From: <obaidb@txu.com>
To: <dhj@nrc.gov>
Date: 5/25/01 11:09AM
Subject: Relief Request Files

Subject: Relief Request Files

Attached are the sample and shell files for ISI/impractical relief requests.

(See attached file: sample1.wpd)(See attached file: ISI RR shell.wpd)

Ref: 10 CFR 50.55a(g)(5)(iii)

CPSES-2001nnnnnn
Log # TXX-01nnn
File # 10010.1

May 15, 2001

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION UNIT 2
DOCKET NO. 50-446
REQUEST FOR RELIEF NO. B-1
FROM INSERVICE INSPECTION REQUIREMENTS
(1986 EDITION OF ASME CODE, SECTION XI, NO ADDENDA;
UNIT 2 INTERVAL DATES: AUGUST 3, 1993 TO AUGUST 3, 2003;
FIRST INTERVAL)

Gentlemen:

Attached is Request for Relief No. B-1 from certain inservice inspection requirements for CPSES Unit 2. This request is submitted in accordance with 10 CFR 50.55a(g)(5)(iii). Approval is requested by February 15, 2002 (this date is not based on an operational or compliance requirement for TXU Electric but is proposed for planning purposes while allowing sufficient time for NRC review and resolution).

This communication contains no new licensing basis commitments regarding CPSES Units 2. Should you have any questions, please contact Obaid Bhatty at 254-897-5839.

Sincerely,

C. L. Terry

By: ____________________
Roger D. Walker
Regulatory Affairs Manager

LLE/wp

Attachments
Enclosures

c - E. W. Merschoff, Region IV
D. N. Graves, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

TXU Electric
Comanche Peak Steam Electric Station (CPSES), Unit 1
Second 10-Year Interval
Request for Relief No. B-1

I. System/Component for Which Relief is Requested:

Five Pressurizer Nozzle to Vessel Welds.

Examination Category B-D, Item No. B3.110

4" Pressurizer Spray Nozzle to Vessel Weld (Weld TBX-1-2100-12)
6" Pressurizer Safety Nozzle to Vessel Weld (Weld TBX-1-2100-13)
6" Pressurizer Relief Nozzle to Vessel Weld (Weld TBX-1-2100-14)
6" Pressurizer Relief Nozzle to Vessel Weld (Weld TBX-1-2100-15)
6" Pressurizer Safety Nozzle to Vessel Weld (Weld TBX-1-2100-16)

II. Code Requirement:

Section XI, Table IWB-2500-1, Examination Category B-D, Item No. B3.110 requires complete ultrasonic examinations of the volume defined by Figure IWB-2500-7(b).

III. Code Requirement from Which Relief is Requested:

Relief is requested from performing complete ultrasonic examinations of the volume defined by Figure IWB-2500-7(b).

IV. Basis for Relief:

Complete examination of the volume defined by Figure IWB-2500-7(b) is impractical for the subject welds because of the geometrics of the examination area for these welds . The specific examination area geometries for the five nozzle to vessel welds preclude the complete examinations of the volume required by Figure IWB-2500-7(b) (i.e., the curvature of the surface in combination with the limitations of the required transducer prohibited the scan from reaching the entire volume to be examined). Approximately 26% of the weld volume for TBX-1-2100-12, -13, -14, -15, and -16; spray, safety, and relief nozzle to vessel welds, did not receive the full code required coverage. Refer to pages 3 through 8 for the weld locations and the examination area configurations.

Best effort examinations were performed and consisted of two separate beam angles. Full circumferential scans were obtained for all of the subject welds and the required base metal areas. Axial scan coverage of 93% was achieved in at least one beam path direction with two different angles for each of the spray, safety, and relief nozzle to vessel welds. Axial scan coverage of 96% was achieved in at least one beam path direction with one beam angle for each of the spray, safety, and relief nozzle to vessel welds. There were no recordable indications identified by the best effort examinations.

V. Alternate Examinations:

No alternate examinations are proposed in lieu of the Ultrasonic examinations conducted for the subject welds.

