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September 15, 2006

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-446
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION, RELIEF REQUEST A-1 FOR THE UNIT 2 INSERVICE INSPECTION FOR APPLICATION OF AN ALTERNATIVE TO THE ASME BOILER AND PRESSURE VESSEL CODE SECTION XI EXAMINATION REQUIREMENTS FOR CLASS 1 AND 2 PIPING WELDS (INTERVAL START DATE-AUGUST 3, 2004, SECOND INTERVAL)
TAC NUMBER MC9503

REF: TXU Power letter, logged TXX-05204, from Mike Blevins to the NRC dated December 15, 2005

Gentlemen:

By means of the referenced letter, TXU Generation Company LP (TXU Power) previously submitted relief from the ASME Section XI code examination requirements for inservice inspection of Class 1 and 2 piping welds (Categories B-F, B-J, C-F-1, and C-F-2) for Comanche Peak Steam Electric Station (CPSES) Unit 2.

Based upon questions provided by Mr. Mohan Thadani of the NRC in an email dated September 11, 2006, and discussions with the NRC staff on September 13, 2006, TXU Power hereby provides the following additional information. The attachment to this letter contains the NRC questions and TXU Power's response immediately following each question.

A047

This communication contains no new licensing basis commitments concerning CPSES Unit 2.

If you have any questions regarding this request, please contact Jack Hicks at (254) 897-6725.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC,
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Regulatory Affairs

JCH

Attachment

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

TXU POWER
COMANCHE PEAK STEAM ELECTRIC STATION

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELIEF REQUEST A-1 FOR THE UNIT 2 INSERVICE INSPECTION FOR APPLICATION OF AN ALTERNATIVE TO THE ASME BOILER AND PRESSURE VESSEL CODE SECTION XI EXAMINATION REQUIREMENTS FOR CLASS 1 AND 2 PIPING WELDS

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1. **NRC Question:**

Regulatory Guide (RG) 1.178, An Approach for Plant-Specific Risk-Informed Decision Making for Inservice Inspection of Piping, Revision 1, dated September 2003, replaced the original *"For Trial Use"* RG dated September 1998. Revision 1 of the RG 1.178 includes guidance on what should be included in risk-informed inservice inspection (RI-ISI) submittals, particularly in dealing with probabilistic risk assessment (PRA) issues. Specifically, on Page 28 of RG 1.178, the following is stated regarding the information that should be included in a submittal:

> "A description of the staff and industry reviews performed on the PRA. Limitations, weakness, or improvements identified by the reviewers that could change the results of the PRA should be discussed. The resolution of the reviewer comments, or an explanation of the insensitivity of the analysis used to support the submittal to the comment, should be provided."

In your original risk-informed inservice inspection (RI-ISI) submittal, dated February 15, 2001, you discussed PRA quality, noting that "the current PSA is scheduled to undergo the Westinghouse certification process in 2001". Hence, at that time your PRA had not yet been industry peer-reviewed, but it apparently has been since.

In your current relief request you note that "An update to the PRA was performed at the end of 2004. Although the revision to the PRA model occurred after the end of the interval, it was decided to include the revision in this evaluation and update." The staff concurs with the decision to re-perform the analysis, given the update to the PRA model.

However, between the original RI-ISI submittal and the current relief request, there is no summary of results of the above industry peer review, or status of the PRA model used to re-perform the analysis relative to it. Hence, to establish confidence that the quality of the PRA is sufficient to support your recent RI-ISI analysis:

a. Please provide a listing of the Level A and B Facts and Observations (F&Os) from the above peer review, along with their resolutions. If there are outstanding F&Os that were not resolved at the time of your re-performed analysis, please explain why resolving them would not have a potentially significant impact on the RI-ISI program (either from the risk-significance of pipe segments or from an overall delta-risk perspective).
b. In addition, please identify any other "open items" with the PRA model that was used to re-perform the RI-ISI analyses that would meet the threshold of a Level A or B F&O, and explain why resolving them would not have a potentially significant impact on the RI-ISI program (again, either from the risk-significance of pipe segments or from an overall delta-risk perspective).

TXU Response to 1a:

The Westinghouse Owners Group (WOG) Peer Review was performed during the spring of 2002. The conclusion of the peer assessment was that the Comanche Peak PRA can be effectively used to support risk significance evaluations with deterministic input, subject to addressing the items identified as significant in the technical element summary and Facts & Observations (F&O) sheets. CPSES has addressed and resolved each of the Category A and B F&Os. The following table provides the category A & B F&Os and their dispositions.

