouse 17x17 XL Standard fuel. The V5H fuel has the same rod outer diameter but has zircaloy mid-grids. The only direct effect of the use of this fuel type on the non-LOCA safety analysis is the potential for a change in the rod drop time from gripper release to dashpot entry. For South Texas, the current licensed rod drop time of 2.8 seconds has been shown to be sufficiently conservative so as to remain bounding even with V5H fuel and zircaloy mid-grids. Therefore, there is no need for a safety analysis rod drop time assumption change.

There may be some indirect effects of the V5H fuel and the use of IFBAs in the core design which cannot be separated from the types of changes which typically occur at the time of a fuel reload design. Any indirect impact is incorporated into all of the non-LOCA safety analyses via the assumed physics parameters and verified at the time of the reload safety evaluation. There is no impact to the LOCA analyses.

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Increased Peaking Factors

The peaking factors justified by the non-LOCA safety analyses include an $F_{\Delta H}$ of 1.62 and an F_Q of 2.7. The peaking factors are explicitly assumed in the generation of some non-LOCA safety analysis input. A value for $F_{\Delta H}$ is assumed in the calculation of the core thermal limits which in turn are used to generate the Overtemperature ΔT and Overpower ΔT reactor trip setpoints. Also, this parameter is directly used in the calculation of input to the Partial Loss of Flow (UFSAR Section 15.3.1), Complete Loss of Flow (UFSAR Section 15.3.2) and Locked Rotor (UFSAR Sections 15.3.3 and 15.3.4) analyses. F_Q is used explicitly to calculate input for the Rod Ejection (UFSAR Section 15.4.8) analysis.

There may be some indirect effects of the increased peaking factors which can not be separated from the types of changes which are typically seen at the time of a fuel design. Any indirect impact is incorporated into all of the non-LOCA safety analysis via the assumed physics parameters and verified at the time of the reload safety evaluation.

The revised safety analysis predicts that once burned Standard fuel would have a zirc/oxide thickness over the 17% limit at the higher proposed peaking factors of F_Q=2.7 and $F_{\Delta H}$=1.62, and this would reduce the margin to safety. Therefore, the $F_{\Delta H}$ is reduced to 1.55 for the Standard fuel to obtain acceptable results. F_Q remains at 2.7 for both fuel types. STP will, therefore, carry dual Technical Specifications (presented in the COLR) for $F_{\Delta H}$: one for Standard fuel and one for V5H fuel.

RCS Average Temperature Range

A range of RCS Average Temperatures (vessel average temperature from 582.3 to 593.0°F at full power) was incorporated into the safety analyses. This technique will give South Texas the flexibility to choose the RCS full power temperature at the beginning of a fuel cycle design and implement the programmed value in that cycle without making further changes to the licensing basis safety analyses.

Table 1 indicates the vessel average temperature assumption used in the various non-LOCA accident analyses. In some cases, it was necessary to analyze the event at the high and low end of the temperature range to determine a limiting condition. For others, existing sensitivities were cited as justification for choosing one temperature or the other. Events which primarily examine the margin to DNB were analyzed assuming a vessel average temperature of 593.0°F because high temperatures are more conservative for DNB calculations.

The results of all of the non-LOCA safety analyses were acceptable with respect to the event-specific criteria when analyzed at the limiting temperature condition. Therefore, any RCS vessel average temperature between 593.0 and 582.3°F will be acceptable, so long as each fuel design considers the chosen operating temperature for the cycle. The LOCA analyses show that the limiting conditions remain with the higher RCS average temperature of 593°F.

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Revised Thermal Design Procedure

The thermal design procedure currently used in the justification of the DNB design basis for South Texas is referred to as Standard Thermal Design Procedure (STDP). With the fuel upgrade, the Revised Thermal Design Procedure (RTDP) methods for calculating the DNB design basis are used for selected transients (Reference 2). The uncertainties in initial conditions for non-LOCA transients are handled differently, depending on whether RTDP or STDP is used in the determination of the safety analysis limit DNBR.