VI. Justification for the Granting of Relief:

The subject welds were examined to the maximum extent possible (approximately 74% of examination completed in all cases) and yielded no indications. Based on the high percentage of the examination completed, the lack of any indications and the knowledge that the portions of the welds not inspected were performed using the same personnel, equipment and materials as the examined portions, there is a high level of confidence in the continued structural integrity of the weld, there is no anticipated impact upon the overall plant quality and safety, and the health and safety of the public should not be jeopardized by the granting of relief.

VII. Implementation Schedule:

All five of the subject weld examinations were performed during the 1st outage, 1st period, of the second 10-year interval for CPSES, Unit 1.

Ref: 10 CFR 50.55a(g)(5)(iii)

CPSES-yyyynnnnnn
Log # TXX-yynnn
File # 10010.1

[Insert Date}

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION UNIT[S 1 AND 2]
DOCKET NO[S]. 50-44[5 and 50-446]
REQUEST[S] FOR RELIEF NO[S]. [B-1]
FROM INSERVICE INSPECTION REQUIREMENTS
[(1986 EDITION OF ASME CODE, SECTION XI, NO ADDENDA;
UNIT 2 INTERVAL DATES: AUGUST 3, 1993 TO AUGUST 3, 2003;
FIRST INTERVAL)]

REF: 1) NRC Letter from Someone to Someone else dated Something

2) TXU Electric Letter, logged TXX-xxxxx, from Someone to Someone else dated Something

Gentlemen:

Attached [is/are] Request[s] for Relief No[s]. [B-1] from certain inservice inspection requirements for CPSES Unit [2]. This request is submitted in accordance with 10 CFR 50.55a(g)(5)(iii). Approval is requested by [request date plus 9 months].

This communication contains no new licensing basis commitments regarding CPSES Units 2. Should you have any questions, please contact Obaid Bhatty at 254-897-5839.

Sincerely,

C. L. Terry

By: ____________________
Roger D. Walker
Regulatory Affairs Manager

LLE/wp

Attachments
Enclosures

c - E. W. Merschoff, Region IV
[Regional Branch Chief], Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

TXU Electric
Comanche Peak Steam Electric Station (CPSES), Unit [1]
Second 10-Year Interval
Request for Relief No. [B-1]

I. System/Component for Which Relief is Requested:

[Identify the number of items for which relief is being requested. Provide an itemized list of the specific welds(s) and/or component(s) requiring relief.]

II. Code Requirement:

[Identify the code requirement(s) that apply to the weld(s) and component(s) listed in 1 above. Identify the specific Code paragraph that applies and a verbal description of the requirements (i.e., the examinations that apply to these items).]

III. Code Requirement from Which Relief is Requested:

[Identify the requirements (out of those identified in 2 above) from which the relief is being requested. Normally this will be a specific examination requirement. For example, "Relief is requested from [specific requirement which is impractical to perform]."]

IV. Basis for Relief:

[Explain why the requirement is impractical. For example, "The volumetric examination requirement is impractical to perform because of the geometry of the subject welds (see attached drawing on pages 3 of 4 and 4 of 4)."

Provide an estimate of the percentage of the Code-required examination that can be completed for each of the individual welds and components subject to relief.

If the Code required examination cannot be performed because of a limitation or obstruction, describe or provide drawings showing the specific limitation or obstruction (i.e., define what code requires and why it is impractical given the surface geometry). Attache drawings.]

V. Alternate Examinations:

No alternate examinations are proposed.