Number	Description Summary	Level	Disposition
AS-01	Provide guidance for and discussion of the process for applying PRA recovery terms.	B	This item does not adversely affect the technical adequacy of the PRA because it is associated with documentation. A new notebook to address post recovery file development and maintenance has been developed. Notebook is R&R-PN-039 "Post Quantification Files"
HR-03	The input received from the operators in the recent round of comments should be documented as part of the analysis to demonstrate continuing PRA fidelity with the as-operated plant.	B	This item does not adversely affect the technical adequacy of the PRA because it is associated with documentation. Original operator interview records of conversation are available as background information and can be used to demonstrate PRA fidelity with the as-operated plant. The HRA analysis documentation has been updated to use the EPRI HRA Calculator and the updated operator interviews have been summarized and documented with the HRA Calculator. PRA desktop instruction R&R-DI-005 "Human Reliability Analysis" section 4.0 was revised to document future operator interviews and training practice changes in the HRA notebook rather than in more informal records of conversation.
HR-04	Resolution of discrepancies in the quantification of HEPs including insufficient documentation detail to reproduce human error probabilities.	B	This item does not adversely affect the technical adequacy of the PRA because it is associated with documentation discrepancies and lack of detail. Revised guideline R&R-DI-005 "Human Reliability Analysis" to ensure documentation is sufficient to reproduce human error probabilities. This was achieved as part of Revision 3 update of the HRA guideline.
HR-05	Applicability of using only 2 Cause Based Decision Trees for Human Reliability Analysis and development of a Cause Based Decision Tree basis. Since the manner in which the selected approach is implemented can affect the results, the implementation should be clearly explained, with key assumptions noted.	B	This item does not adversely affect the technical adequacy of the PRA. The methodology used for the CPSES HRA was considered appropriate and was found acceptable by the NRC. HRA methods have evolved and improved over time. Guideline R&R-DI-005 "Human Reliability Analysis," was revised to ensure HRA updates use current, clearly defined methodology, data and tools. The current revision of the guideline uses the EPRI HRA Calculator to 'quantify' the HEP values.
HR-06	Improve HRA documentation for operator action time window basis.	B	This item does not adversely affect the technical adequacy of the PRA. Documentation exists in previous analysis referenced by the current documentation. Guideline R&R-DI-005 "Human Reliability Analysis," was revised to reduce references to previous analysis such that analysis traceability is improved. Revision 3 of the HRA analysis provided enhanced documentation of the Operator action, time windows available and time required to perform the action.
HR-10	Evaluate cutsets with multiple human errors and revise dependency calculations if necessary.	A	This item was found not to adversely affect the technical adequacy of the PRA. A PRA utility program identified unique combinations of multiple human actions. These were reviewed based on the scenario to ensure dependencies were identified and handled as appropriate. This process of evaluating cutsets with multiple human errors is included in the quantification guide (R&R-DI-002) and shown in the revised HRA notebook.
IE-02	The process for developing the loss of offsite frequency at CPSES involves screening events from an EPRI database. This screening process is somewhat subjective and leads to questions concerning deletion of events. A process more accepted in the industry is to take a generic distribution and Bayesian update with plant specific in formation. The frequency obtained is approximately the same as the CPSES frequency but is simpler and easier to defend.	B	Use of the recommended Bayesian update process is not appropriate for the EPRI data because it already contains CPSES data and a Bayesian update would result in double counting. The data screening performed by CPSES is straightforward and is defendable (e.g. screened out events involving salt spray, etc.) The actual value currently being used at CPSES was considered to be adequate by the peer reviewer. No action needed.