For events which focus primarily on DNB, the following applies:

If STDP is used, instrument uncertainties are applied to the initial condition assumptions for power, pressure, flow and temperature in the conservative direction with respect to DNBR calculations. (An example is the analysis of the Startup of an Inactive Loop (UFSAR Section 15.4.4). While this event is primarily concerned with ensuring that the DNB design basis is met, the initial reactor coolant flow is below the range for which RTDP uncertainties have been defined. Therefore, Standard Thermal Design Procedure is used for this event.)

If RTDP is used, only instrument biases are applied to the transient initial conditions, in the conservative direction with respect to DNBR calculations. Instrument uncertainties are statistically combined during the calculation of the safety analysis limit DNBR. RTDP is not used in analyses at zero power because the power levels reached during the transient are insufficient to apply the DNB correlations used in conjunction with RTDP. Further, the uncertainties on power and RCS average temperature used in the development of the safety analysis limit DNBR are not appropriate at zero power.

For events which focus on criteria other than DNB, such as pressurizer overfill or saturation in the hot legs, the instrument uncertainties for pressure, power, temperature and flow were applied to the initial conditions as is currently done in the South Texas UFSAR (i.e. RTDP method was not applied).

DNBR limits are revised to reflect the reallocation of uncertainties. See Table 1 for a separation of RTDP and STDP transients.

The analyses completed for the upgrade justify the following instrument uncertainties:

Power:	±2% Calorimetric error
Temperature:	±6.25°F [includes a bias of 0.25°F] (See Note 1)
Pressure:	±46 psi [includes a bias of 13 psi] (See Note 2)

Thermal Design Flow: 381,600 gpm
Minimum Measured Flow: 392,300 gpm (See Note 3)

Notes:

1) DNB margin has been allocated to cover an additional 0.55°F bias on temperature

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2) The Locked Rotor Pressure transient used an uncertainty of 46 psi. The Feedline Break analysis used an uncertainty of 50 psi. All other analyses used a pressurizer pressure uncertainty of +63 psi.

3) The Minimum Measured Flow requirement includes a measurement uncertainty of 2.7% plus an additional 0.1% allowance for venturi fouling. Total: 2.8%. The calibration requirements for the Special Test Equipment have been removed from the Technical Specifications, since their specifications are mandated by the RTDP document and will be controlled administratively. Also, the <75% precision flow test has been deleted since its requirements have been determined to be met by the 12 hour flow surveillance. (Surveillance 4.2.5.1 and 4.2.5.2 are affected. The changes do not represent a reduction in safety or adversely affect any analyzed accident scenarios.)

Positive Moderator Temperature Coefficient (PMTC)

A Positive moderator temperature coefficient (PMTC) specification of +5 pcm/F from 0% rated thermal power (RTP) to 70% RTP and a linear ramp from +5 pcm/F at 70% RTP to 0 pcm/F at 100% RTP is proposed.

PMTC may be used in future core designs to improve fuel economy and design flexibility. Table 1 shows those non-LOCA analyses in which a positive MTC was assumed. Generally, a PMTC is assumed in transients which result in a heatup of the primary system prior to reactor trip. A PMTC accentuates the heatup by adding positive reactivity. Generic sensitivities and PMTC safety evaluations for other Westinghouse PWRs were used as justification to determine in which analyses a PMTC should be assumed.

For conservatism and to bound the transients at all operating conditions, PMTC was assumed up to full power. The Locked Rotor (UFSAR Section 15.3.3) pressure transient is an exception. In order to obtain acceptable results, it was necessary to run the analysis at full power assuming a 0 MTC. An additional case was conservatively examined at 90% RTP to justify the PMTC of +5 pcm/°F. The Locked Rotor rods in DNB case was also examined at a full power PMTC of +5 pcm/°F and found to have acceptable results.

All analyses demonstrated that the presence of a PMTC did not cause a violation of the event-specific safety analysis criteria. Therefore, it is acceptable to incorporate this change into future core designs, as needed.