Or

TXU Electric proposes, in lieu of the Code required [ ] examination, the [subject components} will receive [ ] examination. [Provide sufficient explanation to adequately describe the alternate examination being proposed]

Or

TXU Electic proposes to supplement the partial [ ] examinations performed per the Code requirements with [ ] examination[s]. [Provide sufficient explanation to adequately describe the alternate examination being proposed]

VI. Justification for the Granting of Relief:

[Address the following regarding why relief should be granted: (a) How the proposed alternatives or partial examination provide a reasonable assurance of the continued structural integrity, (b) the burden upon the TXU should the Request for Relief be denied, and (c) why public health and safety will not be jeopardized by the granting of relief. If it is not possible to perform alternative examinations, discuss the impact on the overall level of plant quality and safety. For example, "The subject welds were examined to the maximum extent possible (approximately 74% of examination completed in all cases) and yielded no indications. Based on the high percentage of the examination completed, the lack of any indications and the knowledge that the portions of the welds not inspected were performed using the same personnel, equipment and materials as the examined portions, there is a high level of confidence in the continued structural integrity of the weld, there is no anticipated impact upon the overall plant quality and safety, and the health and safety of the public should not be jeopardized by the granting of relief."]

VII. Implementation Schedule:

[Note: Requests for relief are only applicable for the 10 -year inspection interval during which relief was requested and approval does not apply for subsequent inspection intervals. Specifically identify the periods and intervals for which the relief has been or will be applied (e.g., All five of the subject weld examinations were performed during the 1st outage, 1st period, of the second 10-year interval for CPSES, Unit 1.)]

From: "Michael Riggs" <mriggs1@txu.com>
To: <dhj@nrc.gov>
Date: 5/23/01 3:08PM
Subject: Preliminary Response to RAI re: LAR 01-06

Mr. Jaffe,

Attached is a WordPerfect file with responses to questions on CPSES License Amendment Request (LAR) 01-06 as discussed on May 9, 2001.

Mike Riggs

(See attached file: Rai.wpd)

CC: <skarpyak@txu.com>, "Dan Tirsun" <dtirsun1@txu.com>, "Don Woodlan" <dwoodla1@txu.com>

SUBJECT: Preliminary Response to Request For Additional Information Regarding Comanche Peak Proposed Technical Specification Change as Submitted in License Amendment Request (LAR) 01-06

REF: Conference Call on May 9, 2001 between NRC's D. H. Jaffe and Millard Wohl, and CPSES' Steve Karpyak, Dan Tirsun, and Michael Riggs

The following questions were asked by Millard Wohl / Dave Jaffe regarding the CPSES DG AOT submittal (LAR 01-06) : [Questions have been paraphrased based on participant notes.]

1. Provide some additional bases for excluding externals from the quantitative assessment. These bases can be qualitative.

Response: CPSES has prepared an engineering report in support of the subject submittal. The following excerpts from this report provide the basis for excluding external events.

External Events

Fires

The IPEEE fire analysis results for Comanche Peak were not combined with the internal events PSA results. The risk metrics calculated for this submittal, therefore do not include contributions from internal fires. However, the IPEEE fire risk assessment at Comanche Peak did not identify any vulnerabilities associated with diesel generators.

In order for fires to affect the risk metrics evaluated for the EDG AOT submittal, they would have to either a) cause a Loss of Offsite Power through cable damage, b) cause a LOSP and fail a EDG at the same time (while not failing the electrical bus).

Due to the actual installed cable routing and separation criteria, a significant fire that affects multiple compartments and multiple trains of equipment would be required to initiate a LOSP. The probability of occurrence of a fire of this magnitude is at least two orders of magnitude below the frequency of a random LOSP.

The change in risk (as determined by CDF and LERF) due to the increased Completion Time is dominated by accident sequences involving independent EDG maintenance unavailabilities. The proposed changes to the EDG Completion Time has a negligible effect, if any, on fire risk. A similar argument applies to the start-up transformers.

Tornadoes

The inclusion of LOSP due to tornadoes can not increase LOSP-induced CDF by more than 10% even if conservative assumptions are used. The CDF calculations still support the extension of EDG AOT, although it is necessary to argue the differential risk by moving the EDG overhaul from shutdown to Mode 1.