IE-04	Include the other unit SSW pumps in the common cause group	B	A 4/4 failure of the site Service Water pumps was input into the Dual Unit Model (PRA model Rev. 2) during the time of the peer review. The change is documented in R&R-PN-006 "Service Water System" No additional action is needed.
IE-05	The ISLOCA analysis does not include a correlation of variables for cutsets that contain, for a given lambda, a lambda squared term. This is a required step, as described in such documents as Volume 5 of NUREG/CR-4350, NUREG/CR-5102, and NUREG/CR-5744.	B	The ISLOCA analysis was performed using the guidance from NSAC-154 "ISLOCA Evaluation Guidelines" which does not include the described lambda squared term. This methodology is judged to be acceptable and no action is needed. The Nuclear Safety Analysis Center (NSAC) is operated by EPRI.
L2-01	Incorporate flooding sequences in the LERF calculation	B	This item does not adversely affect the technical adequacy of the PRA. Flood sequences potentially impact containment spray and containment isolation. However, CPSES has a large dry containment and important containment isolation valves fail closed such that containment spray and isolation have a small impact on LERF. No action needed.
L2-03	The Steam Generator Tube Rupture contribution to LERF appears to be unusually low relative to contributions typically found in other PRAs. Address the potential for SGTR to be under represented in the LERF analysis.	B	Steam Generator related modeling observations were evaluated incorporated and documented as appropriate during implementation of the Dual Unit Model (PRA model Rev. 2). Results of the requantification resulted in a SGTR LERF contribution change from less than 1% up to 18%. This is a significant increase that clearly indicates the potential for SGTR is represented in the new PRA model. No additional action is needed.
L2-04	Expand on the analysis of LERF contributions to discuss contributions from containment failure modes including those mapped in from the IPE, and provide a perspective on the degree of conservatism inherent in the current LERF model, to support LERF sensitive applications.	B	This item does not adversely affect the technical adequacy of the PRA. Sufficient information is available to derive LERF contribution conclusions and additional documentation has been added to the quantification notebook that addresses LERF contribution from initiating events and equipment. Furthermore, core damage frequency dominates risk importance considerations at CPSES. Revised R&R-DI-007 "Containment Performance Analysis" so that LERF contributions are clearly documented when Level 2 analysis updates occur.
L2-05	Potential for SGTR to be under represented in the LERF analysis because the success criteria for SGTR appear to have misapplied the 24 hour mission time concept.	A	Steam Generator related modeling observations were evaluated, incorporated and documented as appropriate during implementation of the Dual Unit Model (PRA model Rev. 2). These changes included consideration safety impact beyond 24 hours for Steam Generator Tube Ruptures. Results of the requantification resulted in a SGTR LERF contribution change from less than 1% in the Rev. 1 PRA model up to 18% in the Rev 2 PRA model. This is a significant increase that clearly indicates the potential for SGTR is represented in the new PRA model. The success path basis, specific model changes and requantification results are documented in R&R-PN-004 "Auxiliary Feedwater System", R&R-PN-013 "Accident Sequence Analysis" and R&R-PN-022 "Accident Sequence Quantification". No additional action is needed.
L2-07	Review the level 2 analysis and remove conservatisms as they relate to severe accident phenomena. The dominance of Loss of Offsite Power (LOOP) in the LERF analyses could mask other LERF contributions.	B	This item does not adversely affect the technical adequacy of the PRA. Core damage frequency dominates risk importance considerations at CPSES. In the case of LERF, core damage bins (accident sequence groups) provide the input from the Level 1 to Level 2 PRA analysis. The core damage bin that contains LOOP sequences also contains other sequences that require similar containment response. Therefore, it is not appropriate to imply that LOOP could mask other LERF contributions. Conservatisms are associated with industry accepted Level 2 methodologies in use at the time the analysis was performed. Revised R&R-DI-007