STP has elected not to incorporate PMTC at this time (but may do so in the future) and this is reflected in the COLR.

Shutdown Margin Reduction from 1.75% $\Delta k/k$ to 1.3% $\Delta k/k$

Table 1 lists those transients in which shutdown margin is assumed as an explicit input to the non-LOCA analyses. The results of each of these transients met the event-specific criteria therefore, the reduced shutdown margin is acceptable. The post-LOCA shutdown analysis also shows the reduction in shutdown margin to be acceptable.

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Modified Overtemperature and Overpower ΔT Setpoints

The Overtemperature ΔT setpoints were revised because the core thermal limits were changed to reflect the use of RTDP. The Overpower ΔT K_6 was changed to reflect the Veritrak component only. The setpoints were determined using the same methods as currently used in the UFSAR. (Refer to Table 2 for those events in which the safety analysis predicted that Overtemperature ΔT actuated for reactor protection.) A full power steamline break analysis was completed to ensure that the Overpower ΔT setpoints remained acceptable. The event-specific criteria were met for each of the analyses in Table 2, therefore, the revised setpoints are acceptable.

10% Steam Generator Tube Plugging

The analyses completed for the fuel upgrade considered the most limiting steam generator tube plugging condition. For cooldown events, it is generally more conservative to assume 0% steam generator tube plugging so that the heat transfer area is maximized and the cooldown is accentuated. Conversely, in an event which results in a heatup of the primary system, typically the highest level of steam generator tube plugging is assumed so that the heat transfer area is minimized and the heatup accentuated. It should be noted that it is expected that the higher level of tube plugging will not cause the reactor coolant flow to decrease below current design values. Therefore, in the analyses which model 10% steam generator tube plugging, it was not necessary to reduce the reactor coolant flow assumption.

Table 1 identifies the transients which assumed 10% steam generator tube plugging and those which assumed 0%. As with the previous assumptions, all transients met the event-specific criteria, therefore, with respect to the non-LOCA and LOCA accident analyses, it is acceptable to plug up to 10% of the steam generator tubes.

Added Tolerance for Pressurizer Safety Relief Valve Drift and Loop Seal Purge Time

The non-LOCA safety analysis justified a pressurizer safety valve tolerance of $\pm$1% (ASME code setting) and $\pm$1% for drift during operation.

WCAP-12910 (Reference 6) examines the effect of a loop seal on the relief characteristics of the pressurizer safety valves. The pressurizer safety valves, when set with a loop seal as opposed to steam, will have three effects which should be considered in the safety analysis. In addition to the ASME code tolerance, for STP, a shift is assumed to be 1% of the set pressure to account for drift. Also, the loop seal must be purged from the valve before primary relief can occur. A pressurizer loop seal purge time of 1.12 seconds was assumed in the non-LOCA accident analysis. In addition, a 3% accumulation was also assumed.

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The non-LOCA analyses considered the effect of the loop seal and the increased tolerance where appropriate. The transients which primarily examine peak RCS pressure include the Loss of Load/Turbine Trip (UFSAR 15.2.2/15.2.3) event and the Locked Rotor (UFSAR 15.3.3) event. In these events, the effects of the loop seal and the increased tolerance were explicitly modeled. The results of these analyses and the evaluations of the remaining non-LOCA transients demonstrate that the pressurizer safety valves will provide sufficient relief even when a loop seal and an analytical tolerance totalling $\pm$2% is assumed.

Added Tolerance for Steamline Safety Relief Valve Drift

Secondary pressure must remain below 110% of the steam generator shell design pressure during non-LOCA transients. Steamline safety valves ensure that this limit is met during a non-LOCA transient. Currently, the tolerance on the steam generator safety valves is $\pm$1% of the set pressure (ASME code tolerance only). In the analyses for the V5H fuel upgrade, an additional 2% was added for drift with an additional 3% for accumulation. The safety analyses took credit for the staggered pressure setpoints currently in the Technical Specifications. In previous analyses, the most limiting setpoint was assumed for all valves.