The base case (internal events excluding fires and floods) CDF for the updated Comanche Peak PSA is 2.0E-5/yr while in Mode 1. The PSA includes an initiator for LOSP. The IE frequency and recovery probabilities for LOSP are derived from generic and plant specific experience and as such include the effect of tornado-induced LOSP. The IPEEE (completed in 1995) addresses CDF specifically from tornadoes. The probability of a direct hit was 5E-4/ year. The IPEEE calculates CDF due to tornadoes as

3E-6. However, the recent update to the PRA changed the data for SBO-related events. Rather than revise the IPEEE, a scoping assessment of tornadoes has been performed.

It is assumed that occurrence of a tornado is 5E-4/yr, which will guarantee a LOSP and eliminate the possibility of recovery for 24 hours. The scoping assessment is made by using the event importance values for the base case PSA. The mission time is 24 hours. This coincides with the mission time for the diesel generators.

In the updated PSA, the base case contribution of LOSP to CDF is 1.64E-5/yr. Virtually all (98%) of this CDF is due to station blackout (i.e., failure of both EDGs). LOSP with one EDG operable is a minimal contributor. If there is no recovery of OSP, the CDF is raised by 7.14E-5/yr. to 7.52E5/yr.

If a tornado initiating event frequency of 5E-4/yr. is assumed, (guaranteed LOSP and no recovery for 24 hours), the CDF from tornadoes is 9.04E-7/yr. If the EDG overhaul is allowed during Mode 1, the increase in EDG unavailability will increase the CDF by 1.35E-7.

Based on the above scoping assessment, specifically including tornado as an IE increases the base case CDF by 5%. If the 14-day AOT is allowed at power, the CDF is further increased by 1.35E-7/yr., an insignificant increase.

If the risk trade off between shutdown and power operation is considered, the consideration of tornadoes has no effect on the EDG AOT extension. A similar argument applies to the start-up transformers. The conditional probability of core damage for the 24-hour station blackout is 1.0 for all operating modes, with possibly the exception of Mode 6 with high water level. So the ICCDP for EDG overhaul is the same regardless if it occurs in Mode 5 or Mode 1, and thus the ICCDP as calculated by RG 1.177 is not an *increase*, but rather a moving of core damage probability from Mode 5 to Mode 1.

2. Was a corrective maintenance case run for the Diesel Generators with a common cause beta included? Did it show the ratio of CM to PM?

Response: The corrective maintenance with common cause beta has been run to support some information for the WOG submittal. We will extract that calculation and discussion from the report and use it here.

As part of CPSES participation in the WOG RI-DG AOT submittal, additional analyses were required to support this effort. To evaluate the impact of diesel generator major maintenance activities, the following steps were performed using the Westinghouse Owner's Group guidance presented in "General Process for Safety Impact of Changes to Technical Specification Allowed Outage Times", Westinghouse Owner's Group, March 10,1999. The following case studies were performed.

Train 'A' EDG Out of Service for Corrective Maintenance.

The Safety Monitor™ Administrator Module was used to modify the values of the following basic events:

- EPCCFDGD12 0.00E-0
- EPCCFDG012 0.00E-0
- EPBDGGEE02NN 3.12E-2
- EPBDGGEE02FN 4.01E-2

The change in the basic events list above reflect the WOG methodology in which the failure rates associated with the remaining operable DG are increased by the Beta CCF factor and the original model CCF events are set to 0.0. This is based on the WOG methodology when one EDG is assumed out of

service for corrective maintenance.
The following configuration changes along with the basic event probability modifications define the input into the Safety Monitor™ (Case 300).

- Train 'A' EDG removed from service
- Alignments were change to show Train 'B' equipment running and Train 'A' equipment in standby.

Train 'B' EDG Out of Service for Corrective Maintenance.

The Safety Monitor Administrator™ Module was used to modify the values of the following basic events:

- EPCCFDGD12 0.00E-0
- EPCCFDG012 0.00E-0
- EPADGGEE02NN 3.12E-2
- EPADGGEE02FN 4.01E-2

The change in the basic events list above reflect the WOG methodology in which the failure rates associated with the remaining operable DG are increased by the Beta CCF factor and the original model CCF events are set to 0.0. This is based on the WOG methodology when one EDG is assumed out of service for corrective maintenance.

