UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

June 2, 2000

William T. Cottle, President and
Chief Executive Officer
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, Texas 77483

SUBJECT: NRC's SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION INTEGRATED INSPECTION REPORT NO. 50-498/00-07; 50-499/00-07

Dear Mr. Cottle:

On May 6, 2000, the NRC completed an inspection at the South Texas Project Electric Generating Station, Units 1 and 2, facility. The enclosed report presents the results of that inspection. The results of this inspection were discussed with Mr. T. Cloninger and other members of your staff on May 9, 2000.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Four issues were evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. Of the four issues, one was determined to involve a violation of NRC requirements, but because of its very low safety significance, the violation is not cited, consistent with Section VI.A of the enforcement policy. The NCV is described in the subject inspection report. If you contest the violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the South Texas Project Electric Generating Station, Units 1 and 2, facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if requested, will be placed in the NRC Public Document Room (PDR), and will be available on the NRC Public Electronic Reading Room (PERR) link at the NRC home page, http://www.nrc.gov/NRC/ADAMS/index.html.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Joseph I. Tapia, Chief
Project Branch A
Division of Reactor Projects

Docket Nos.: 50-498
50-499
License Nos.: NPF-76
NPF-80

Enclosure:
NRC Inspection Report No.
50-498/00-07; 50-499/00-07

cc w/enclosure:
T. H. Cloninger, Vice President
Engineering & Technical Services
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, Texas 77483

S. M. Head, Supervisor, Licensing
Quality & Licensing Department
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, Texas 77483

A. Ramirez/C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, Texas 78704

M. T. Hardt/W. C. Gunst
City Public Service Board
P.O. Box 1771
San Antonio, Texas 78296

D. G. Tees/R. L. Balcom
Houston Lighting & Power Company
P.O. Box 1700
Houston, Texas 77251

Jon C. Wood
Matthews & Branscomb
One Alamo Center
106 S. St. Mary's Street, Suite 700
San Antonio, Texas 78205-3692

A. H. Gutterman, Esq.
Morgan, Lewis & Bockius
1800 M. Street, N.W.
Washington, D.C. 20036-5869

G. E. Vaughn/C. A. Johnson
Central Power & Light Company
P.O. Box 289
Mail Code: N5012
Wadsworth, Texas 77483

INPO
Records Center
700 Galleria Parkway
Atlanta, Georgia 30339-5957

Bureau of Radiation Control
State of Texas
1100 West 49th Street
Austin, Texas 78756

Jim Calloway
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, Texas 78701-3326

John L. Howard, Director
Environmental and Natural Resources Policy
Office of the Governor
P.O. Box 12428
Austin, Texas 78711-3189

Judge, Matagorda County
Matagorda County Courthouse
1700 Seventh Street
Bay City, Texas 77414

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(NFO)**
Branch Chief, DRP/A **(JIT)**
Senior Project Engineer, DRP/A **(DNG)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
STP Site Secretary **(LAR)**

DOCUMENT NAME: R:_STP\ST2000-07RP-NFP.wpd
To receive copy of document, indicate in box: "C" = Copy without enclosures **"E"** = Copy with enclosures **"N"** = No copy

RIV:SRI		RI		EMB:DRS		C:EMB		C:DRP/A	
NFO'Keefe/tbh		GLGuerra		LEllershaw		DAPowers		JITapia	
05/25/00 ***/RA/***		05/24/00 ***/RA/***		05/26/00 ***/RA/ JEW***		05/26/00 ***/RA/ JEW***		06/02/00 ***/RA/***	

OFFICIAL RECORD COPY

ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-498 50-499
License Nos.:	NPF-76 NPF-80
Report No.:	50-498/00-07 50-499/00-07
Licensee:	STP Nuclear Operating Company
Facility:	South Texas Project Electric Generating Station, Units 1 and 2
Location:	FM 521 - 8 miles west of Wadsworth Wadsworth, Texas 77483
Dates:	April 2 through May 6, 2000
Inspectors:	Neil F. O'Keefe, Senior Resident Inspector Gilbert L. Guerra, Resident Inspector L. E. Ellershaw, Senior Reactor Inspector C. A. Clark, Reactor Inspector
Approved By:	J. I. Tapia, Chief, Project Branch A