This steamline safety valve model was included in the analysis of the Loss of Normal Feedwater\Loss of Offsite Power analysis as well as in the Feedwater Line Break transient. The effects of reduced Auxiliary Feedwater requirements were evaluated at the higher relief valve settings and found to be acceptable.

Steamline Break Mass and Energy Release Inside Containment

In addition to the re-analysis of the Chapter 15 non-LOCA transients, the Steamline Break Mass and Energy releases for use in a containment pressure and temperature analysis were calculated. V5H fuel and the associated plant changes were included in those calculations.

Increased Maximum Allowable Fuel Enrichment

The proposed changes include an increase in the maximum nominal enrichment for fuel assemblies from 4.5 weight percent (w/o) uranium-235 to 5.0 w/o.

The New Fuel Racks and In-Containment Fuel Storage Racks were reanalyzed to allow storage of Westinghouse 17x17 XL fuel assemblies with enrichments up to 5.0 w/o uranium-235. The Spent Fuel Storage Racks were previously approved for storage of assemblies with enrichments up to 5.0 w/o uranium-235 (References 7 and 8). The New Fuel and In-Containment Fuel Storage Rack criticality analyses are based on maintaining K_{eff} less than or equal to 0.95 for storage of Westinghouse 17x17 fuel assemblies under full water density conditions and less than or equal to 0.98 under low water density (optimum moderation) conditions. The optimum moderation condition applies only to the fresh fuel rack, since this rack is used to store fuel in a dry configuration.

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Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. For the New Fuel and In-Containment racks, this is accomplished by fixing the minimum separation between fuel assemblies. The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95% probability at a 95% confidence level that the effective neutron multiplication factor, K_{eff}, of the fuel assembly array will be less than 0.95, as recommended by ANSI 57.2-1983, ANSI 57.3-1983, and the NRC letter to power reactor licensees, "Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978. The 0.95 K_{eff} limit applies to both the New Fuel and In-Containment fuel racks under all conditions, except for the New Fuel Storage Rack under low water density (optimum moderation) conditions, where the K_{eff} limit is 0.98, as recommended by NUREG-0800 (Reference 9).

The criticality analyses, including IFBA Credit Reactivity Equivalencing and a review of postulated accidents that would increase reactivity, concluded that the acceptance criteria for criticality is met for the New Fuel and In-Containment Fuel Storage Racks for the storage of all Westinghouse 17x17 fuel assemblies with the following configurations and enrichment limits:

New Fuel Rack	Storage of fuel assemblies with nominal enrichments up to 5.0 w/o in any location. There are no requirements on position or minimum IFBA for these assemblies.
In-Containment Rack	Storage of fuel assemblies with nominal enrichments up to 4.5 w/o in any rack location. Fuel assemblies with enrichments above 4.5 w/o can also be stored, but each assembly must contain sufficient Integral Fuel Burnable Absorbers (IFBA) to satisfy the requirements specified in the criticality report.

The criticality analysis report for the New Fuel and In-Containment Storage Racks is found in Reference 4. The reanalysis of the New and In-Containment fuel racks criticality analysis does not compromise the performance of any safety-related components or system.

The impact of the increased fuel maximum enrichment on the radiological consequences of accidents has been evaluated taking into account the changes discussed in Section 3.0 and PSE Section 5.0 (Reference 1). A discussion of this evaluation is presented in PSE Section 5.4. The evaluation concludes that there would be no adverse impact on the consequences. Thus, the doses reported in the UFSAR remain bounding. An evaluation of the impact of increased discharge burnup on radiological source terms was previously submitted to the USNRC, and subsequently approved, in References 10 and 11, respectively.

Pursuant to 10CFR50.91, this analysis provides a determination that the proposed changes to the Technical Specifications do not involve significant hazards considerations as defined in 10CFR50.92.