The following configuration changes along with the basic event probability modifications define the input into the Safety Monitor™ (Case 301).

- Train 'B' EDG removed from service
- Alignments were change to show Train 'A' equipment running and Train 'B' equipment in standby.

TABLE 1 Summary of Corrective Maintenance Cases				
300	A EDG OOS for corrective maintenance	CDF= 1.28E-4 per year (1)	LERF= 1.67E-5 per year (1)	Adjusts common cause failure rates to 0 and increases the failure probability for the B EDG by the common cause Beta factor.
301	B EDG OOS for corrective maintenance	CDF= 1.27E-4 per year (1)	LERF= 1.67E-5 per year (1)	Adjusts common cause failure rates to 0 and increases the failure probability for the A EDG by the common cause Beta factor.

Note 1: Indicates average Test and Maintenance on the associated train for equipment out of service in addition to the EDG.

These results show that the ratio of CDF CM to CDF PM is 1.28/1.12 = ~ 1.14

The following table shows the values, methodology and results of calculations for the various preventive and corrective maintenance cases.

Comanche Peak 14 Day AOT – CDF Calculations for Corrective Maintenance Common Cause Failure Cases

Required Information	Parameter
DG fail to start failure probability	8.418E-03
DG fail to run failure probability	3.356E-02
DG required mission (run) time in hours	23
DG common cause failure model	MGL Methodology
DG fail to start common cause failure probability (all DGs)	2.624E-4 Beta of 3.12E-2
DG fail to run common cause failure probability (all DGs)	1.402E-3 Beta of 4.01E-2
CDF (current AOT)	1.17E-5
CDF (proposed AOT)	2.55E-5
CDF increase	1.38E-5
CCDF (with one DG out of service due to test or scheduled maintenance activity)	1.12E-4[1] Case 102C
CCDF (with one DG out of service due to corrective/repair maintenance activity)	1.28E-4[1] Case 301
ICCDP (with one DG out of service due to test or scheduled maintenance activity)	3.74E-6
ICCDP (with one DG out of service due to corrective/repair maintenance activity)	4.28E-6

[1] Indicates average T&M on the associated train for equipment out of service in addition to the EDG

3. Shutdown risk is dominated by the mid-loop. It appears that the DG AOT is scheduled during the mid-loop. Please confirm that. Discuss how the DG outage is timed/scheduled with respect to mid-loop, in particular in the early stages of the outage.

Response: CPSES does start the DG outage as soon as TS allow operation with only one DG, at start of mode 5. That means that one DG is unavailable during the early mid-loop and accounts for the risk level. Depending on the length of the DG outage, it is possible that the other DG could be out during the late mid-loop. This is normally what is scheduled and done during outages at CPSES.

4. Is there an editorial problem on the wording of the lead-in to the bulleted list on Page 33, or was something left out?

Response: This is editorial. We will correct the lead-in sentence to read " Updated the PRA model . . ."

5. Do you use Safety Monitor for on-line and ORAM for shutdown?

Response: Yes, Safety Monitor is used for modes 1 and 2 on-line and ORAM is used for shutdown modes 5 and 6. The Safety Monitor is also capable of analyzing transition modes (modes 3-4) and shutdown (modes 5 and 6).

6. Does CPSES use Maintenance Rule a(4) and Configuration Risk Monitoring Program (CRMP)?

Response: Yes, CPSES currently has both a(4) and CRMP processes for controlling maintenance configuration risk. These are considered redundant but CPSES has not requested in this LAR that CRMP be deleted from technical specifications.

7. How is Spent Fuel Pool enveloped in this analysis? Does the CPSES Safety Monitor model SFP releases or just cooling?

Response: The Safety Monitor models SFP cooling, however, it does not model SFP releases. The Safety Monitor model calculates both time to boil and core damage. Both metrics are calculated based on time after shutdown and assuming that once fuel transfer begins, the pool's decay heat load is based on full core off-load with existing fuel accounted for in the decay heat calculation.