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: NRC's Revised Reactor Oversight Program

SUMMARY OF FINDINGS

South Texas Project Nuclear Station, Units 1& 2
NRC Inspection Report 50-498/00-07, 50-499/00-07(DRP)

The report covers a 5-week period of resident inspection, and an announced inspection of the inservice inspection program by two regional specialist inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process in draft Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- Green. Unit 1 operators did not thoroughly determine the impact of a tagout to deenergize an electrical panel before authorizing the tags. This caused an unexpectedly loss of a portion of the instrument air system that impacted safety cooling water systems for both units. Operators in Unit 2, which was operating at full power, responded to the resulting low intake bay level in the Train C essential cooling water system by declaring the train, and thus all supported equipment, inoperable (Section 1R04).

  This issue was characterized as a “green” finding using the significance determination process. It was determined to have very low risk significance because the remaining two trains of ESF equipment were sufficient to maintain sufficient mitigating system capability.

- Green. The inspectors determined that post-maintenance testing performed following replacement of high voltage power supplies in both source range nuclear instruments in Unit 1 were inappropriate to demonstrate proper instrument performance. The tests specified observing proper indications for existing plant conditions. The tests were signed as completed despite the fact that the core was defueled, which prevented obtaining any instrument response that demonstrated proper operation (Section 1R19).

  This issue was characterized as a “green” finding using the significance determination process. It was determined to have a very low risk significance because the core was defueled and the instruments were determined to have been operable prior to fuel loading.

Cornerstone: Barrier Integrity

- Green. During a design review for the replacement steam generators, the licensee identified that both units were outside their licensing basis because both charging pumps restarted automatically upon a loss of offsite power. The safety analysis for loss of offsite power assumed that the charging pumps would not restart upon a loss of power because this condition may result in overfilling the pressurizer. The licensee promptly modified both units to restore the facility to within the license basis. The failure to properly incorporate the licensing basis into the plant as-built design was a violation of 10 CFR Part 50, Appendix B, Criterion III. This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy and is in the licensee’s corrective action program as Condition Report 00-3229 (Section 1R17).

This violation was characterized as a “green” issue using the significance determination process. It was determined to have very low risk significance because the existing plant configuration (power-operated relief valve capacity) was sufficient to prevent challenging the pressurizer safety valves. To create a loss of reactor coolant system integrity via the power-operated relief valves would require a loss of offsite power, a power-operated relief valve failure, and a power-operated relief valve block valve failure, which is a low probability scenario.

Cornerstone: Initiating Events

- Green. An uncontrolled power increase from full power occurred in Unit 2 while calibrating a deaerator level instrument in the steam plant. The procedure, which was normally performed when the plant was shutdown, did not alert the technicians to the effects of performing the procedure while at power. Operator action terminated the power increase at a peak of 103.7 percent and restored power below 100 percent within 3 minutes (Section 4OA3.1).

  This issue was characterized as a “green” finding using the significance determination process. It was of very low risk significance due to the brief duration of the transient and no thermohydraulic limits were exceeded.

Report Details

Summary of Plant Status: Unit 1 was shutdown for refueling and steam generator replacement. Unit 2 operated at or near full power throughout the inspection period.

2. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignments

.1 Tagout Unexpectedly Rendered a Safety Train Inoperable

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the loss of a Unit 2 emergency cooling water pump which resulted from a planned maintenance evolution.

b. Issues and Findings

Operators in Unit 1 did not adequately determine the impact of a tagout to deenergize an electrical panel before authorizing the tags for outage work. Deenergizing the electrical panel resulted in a Unit 2 safety train being declared inoperable for 28 minutes.