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TABLE 1

SAFETY ANALYSIS ASSUMPTIONS

UFSAR SECTION		Full Power Vessel Avg. Temp.	PMTC	RTDP	% SGTP	SDM
15.1.2	Feedwater Malfunction	593.0	No	Yes	0%	NA
15.1.3	Excess Increase in Steam Flow	593.0	No	Yes	0%	NA
15.1.4	Main Steam Depressurization	593.0	No	No	0%	1.3
15.1.5	Steamline Break (Core Response)	593.0	No	No	0%	1.3
15.2.2 15.2.3	Loss of Load/ Turbine Trip	593.0	Yes	Yes	10%	NA
15.2.6 15.2.7	Loss of Offsite Power Loss of Normal Feedwater/LOOP	582.3/ 593.0	Yes	No	10%	NA
15.2.8	Feedwater System Pipe Break	582.3/ 593.0	No	No	10%	1.3
15.3.1	Complete Loss of Flow	593.0	Yes	Yes	10%	NA
15.3.2	Partial Loss of Flow	593.0	Yes	Yes	10%	NA
15.3.3 15.3.4	Locked Rotor Shaft Break	593.0	No[1]	No	10%	NA*
15.4.1	Rod Withdrawal from Subcritical	NA	Yes	No	NA	NA
15.4.2	Rod Withdrawal at Power	593.0	Yes	Yes	10%	NA
15.4.4	Startup of an Inactive Coolant Loop	593.0	Yes	No	10%	NA
15.4.6	Boron Dilution	593.0	NA	NA	NA	1.3
15.4.8	RCCA Ejection	NA	Yes	No	NA	NA

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UFSAR SECTION		Full Power Vessel Avg. Temp.	PMTC	RTDP	% SGTP	SDM
15.5.2	CVCS Malfunction	582.3/ 593.0	Yes	No	10%	NA
15.6.1	RCS Depressurization	593.0	Yes	Yes	10%	NA
15.6.3	Steam Generator Tube Rupture	582.3/ 593.0	Yes	NA	15%[2]	NA
15.6.5	Large Break and Small Break LOCA	593.0	NA	NA	10%	NA
15.6.5	Long Term Cooling Post-LOCA Shutdown	NA	NA	NA	NA	NA
15.6.5	Long Term Cooling Boron Precipitation	NA	NA	NA	NA	NA

* Table 1 reflects pressure transient assumptions.

[1] PMTC = 0 at 100% RTP, PMTC=+5 pcm/°F at 90% RTP. (PMTC was utilized for Rods-In-DNB analysis.)

[2] 15% tube plugging was assumed for margin to overfill analysis and 0% tube plugging was assumed for offsite does analysis.

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TABLE 2

NON-LOCA TRANSIENTS IN WHICH
OVERTEMPERATURE DELTA-T AND OVERPOWER DELTA-T
REACTOR TRIP SETPOINTS MAY ACTUATE

UFSAR SECTION	TITLE
15.2.3	Loss of External Electrical Load
15.4.2	Uncontrolled RCCA Bank Withdrawal at Power
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve
NA	Steamline Break at Power

NOTE: The Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (UFSAR Section 15.4.6) event uses the results from a case of the Uncontrolled RCCA Bank Withdrawal at Power. That case may trip on Overtemperature ΔT.

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As required by 10CFR50.92, the following analysis of the issue of no significant hazards consideration is presented:

(1) **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

Each proposed change has been evaluated with respect to the above question. The evaluations are presented below.

The changes to the Updated Safety Analysis Report (UFSAR) due to the mechanical changes to the fuel have been reviewed internally by HL&P, per 10CFR50.59, have been found not to constitute an unreviewed safety question and, therefore, do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The only direct effect of these fuel mechanical changes on the non-LOCA safety analyses is the potential for a change in the rod drop time (from gripper release to dashpot entry). The currently licensed rod drop time of 2.8 seconds has been shown to be sufficiently conservative to remain bounding even with V5H fuel and zircaloy mid-grids.