8. Discuss the organizations and some of the names of individuals who participated in

the reviews of the CPSES PRA, including the IPE.

Response: Provide a listing of the companies and the individuals who reviewed the CPSES IPE/PRA and provide a listing of companies and individual who assisted in the PRA update. The following organizations and individuals provided independent review of the initial PRA:

J. Gaertner, ERIN; D. Wakefield, PLG; B. Najafi, B. Putney, R. Anoba, Z. Mendoza, SAIC; A. Spurgeon, APG; A. Torri, Risk and Safety Engineering; J. Zamani; F. Hubbard, FRH, Inc.

The following organizations and individuals (principals) provided review and individual expertise in support of updates of the PRA:

D. Jones, Scientech; J. Julius, Scientech; R. Anoba, Anoba Consulting; C. Cragg, DS&S; S. Rao, J.C. Lin, PLG.

From: <jseawright@txu.com>
To: <dhj@nrc.gov>
Date: 5/21/01 10:11AM
Subject: RAI

Dave,

Don asked me to email this proposed response to you.

JDS

(See attached file: SPSB RAI.doc)

SPSB1. What design bases parameters, assumptions or methodologies were changed in

the radiological design basis accident analyses because of the proposed changes? If there are many changes, it would be helpful to compare and contrast them in a table. Also, please provide justification for any changes.

SPSB2 Please describe how the source terms utilized for your dose analyses were generated. Provide the methodology, codes, and databases utilized.

SPSB3. Please provide the offsite and control room dose results from your accident analyses.

DRAFT INITIAL RESPONSE (05/15/2001)

In response to SPSB1, SPSB2, and SPSB3 above, CPSES has not changed any of the licensed design bases to the control room and offsite dose consequences presented in the FSAR. Cycle specific assessments are performed as part of each reload analyses to confirm that the radiological analyses presented in the FSAR remains bounding.

DRAFT Follow up on based on 5/17/2001 telecon with Dave Jaffe and Mark Blumeburg:
The licensing basis dose consequences reported in the FSAR are based upon the computer analysis tools used for dose consequence calculations, which are listed in the FSAR, and a reactor power of 3565 MWth (104.5% of 3411 MWth) likewise listed in the FSAR. Neither the reactor power, of 3565 MWth, nor the licensing basis methodologies have been changed in support of the proposed amendment to increase Units 1 and 2 reactor power to 3458 MWth (1.4% and 0.4% increases).

The dose consequences that provides the license basis, as reported in the FSAR, are based on a fission product inventory derived from an assumed reactor power of 3565 MWth (104.5% of original licensed power level) and a standard three region 12 month fuel cycle at equilibrium. (i.e. a total core mass loading of 89.05 MTU, Core Average Burnup of 24,018 MWD/MTU, and12 month fuel cycle with 3 fuel burnup regions of 300, 600, and 900 EFPD.) The FSAR license basis dose consequences derived from the above fission product inventory has continued to remain bounding through the increase in fuel enrichments and cycle lengths as provided for in amendments 17/3 and 27/13 of the Technical Specifications because of the significant margin provided by the assumed power level of 3565 MWth provided in the license bases. The FSAR license bases dose consequences will continue to remain bounding upon implementation of the proposed amendment to increase reactor thermal power to 3458 MWth for Units 1 and 2.

The FSAR license bases values for dose consequences remain bounding, by determining that the cycle specific fission product inventory is overall less severe and that the licensing basis (i.e. dose consequences) remains unchanged from that already reported in the FSAR.

Cycle specific fission products that were submitted in the proposed amendment provide an example as to how the overall effects of the fission product inventories assessed to assure that the FSAR license basis dose consequences remain bounded on a cycle specific basis.