When Unit 1 operators tagged out an electrical panel for planned work on April 18, the licensee inadvertently deenergized the instrument air valve that supplied air to systems outside the power block. This resulted in loss of indication of pump suction bay levels in all trains of essential cooling water (ECW) in both units. Control room operators in both units recognized the problem from a number of alarms in different systems.

Unit 1 was unaffected because all three trains of ECW were running. However, Unit 2 was in a normal at-power lineup running two trains of ECW, with the remaining train in standby. ECW screen wash booster Pump 2C automatically started in response to an indicated low intake bay level. Since this pump was supplied from the output of the associated ECW pump (which was idle), operators placed the screen wash booster pump in pull to lock for equipment protection and declared the ECW train and all supported systems inoperable for about 28 minutes until instrument air was restored.

Phase 1 and 2 significance determinations process were performed to evaluate the loss of one train of safety equipment in Unit 2, conservatively assuming no operator action to recover the train in the event of an accident. This issue was determined to be within the licensee control band because the remaining two trains of engineered safety feature equipment (ESF) were available (green). No violation was identified. This issue was entered into the licensee’s corrective action program as Condition Report 00-7094.

1RO8 Inservice Inspection Activities

a. Inspection Scope

The inspectors reviewed the South Texas Project Electric Generating Station's (STPEGS) inservice inspection program activities to verify that the program for monitoring degradation of the reactor coolant system boundary was effective. As part of this effort, the inspectors:

- Reviewed "Inservice Inspection Plan For The First Inspection Interval of the South Texas Project Electric Generating Station-Unit 1," Revision 0, Supplement 3, and "Examination Plan for the 1RE09 Inservice Inspection of Welds and Component Supports," dated February 2000.

- Reviewed a sample of licensee and contractor (Bechtel Energy Corporation) nondestructive examination procedures.

- Reviewed a sample of completed Unit 1 Refueling Outage 9 inservice inspection (magnetic particle examination and ultrasonic examination) records (preservice inspection of reactor coolant system, feedwater, main steam pipe-to-pipe and pipe-to-nozzle welds associated with the licensee's replacement of the Unit 1 steam generators).

- Observed in-process radiography that was performed for information only (i.e., prior to postweld heat treatment) on reactor coolant system and feedwater system pipe-to-nozzle welds and reviewed associated technique sheets and reader sheets.

- Reviewed VT-3 visual examinations performed on essential service water system component supports.

- Verified that applicable inservice inspection requests for relief had been approved by NRC prior to use by the licensee.

- Reviewed applicable ASME Code Cases and the licensee's use of them.

b. Issues and Findings

There were no findings identified during this inspection.

1R17 Permanent Plant Modifications

Modification to Load Sequencer to Comply with License Basis

a. Inspection Scope

The inspectors reviewed design and license basis documents and compared them to the documentation in Condition Report 00-3229 and the related design change packages.

The implementation work documents were reviewed and installation in Unit 2 was observed. The inspectors also reviewed the as-built condition and the licensee's actions to eliminate the nonconforming condition per Generic Letter 91-18.

b. Issues and Findings

On March 1, 2000, the licensee identified that, during a loss of offsite power (LOOP) event, the ESF load sequencers would start both centrifugal charging pumps (CCPs), contrary to the license basis of the plant. The licensing basis analysis for a LOOP assumed that the CCPs would not be started to assure that the pressurizer would not be filled and challenge the power-operated relief valves (PORVs) or safety valves. This condition was determined to be an unreviewed safety question that had existed since the units were licensed. The licensee promptly restored the plant to the licensed configuration.

The design basis LOOP scenario was a loss of feedwater that resulted in a turbine trip, which caused a LOOP when the main generator tripped. A LOOP would result in a loss of instrument air since it was not powered from Class 1E power. A loss of instrument air would cause reactor coolant letdown to isolate and the charging flow control valve to fully open. A postulated failure of the ESF actuation system would cause a failure to start both Trains A and D of auxiliary feedwater. The associated steam generators would eventually boil dry, with the resulting loss of heat removal causing a coolant expansion in the primary plant.