The STP analyses supporting this amendment request have incorporated the following changes to the methodology/model used in the previous licensing basis:

1) Revised Thermal Design Procedure
2) 10% steam generator tube plugging
3) Increased allowable range for T_{AVE}
4) Reduced AFW flow rate (Technical Specification (TS) 3/4.7.12)
5) Revised reactor containment building volume and P_a (TS 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 3/4.6.1.5, 5.2.2)
6) Increased core peaking factor allowance in COLR
7) Positive moderator temperature coefficient (TS Fig. 3.1-2a)
8) Tolerance for pressurizer safety valve drift and loop seal purge time
9) Tolerance for steam line safety valve drift
10) Revised DNB parameters (TS 3/4.2.5)
11) Shutdown margin reduction from 1.75%Δk/k to 1.3%Δk/k (Fig. 3.1-1, 3.1-2)

The proposed changes to page viii of the index and the Section 5.2.2 are editorial for consistency and clarity. They have no impact on any of the 10CFR50.92 criteria and involve no significant hazards consideration.

Chapter 15 accidents have been analyzed considering the cumulative effects of all proposed changes, including revised setpoints described in the proposed changes to Technical Specification Tables 2.2-1 and 3.3-4. No significant increase in consequences were identified in the analyses other than small increases in the PCT results for certain Condition IV events (LOCA and non-LOCA). In any case, these results were found to be within the limits of 10CFR50.46.

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Accident probability is influenced by changes which increase the frequency of initiating events. The changes described above are analytical revisions and do not involve system modifications other than setpoint changes. The changes are expected to benefit operational flexibility overall and, in the case of OTΔT setpoint changes, reduce the potential for reactor trip. Consequently, HL&P has concluded that there is no significant increase in accident probability.

In addition to the re-analysis of the Chapter 15 non-LOCA transients, the Steamline Break Mass and Energy releases for use in a containment pressure and temperature analysis were calculated. V5H fuel and the associated plant changes were included in those calculations.

The containment and safety-related systems inside containment remain operable as previously analyzed. The changes in the containment volume, pressure, and temperature do not increase or cause an increase in the likelihood of a LOCA, MSLB, or any other DBA. The increase in peak containment post accident pressure and temperature resulting from the reduced free volume is bounded by the original containment and equipment design pressure (Reference 3).

Criticality analyses, including IFBA Credit Reactivity Equivalencing and a review of postulated accidents that would increase reactivity, concluded that the acceptance criteria for criticality is met for the New Fuel and In-Containment Fuel Storage Racks for storage of all Westinghouse 17x17 fuel assemblies with the specified configurations and enrichment limits. Reanalysis of the New and In-Containment fuel racks criticality analysis does not compromise the performance of any safety-related components or system.

The licensing basis of maintaining a K_{eff} of less than or equal to 0.95 is met by the physical design of both the New Fuel racks and the In-containment storage racks and by the use of administrative controls. Based upon SER Supplement 6 (Reference 12) which presents the NRC acceptance criteria, the proposed changes to the UFSAR and to Section 5.6 of the Technical Specifications meet the accepted NRC acceptance criteria for rack subcriticality. Therefore, the changes in the utilization of the New Fuel racks and the In-containment Fuel racks do not involve a significant increase in the probability or consequences of an accident previously evaluated.

No new performance requirements are being imposed on any system or component in order to support the revised analysis assumptions. Subsequently, overall plant integrity remains consistent with that established by the original licensing basis. Furthermore, the features of V5H which are different from Standard fuel and the associated fuel upgrade related changes included as modifications to the safety analysis procedures or input are associated with features used as limits or mitigators to assumed accident scenarios and not accident initiators.

The evaluation of the impact of the proposed changes specified in Section 3.0 of the Safety Evaluation is discussed in Section 4 of the Safety Evaluation and the Plant Safety Evaluation (PSE) Sections 5.4 and 5.6 (Reference 1). The evaluation addressed a full core of V5H as well as transition cores consisting of V5H and Standard fuel.

Therefore, the probability of occurrence of an accident previously evaluated in the safety analysis report is not increased, and there is no significant increase in the consequences of an accident previously evaluated in the safety analysis report.