From: <jseawright@txu.com>
To: <dhj@nrc.gov>
Date: 6/1/01 9:47AM
Subject: RAI

(See attached file: SPSB RAI r1.doc)

CC: "Don Woodlan" <dwoodla1@txu.com>

SPSB1. What design bases parameters, assumptions or methodologies were changed in

the radiological design basis accident analyses because of the proposed changes? If there are many changes, it would be helpful to compare and contrast them in a table. Also, please provide justification for any changes.

SPSB2 Please describe how the source terms utilized for your dose analyses were generated. Provide the methodology, codes, and databases utilized.

SPSB3. Please provide the offsite and control room dose results from your accident analyses.

DRAFT RESPONSE

In response to SPSB1, SPSB2, and SPSB3 above, CPSES has not changed any of the licensed design bases to the control room and offsite dose consequences presented in the FSAR. Cycle specific assessments are performed as part of each reload analyses to confirm that the radiological analyses presented in the FSAR remains bounding.

The radiological dose consequences reported in FSAR Chapter 15 are based upon the computer analysis tools used for dose consequence calculations listed in FSAR Appendix 15B and a reactor power of 3565 MWth (104.5% of 3411 MWth). Neither the assumed reactor power of 3565 MWth, nor the licensing basis methodologies have been changed in support of the proposed amendment to increase the Rated Thermal Power for Units 1 and 2 to 3458 MWth (1.4% and 0.4% increases, respectively).

The radiological dose consequences are based on a fission product inventory derived from an assumed reactor power of 3565 MWth (104.5% of original licensed power level) and a standard three region 12 month fuel cycle at equilibrium. (i.e., a total core mass loading of 89.05 MTU, core average burnup of 24,018 MWD/MTU, and a 12 month fuel cycle with 3 fuel burnup regions of 300, 600, and 900 EFPD). The radiological dose consequences derived from the above fission product inventory has continued to remain bounding through the increase in fuel enrichments and cycle lengths as provided for in amendments 17/3 and 27/13 to the Technical Specifications because of the significant margin provided by the assumed power level of 3565 MWth. The radiological dose consequences presented in the FSAR continue to remain bounding upon implementation of the proposed amendment to increase the Rated Thermal Power to 3458 MWth for Units 1 and 2. This conclusions has also been confirmed to remain valid when an additional allowance of +0.6% has been included to address the power calorimetric uncertainty; (i.e. the assessments for this submittal were performed at 3479 MWth).

The cycle specific fission product inventories submitted in the proposed amendment provide an example, from a previous cycle, as to how the overall effects of the fission product inventories are assessed to assure that the radiological dose consequences remain valid for each cycle. The current cycles for Unit 1 and Unit 2 have been assessed at 3479 MWth, and, as before, it has been determined that the radiological dose consequences presented in FSAR

Chapter 15 continue to remain valid.

Ref: 10 CFR 50.55a(g)(5)(iii)

CPSES-2001nnnnnn
Log # TXX-01nnn
File # 10010.1

June 20, 2001

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1
DOCKET NO. 50-445
REQUEST FOR RELIEF NO. B-1
FROM INSERVICE INSPECTION REQUIREMENTS
(1986 EDITION OF ASME CODE, SECTION XI, NO ADDENDA;
UNIT 1 SECOND INTERVAL, FIRST PERIOD, FIRST OUTAGE
DATES: AUGUST 3, 2000 TO AUGUST 3, 2010;)

Gentlemen:

Attached is Request for Relief No. B-1 from certain inservice inspection requirements for CPSES Unit 1. This request is submitted in accordance with 10 CFR 50.55a(g)(5)(iii). Approval is requested by February 15, 2002 (this date is not based on an operational or compliance requirement for TXU Electric but is proposed for planning purposes while allowing sufficient time for NRC review and resolution).

This communication contains no new licensing basis commitments regarding CPSES Unit 1. Should you have any questions, please contact Obaid Bhatty at 254-897-5839.