The license basis analysis assumed that no charging pumps restarted following a LOOP to prevent filling the pressurizer to a water solid condition. This acceptance criteria was used by the NRC because the pressurizer safety valves were not qualified to pass liquid water and would be assumed to fail open if challenged with water.

The licensee evaluated the impact of the as-built configuration on the license basis scenario. The result was that with both CCPs starting automatically, the pressurizer would reach a water-solid condition in less than 10 minutes following a LOOP. The licensee evaluated possible operator actions to mitigate filling of the pressurizer, but simulator tests showed that avoiding overfilling could not be assured. The licensee determined that the PORVs were qualified and adequately sized to mitigate the pressure rise by relieving liquid water thus avoiding a challenge to the safety valves. Although the facility was considered to be within the design basis, because the plant could respond to the scenario, it was still outside the license basis because the license basis acceptance criteria would be exceeded.

As a result of this unreviewed safety question, the licensee chose to modify the plant to remove the automatic restart signal for the CCPs from the LOOP load sequencers and restore the plant to within its license basis. The modification was performed as Design Change 00-3229 on April 12.

The inspectors reviewed the potential consequences of the existing plant condition. The CCPs had been part of the LOOP load sequence to ensure that a CCP was available to provide seal flow to the reactor coolant pumps to avoid seal failure and a resulting loss of coolant accident. Restoring the plant to within its license basis created the condition that seal flow was no longer assured. The licensee implemented procedure changes to have operators evaluate the availability of instrument air and letdown and start a CCP to restore seal flow if it could be done without overfilling the pressurizer. If operators failed to restore seal flow, the reactor coolant pumps' seals would still be protected from overheating and subsequent failure by component cooling water flow from any of the three trains. If operators restored flow inappropriately and caused a pressurizer overfill, the PORVs would adequately relieve water (although the original analysis assumed they would not) to prevent failure of the pressurizer safety valves. The steam/water mixture would discharge to the pressurizer relief tank inside containment. Eventually, the pressure relief tank would pressurize, bursting the rupture disk and releasing steam/water to the containment. This was within the design basis of the containment, with no consequences to the public. A loss of reactor coolant system integrity via the power-operated relief valves would require a LOOP, a failure of a power-operated relief valve, and a failure of the associated power-operated relief valve block valve. This was considered a low probability scenario. Based on these sequences, the inspectors concluded that the original and modified plant configurations did not have a significant impact on risk. This issue was evaluated using a Level 1 significance determination process and determined to be within the licensee response band (green).

The modification documentation and safety evaluation appropriately evaluated the impact of the plant change and identified procedure changes needed to restore charging and seal flow, where appropriate. Changes in plant risk were evaluated and documented, and the plant risk model was updated.

The inspectors noted that the licensee did not perform a formal operability review, which was called for by Generic Letter 91-18, for the original nonconforming condition. The inspectors performed a review and concluded that no safety functions were affected by the original nonconforming condition. The licensee agreed to document the results of their review. The failure to properly translate the license basis into the existing configuration control documents and systems was a violation of 10 CFR Part 50, Appendix B, Criterion III, and is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy (NCV 50-498/00-07-01; 50-499/00-07-01).

1R19 Postmaintenance Testing

.1 Postmaintenance Test Did Not Assure Proper Indications

a. Inspection Scope

The inspectors reviewed Work Packages 106669 and 106670, which were written to replace the high voltage power supplies in each source range nuclear instrument in Unit 1. The post-maintenance testing was reviewed for the existing plant conditions.

b. Issues and Findings

The inspectors determined that post-maintenance testing performed following replacement of the high voltage power supplies in both source range nuclear instruments in Unit 1 was insufficient to demonstrate proper instrument performance. The tests specified observing proper indications for existing plant conditions. The tests were signed as completed despite the fact that the core was defueled, which prevented obtaining any instrument response that demonstrated proper operation.

The inspectors noted that these instruments were required to be operable prior to beginning refueling to ensure redundant capability to detect changes in core reactivity. The instruments had not been declared operable when the inspectors reviewed the testing because the packages were in routing for closure.