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(2) The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

As noted below, each of the elements of the revised analyses for the proposed Technical Specification changes has been reviewed and determined to be acceptable. These analytical changes do not impact the way systems are operated or introduce new or different failure mechanisms.

The accident parameters which have a direct impact on dose (including fuel damage, offsite steam releases and primary to secondary leakage) have been reviewed. The results of the accident analysis show that these parameters are not adversely affected by the upgrade to V5H and the associated changes. Since these parameters are not adversely affected, the radiological consequences of the accidents do not create the possibility of a different accident or malfunction of equipment important to safety.

Mechanical evaluations have been performed on equipment important to safety to confirm that their function and reliability are not negatively impacted due to the upgrade to V5H and the additional changes discussed in Section 3.0 of the Safety Evaluation. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the fuel transition. The presence of V5H fuel assemblies in the core or the revised analytical assumptions have no adverse effect and do not challenge the performance of any other safety related system.

All safety related equipment remain operable and will continue to perform their functions under the conditions of the new analyses, and , therefore, the probability of failure of such equipment initiating an accident is not increased.

No new performance requirements are being imposed on any system or component in order to support the revised analysis assumptions. Subsequently, overall plant integrity remains consistent with that established by the original licensing basis. Furthermore, the features of V5H which are different from Standard fuel and the associated fuel upgrade related changes included as modifications to the safety analysis procedures or input (see Section 3.0 of the Safety Evaluation) are associated with features used as limits or mitigators to assumed accident scenarios and not accident initiators. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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(3) The proposed change does not involve a significant reduction in a margin of safety.

The upgrade to V5H fuel, use of RTDP and the other changes discussed in this application were evaluated against the applicable acceptance criteria.

1) Fuel-related criteria:
   a) DNBR greater than safety analysis limit
   b) PCT less than 2200°F for LOCA
   c) Fuel centerline temperature less than 4900°F (BOL), 4800°F (EOL)
   d) Average fuel pellet enthalpy less than 200 cal/gm for rod ejection
   e) Fuel melting limited to 10 percent for rod ejection
   f) Remainder of 10CFR50.46 criteria (clad oxidation, hydrogen generation, coolable geometry, long-term cooling)

2) RCS pressure boundary-related criteria:
   a) Pressure less than 110 percent for Condition II and III events
   b) Pressure less than 116 percent for Condition IV events

3) Containment pressure:
   a) Peak pressure maintained below design pressure, and long-term pressure below 50% of design within 24 hours

DNB margin is maintained by the use of the revised thermal design procedure. The non-LOCA analyses confirm that the DNB design basis is met for Standard 17x17 and V5H fuel.

LOCA-related analyses demonstrate that the margins of safety with respect to blowdown reactor vessel and loop forces is preserved, thus satisfying the 10CFR50.46 criteria that the core remain amenable to cooling after a LOCA. Long-term cooling and post-LOCA subcriticality concerns are satisfied by the increased RWST boron concentration. Increased RWST boron concentration, in turn, affects the concern with boron precipitation. The analysis shows hot-leg switchover must be accomplished 10.5 hours after accident initiation, which is acceptable.

The containment and safety-related systems inside containment remain operable as previously analyzed. The increase in peak containment post accident pressure and temperature resulting from the reduced free volume is bounded by the original containment and equipment design pressure.

The margin of safety in the plant licensing basis which is affected by the upgrade to V5H fuel and associated changes discussed in Section 3.0 of the Safety Evaluation is defined in the BASES to those technical specification. These BASES and the supporting technical specification values are defined by the accident analyses which are performed to conservatively bound the operating basis defined by the technical specifications and to demonstrate compliance with the regulatory acceptance limits.

No Significant Hazards Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

The upgrade to V5H fuel and the other changes discussed in Section 3.0 of the Safety Evaluation were evaluated against the applicable acceptance criteria and determined to be acceptable.
Performance of analyses and evaluations for the upgrade to V5H and associated changes have confirmed that the operating envelope defined by the technical specifications continues to be bounded by the revised analytical basis, which in no case exceeds the acceptance limits.