			"Containment Performance Analysis" to consider the latest methodologies when performing Level 2 analysis updates.
QU-06	Perform a parametric uncertainty analysis sufficient to characterize CDF/LERF as mean values.	B	A parametric uncertainty analysis is not necessary at CPSES because sensitivity studies are performed to address uncertainties on a case by case basis. Sensitivity analysis guidelines were included in desktop instruction R&R-DI-014 "PRA Applications". The revision 3 of the quantification notebook provides additional insights with respect to sensitivity analyses that were performed for key assumptions and modeling approaches.
TH-01	Large break analysis Modular Accident Analysis Program (MAAP) code and success criteria issues for large LOCAs. The MAAP code has been used for several analyses supporting success criteria bases. Most of these analyses are within the generally accepted capabilities of this code. However, at least one of the analyses was intended to determine requirements for ECCS injection following a large LOCA (case sb6in4, RXE-LA-CPX/0-062 R0). The MAAP code, even recent versions such as MAAP 4.0, is generally not accepted as providing accurate results for the early blowdown phases of certain classes of large breaks (or for certain rapid depressurizations in general). Hence, its usage in this case may not provide a defendable outcome. Although MAAP 4 addresses some of the documented limitations of MAAP 3B for analysis of this class of events, care is still required in its application. For example, while MAAP 4 may reasonably represent plant response to large hot leg breaks, it may not be appropriate for use in predicting plant response for some larger cold leg breaks. Further, in the discussion in RXE-LA-CPX/0-062 R0 on LLOCA, there is a statement that "The 6" break can be successful without CVCS PUMPs, SIPs, or AF, but accumulators (2/4 should be adequate) and 1 train of RHR are required, based on MAAP4 run sb6in4." The basis for determining that 2/4 accumulators is adequate is not stated, and therefore must be interpreted as a judgment by the analyst. A similar judgment is made regarding requirements for accumulators for the larger end of the break spectrum. Additional	B	The concern with MAAP providing valid results for the early blowdown phase of LOCAs, applies to cold leg breaks. The concern is that because it doesn't have a momentum equation, the code cannot capture the ECCS bypass phenomenon. This is a short lived phenomenon that lasts on the order of 15-20 seconds for double-ended guillotine breaks. Accumulator water from intact cold legs, instead of falling into the downcomer and entering the core, is sucked around the periphery of the downcomer and out the break levitated on the reversed downcomer steam flow caused by the blowdown. There are several reasons why MAAP's inability to model this phenomenon is not a concern for the conclusions in calculation RXE-LA-CPX/0-062. First, large break LOCAs are defined in PRA as breaks between 6" and DEG. Therefore, the success criteria must be determined over this range. The Accumulator and LPI success criteria are dictated by the DEG and those breaks, while affected by the bypass phenomena, were analyzed with TXU Electric's large break LOCA 10CFR50.46 Evaluation Model, which captures it. The lower end of the range (6") is examined to determine the need for AF and the need for high head injection. That is where the SB6IN4 MAAP4 run was used. For that purpose, for which the 6" break was used, the ECCS bypass phenomenon has no bearing on results. This is because the cladding heat up occurs around 1800 seconds at which time any bypass would be over, if it ever even occurred, given that with the much smaller break flow, there will always be substantial downflow in the downcomer in all phases of the accident. Therefore the conclusions of SB6IN4 and the validity of MAAP4 for the application stand. Regarding the second statement regarding use of engineering judgment to conclude that 2/4 accumulators are sufficient for the large LOCA, although all 4 are used in EM models, that judgment is based on extensive experience in LOCA analysis. The EM's must include the features of 10 CFR 50.46 Appendix K which are not required of the PRA success criteria, which can be a best estimate analysis. Of these requirements, the 1.2 multiplier in decay heat and the zero heat transfer coefficient between end of bypass and BOCREC alone, more than offset the 2 accumulators. Nevertheless, a calculation using TXU Electric's EM model (with the Appendix K required inputs off) was performed for the conditions of the success criteria and the PCT was found to be ~300 F lower than the licensing basis PCT. This calculation is documented in the new revision (Rev. 1) of RXE-LA-CPX/0-062. Thus, the conclusion of the previous revision is unchanged. Rev. 1 merely provides a calculation basis to reinforce the previous engineering judgment basis for the conclusion. No further action is needed

	justification / explanation of how the analysis results support this judgment should be provided.		
TH-02	Additional guidance is needed for success criteria basis development.	B	Analysis and methodology for PRA success criteria have been documented in a new PRA notebook, R&R-PN-040, "PRA Success Criteria Notebook."
TH-03	Small Break LOCA success path with AF failed - provide thermal hydraulic analysis or remove from event tree	B	Although RXE-LA-CPX/0-062 states that success cannot be achieved without AF, that does not mean that success cannot be achieved with the next procedural evolution, namely, feed and bleed. What the cited run (SB2IN5) showed was that success could not be achieved a priori without AF, as it can with larger breaks. The discussion is about the need for AF for LOCAs. The larger break ranges are shown not to require AF for success. Feed and bleed follows procedurally any loss of secondary cooling, which would result from say the unavailability of AF. Cases which require AF but where AF is not available will then move to the next recovery evolution: feed and bleed. The success criteria for feed and bleed are given in Section 2.9 of RXE-LA-CPX/0-062. Thus, for small LOCAs, Table 6 in that section applies, and in fact, it would be conservative, since the depressurization from the break itself would help with the bleed part of the feed and bleed evolution. Therefore, the question is answered here and a note was added to the affected calculation. No action is needed
TH-04	The basis for the PRA success criteria analyses should be a clearly-stated definition of core damage that is suited to the analytical tools used.	B	The success criterion is the one described in the conclusions section of RXE-LA-CPX/0-055. The observation that the success criterion should be placed "up front" will be addressed by placing the success criterion in the "Success Criteria Notebook" that has been developed. The various comments on the actual success criterion used are addressed by the following clarification of the CPSES success criterion. Note that the discussion below merely clarifies the CPSES success criterion to address the issues raised in this observation, but the criterion itself is unchanged from what has been used throughout the CPSES IPE and PRA. "The CPSES PRA criterion for success is avoidance of the significant core damage, associated with a severe accident. The word significant applies both to the degree of core damage and to how widespread that damage is. Thus, local occurrence of DNB or exceeding PCT locally is tolerated. This is because exceeding these criteria for hot rods and/or for hot channels, even though possibly resulting in very localized fuel damage, would not necessarily constitute a severe accident. This is an important consideration that distinguishes a PRA success criterion from acceptance criteria used in accident analysis. Accident analysis acceptance criteria are applied locally, i.e. to the hot spot, hot rod and/or hot channel to ensure that no part of the core, even the most minute fraction, would exceed the criteria. However, for there to be "core damage" in the PRA sense there must be damage to a broader region of the core. For the CPSES PRA, the breadth of damage is set to be 100%/7/11 = 1.3% of the core. This is accomplished by nodalizing the core into 7 radial regions and 11 axial regions in MAAP4 and having the PRA success criterion tested at the hottest node. While arbitrary, this criterion that at least 1.3% of the core must exceed the success criterion is in line with the 10CFR50.46 acceptance criterion for LOCA that sets core-wide oxidation at 1%.