In response to the inspectors' concern, the operators discussed the issue with the work group supervisor. They were able to conclude that work steps included in the work packages were sufficient to demonstrate that the instruments were capable of performing properly. The instruments were then declared operable based on that evaluation (not the post-maintenance test). The inspectors also noted that the instruments responded properly during core reloading.

This issue was identified while the plant was defueled and no nuclear instrumentation was required. However, lacking adequate testing, the first opportunity to identify that the instruments might not be working would be after a mode change was made which required that both source range nuclear instruments be operable. The licensee concluded that it would be more appropriate to schedule future work for different outages for each instrument so that both instruments would not be affected at the same time. This issue was placed in the licensee's corrective action program an Condition Report 00-7204. Since the detectors were determined to be operable before refueling began, the issue was determined to be within the licensee response band (green).

4. OTHER ACTIVITIES

4OA3 Event Followup

.1 Overpower Event in Unit 2

a. Inspection Scope

The inspectors responded to the control room upon report of an unplanned power increase in Unit 2. Control room operators present during the event were interviewed. The significance of the event was assessed during discussions with the NRC's senior reactor analysts and licensee risk assessment personnel. The results of the licensee's investigation were reviewed.

b. Issues and Findings

On April 7, an uncontrolled power increase from full power occurred in Unit 2 while calibrating a level instrument in the steam plant. Operator action terminated the power increase at a peak of 103.7 percent and reduced power to less than 100 percent within 3 minutes. This event was of minor risk significance due to the brief time that power was above 100 percent.

During calibration of a deaerator level instrument channel on April 7, Instrumentation and Controls technicians removed a nuclear mixing amplifier card to make an adjustment. When the card was pulled to place on an extender board, both channels of level control and one channel of pressure control failed. The pressure control circuit opened the main steam supply valves to admit high pressure steam to the deaerator, causing a steam demand increase. The licensee determined that power reached 103.7 percent before operator actions reduced power. The plant was above 100 percent for 3 minutes.

The inspectors determined that this transient was bounded by the safety analysis for an uncontrolled increase in steam demand presented in Section 15.1.3 of the Updated Final Safety Analysis Report. This analysis demonstrated that the reactor was protected without relying on the reactor trip system. The risk significance was evaluated in discussions with the Region IV Senior Reactor Analysts and licensee risk assessment personnel and was determined to have a very low risk significance due to the brief duration of the event (green).

Procedure PMI-IC-FW-7175A, “Deaerator Storage Tank Level Calibration,” Revision 0, was normally performed when the steam plant was shutdown. The procedure had been reviewed by the licensee and determined to be acceptable for performance at power. This review did not identify that, if adjustments were necessary, that the nuclear mixing amplifier card would have to be pulled, and that the observed transient would result. The licensee generated CR 00-6348 to initiate corrective action.

.2 <u>(Closed) Licensee Event Report 499/2000-002-00</u>: reactor overpower event during maintenance on steam plant controller. This event is discussed in detail in Section 4OA3.1 above.

.3 <u>(Closed) Licensee Event Report 498/00-004</u>: Temporary nonclass power supplying a safety bus rendered the bus inoperable and was not recognized. Although the bus was inoperable per the Technical Specifications, it remained functional. In accordance with Section IV of the NRC Enforcement Policy, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

4OA5 <u>Management Meetings</u>

<u>Exit Meeting Summary</u>

The inspectors presented the inspection results to Mr. T. Cloninger and other members of licensee management at exit meetings on April 7 and May 9, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

H. Atkins, Manager, Maintenance
M. Berrens, Manager, Work Control
T. Cloninger, Vice President, Generation
J. Crenshaw, Manager System Engineering Department
W. Dowdy, Manager, Operations Support
J. Drymiller, Superintendent, Security
R. Gangluff, Manager, Chemistry
E. Halpin, Manager, Operations
S. Head, Supervisor, Licensing
W. Jump, Manager, Outage Management
A. Kent, Manager, Electrical/Instrumentation and Controls, Systems Engineering
D. Leazar, Director, Nuclear Fuels and Analysis
B. MacKenzie, Manager, Operating Experience Group
F. Mangan, Vice President, Business Services
G. Parkey, Manager, Plant Generation
T. Powell, Manager, Health Physics
P. Serra, Manager, Plant Protection
J. Sheppard, Vice President, Engineering and Technical Services
M. Smith, Manager, Plant Support Quality