Sincerely,

C. L. Terry

By: ____________________
Roger D. Walker
Regulatory Affairs Manager

OAB/ob

Attachments
Enclosures

c - E. W. Merschoff, Region IV
D. N. Graves, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

TXU Electric
Comanche Peak Steam Electric Station (CPSES), Unit 1
Second 10-Year Interval
Request for Relief No. B-1

I. System/Component for Which Relief is Requested:

Five Pressurizer Nozzle to Vessel Welds.

Examination Category B-D, Item No. B3.110

4" Pressurizer Spray Nozzle to Vessel Weld (Weld TBX-1-2100-12)
6" Pressurizer Safety Nozzle to Vessel Weld (Weld TBX-1-2100-13)
6" Pressurizer Relief Nozzle to Vessel Weld (Weld TBX-1-2100-14)
6" Pressurizer Relief Nozzle to Vessel Weld (Weld TBX-1-2100-15)
6" Pressurizer Safety Nozzle to Vessel Weld (Weld TBX-1-2100-16)

II. Code Requirement:

1986 edition of ASME code, Section XI, no addenda, Table IWB-2500-1, Examination Category B-D, Item No. B3.110 requires complete ultrasonic examinations of the volume defined by Figure IWB-2500-7(b).

III. Code Requirement from Which Relief is Requested:

Pursuant to the requirements of 10 CFR 50.55a(g)(5)(iii), relief is requested from performing complete ultrasonic examinations of the volume defined by Figure IWB-2500-7(b).

IV. Basis for Relief:

Complete examination of the volume defined by Figure IWB-2500-7(b) is impractical for the subject welds because of the geometrics of the examination volume for these welds . The specific examination area geometries for the five nozzle to vessel welds preclude the complete examinations of the volume required by Figure IWB-2500-7(b) (i.e., the nozzle curvature of the surface prohibited the beam from reaching the entire volume to be examined). Approximately 26% of the weld volume for TBX-1-2100-

12, -13, -14, -15, and -16; spray, safety, and relief nozzle to vessel welds, did not receive the full code required coverage.

Refer to pages 3 through 8 for the weld locations and the examination area configurations.

Full circumferential scans were obtained for all of the subject welds and the required base metal areas. Best effort examinations were performed in the axial scan directions and consisted of two separate beam angles. Axial scan coverage of 93% for the weld was achieved in at least one beam path direction with two different angles for each of the spray, safety, and relief nozzle to vessel welds. Axial scan coverage of 96% was achieved in at least one beam path direction with one beam angle for each of the spray, safety, and relief nozzle to vessel welds. There were no recordable indications identified by the best effort examinations. Additionally, an inner radius examination was performed on all the subject nozzles. Although this examination is not intended to examine the weld area, the inner radius examination included the area that was not covered the by the Code required examination.

V. Alternate Examinations:

No alternate examinations are proposed in lieu of the Ultrasonic examinations conducted for the subject welds.

VI. Justification for the Granting of Relief:

The subject welds were examined to the maximum extent possible (approximately 74% of examination completed in all cases) and yielded no indications. Based on the high percentage of the examination volume completed, and the lack of any reportable indications, there is a high level of confidence in the continued structural integrity of the welds. There is no anticipated impact upon the overall plant quality and safety, and the health and safety of the public should not be jeopardized by the granting of relief.

VII. Implementation Schedule:

All five of the subject weld examinations were performed during the 1st outage, 1st period, of the second 10-year interval for CPSES, Unit 1.

June 19, 2001

MEMORANDUM TO: Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: David H. Jaffe, Senior Project Manager, Section 1 /RA/
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNIT 2 - DISCUSSION BETWEEN TXU ELECTRIC AND THE U. S. NUCLEAR REGULATORY COMMISSION (NRC) STAFF CONCERNING AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE RELIEF (TAC NO. MB1190)

The NRC staff has had a discussion with TXU Electric (the licensee) concerning ASME Code Relief. In order to facilitate this discussion, the attached, draft information was provided by the licensee. This information was not used in rendering any regulatory decisions.

The purpose of this memorandum is to place the attachment in the Public Document Room.

Docket No. 50-446

Attachment: As stated

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