Therefore, the margin of safety provided by the analyses in accordance with these acceptance limits is maintained and not reduced.

No Significant Hazards Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

Conclusion

The Commission has provided guidance concerning the application of the standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. Example (iv) provides that a significant hazards consideration finding is unlikely for:

> A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the changes are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from a small refinement of a previously used calculation model or design method.

This example appears applicable to the portion of the proposed license amendment request involving a change in the fuel design and improved analytical methodology. As discussed above, for non-LOCA accidents, the consequences have slightly increased. However, they are within the applicable acceptance criteria. Similarly, for LOCA accidents, the proposed changes result in an increase in the peak clad temperature (PCT); however, in all cases, they remain within the acceptance criteria for PCT. For a fuel-handling accident, the increase in the consequences is not significant. Therefore, the proposed changes do not involve a significant hazards consideration. For other Technical Specification changes, there is no increase in the consequences of an accident previously analyzed.

Therefore, based on the above, HL&P concludes the proposed Technical Specification changes do not involve a significant hazards consideration. HL&P, therefore, requests approval of the proposed changes.

No Significant Hazards Evaluation
for the
Proposed VANTAGE 5H Fuel Upgrade

References

1. "Plant Safety Evaluation for South Texas Project Units 1 and 2 Fuel Upgrade". February, 1993, FAL-93-116.

2. WCAP-13441, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology, South Texas Project Units 1 & 2," February 1993.

3. HL&P Plant Analysis, "Effects of Reduced Containment Free Volume, Reduced Initial Temperature and V5H Fuel Upgrade", February 1993.

4. Fecteau, M.W., *et al*, "Criticality Analysis of the South Texas Units 1 & 2 Fresh and In-Containment Fuel Storage Racks", Westinghouse Commercial Nuclear Fuel Division, July 1992. Attachment to Letter ST-UB-HL-1132, R.C. Cobb, WCNFD, to D.F. Hoppes, HL&P, dated July 15, 1992.

5. [not used]

6. WCAP-12910, "Pressurizer Safety Valve Set Pressure Shift," Barret, G.O., *et al*, March 1991.

7. "South Texas Project Electric Generating Station, Units 1 and 2 Docket Nos. STN 50-498, STN 50-499, Proposed Amendment to the Unit 1 and Unit 2 Technical Specifications", Letter ST-HL-AE-4093, from W.H. Kinsey, dated May 26, 1992, to USNRC Document Control Desk.

8. "Issuance of Amendment Nos. 43 and 32 to Facility Operating License Nos. NPF-76 and NPF-80 - South Texas Project, Units 1 and 2 (TAC Nos. M83416 and M83417)", Letter from G.F. Dick, Jr., USNRC, to D.P. Hall, HL&P, dated August 25, 1992.

9. NUREG-0800 Standard Review Plan, Section 6.2., Revision 2, July 1981.

10. "South Texas Project Electric Generating Station, Units 1 and 2 Docket Nos. STN 50-498, STN 50-499, Proposed Revision to Updated Safety Analysis Report for Extended Burnup Fuel", Letter ST-HL-AE-3906, from S.L. Rosen, dated October 30, 1991, to USNRC Document Control Desk.

11. "Issuance of Amendment Nos. 38 and 29 to Facility Operating License Nos. NPF-76 and NPF-80 - South Texas Project, Units 1 and 2 (TAC Nos. M82128 and M82129)", Letter from G.F. Dick, Jr., USNRC, to D.P. Hall, HL&P, dated June 8, 1992.

12. NUREG-0781, Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2, including Supplements 1 through 7; Sections 3.11, and 6.2.

13. "South Texas Project Units 1 and 2, Docket Nos. STN 50-498; 50-499, Proposed Licensing Amendment Concerning the RWST and SI Accumulators Allowable Boron Concentration Ranges", Letter ST-HL-AE-4281 from S.L. Rosen, dated January 14, 1993, to USNRC Document Control Desk.