NRC

None.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

498; 499/00007-01	NCV	Condition outside licensing basis (1R17)

Closed

498; 499/00007-01	NCV	Condition outside licensing basis (1R17)
499/2000-002-00:	LER	Reactor overpower event during maintenance on steam plant controller (Section 4OA3.2)
498/00-004	LER	Temporary nonclass power supplying a safety bus rendered the bus inoperable and was not recognized (Section 40A3.3)

Discussed

None.

LIST OF ACRONYMS AND INITIALISMS USED

CCP	centrifugal charging pump
ECW	essential cooling water
ESF	engineered safety feature
LER	licensee event report
LOOP	loss of offsite power
NCV	non-cited violation
PORVs	power-operated relief valves

LIST OF DOCUMENTS REVIEWED

Procedures

PMI-IC-FW-7175A	Deaerator Storage Tank Level Calibration	Revision 0
OPGP04-ZE-0304	Inservice Inspection Program for Welds and Component Supports	Revision 1
OPQP05-ZA-0018	Dry Powder Magnetic Particle Examination for ASME XI PSI/ISI	Revision 2
OPQP05-ZA-0004	General Ultrasonic Examination	Revision 1
OPQP05-ZA-0023	Visual Examination of Component Supports for ASME XI Inservice Inspection	Revision 1
Radiography Procedure	Bechtel Nondestructive Examination Standard, RT Examination, RT-ASME III	Revision 1
P8-T(RA)	Bechtel Welding Procedure Specification	Revision 2
P3(G3)-AT-Lh (E10018-D2)(CVN+40)	Bechtel Welding Procedure Specification	Revision 0
GWS-1	General Welding Standard	Revision 2
PQR 1041	Procedure Qualification Record for P8-T(RA)	Revision 0
PQR 1235	Procedure Qualification Record for P3(G3)-AT-Lh(E10018-D2)(CVN+40)	Revision 0

Nondestructive Examination Reports

Radiography Report RT99-132, March 30, 2000, technique sheets, reader sheets, and film

Radiography Report RT99-143, April 2, 2000, technique sheets, reader sheets, and film

Radiography Report RT99-161, April 5, 2000, technique sheets, reader sheets, and film

Magnetic Particle Examination Report MT-00-0009, January 26, 2000

Magnetic Particle Examination Report HFW 0251, March 3, 2000

Component Support Examination (VT-3) Reports VTS-00-0021 through VTS-00-0043, March 3-8, 2000

Ultrasonic Examination Report UT-00-0003, January 20, 2000

Ultrasonic Calibration Record UTCAL-00-0006, January 20, 2000

Ultrasonic Calibration Record UTCAL-00-0007, January 20, 2000

Certified Material Test Reports for Welding Material From Weldstar

ER308L, Control Number R027

ER90S B3, Control Number R013 and R014

E7018, Control Number R005 and R006

E9018, Control Number R011 and R012

E10018, Control Number R031 and R032

Requests for Relief

RR-ENG-08 through RR-ENG-31

Condtion Records

CR 00-6166, dated April 6, 2000

CR 00-6318, dated April 6, 2000

ASME Code Cases

N-408, N-426, N-429, and N-491-1

Quality Assurance Audits and Surveillance Reports

Quality Surveillance Report 99-019, June 3, 1999

Quality Audit Report 99-11 (TE), August 16, 1999

Quality Audit Report 99-18 (EMN), December 14, 2000

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an action matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plan, as described in the action matrix.

More information can be found at: http:\\www.nrc.gov\NRR\OVERSIGHT\index.html.