			Regarding the degree of damage, the PRA success criterion is as conservative as, or perhaps more so, than that of 10CFR50.46 for LOCA analyses. The core damage criterion is the onset of oxidation of this 1.3% of the core. That was translated as TCRHOT, the hottest core nodal temperature in MAAP, should be less than 1500 K (~2200 F). This temperature marks the onset of the exothermal Zirconium-water reaction which precedes significant Zr oxidation, eventual clad embrittlement and damage. These core nodal temperatures (TCRHOT) are radial averages across a representative fuel rod for that region, while the oxidation threshold (~2200 F) applies to the clad temperature. This means that this criterion is applied conservatively because the average pin temperature is higher than the clad surface temperature, which is subject to oxidation." An issue is also raised in this observation that uncertainties in the MAAP calculations require that the success criterion itself add conservatisms to bound these uncertainties. All calculational models are analytical representations. There is always a mismatch between the actual phenomenon and its calculation. The standard for phenomenology calculations involving severe accidents is "best estimate". The MAAP models were benchmarked against a licensing version of RELAP5/MOD2 in RXE-LA-CPX/0-055 for feed and bleed calculations and found to provide equivalent results. Therefore, the CPSES MAAP results are not more uncertain than recognized analytical methods and to select an overly conservative success criterion that bounds uncertainties defeats the purpose of PRA and is at odds with the universally accepted "best estimate" standard. Definition of core damage is documented in the new PRA notebook, R&R-PN-040 "PRA Success Criteria Notebook"
TH-07	Clarify the definition of "stable condition" and check that modeled end states are consistent as practical across modeled sequences.	B	PRA success criteria and the definition of "stable condition" has been included in the PRA notebook, R&R-PN-040 "PRA Success Criteria Notebook"
TH-08	Clarify the basis and success paths for the steam generator tube rupture model and modify the model if necessary.	A	Steam Generator related modeling observations were evaluated and it was determined that changes to PRA event and fault trees were needed for long term cooling after a steam generator tube rupture event. These changes were incorporated into the Dual Unit Model (PRA model Rev. 2). The success path basis, specific model changes and requantification results are documented in R&R-PN-004 "Auxiliary Feedwater System", R&R-PN-013 "Accident Sequence Analysis" and R&R-PN-022 "Accident Sequence Quantification". No additional action is needed.
TH-09	Provide references to specific thermal hydraulic analyses, or other bases, for accident sequence timing, including the time available to operator actions.	B	Revised guideline R&R-DI-005 "Human Reliability Analysis" to ensure appropriate references are made for time critical human actions. The Success Criteria notebook and other Thermal Hydraulic calculations provide time basis for the available window for operator actions or other accident sequence timings.

TXU Response to 1b:

Beginning in late 2004 and completed in 2005, CPSES embarked on its third and latest periodic update to the PRA model. This update encompassed data as well as system and top level logic changes. Prior to the start of this update, an internal gap assessment of the CPSES PRA model was completed using the ASME PRA standard as guidance. Items of significance from this assessment were addressed as part of the revision 3 update. The PRA update included:

- Updating the PRA model to reflect the plant as-built configuration including all changes made since 2000.
- Updating component failure rates and unavailabilities with plant-specific data where available.
- Updating the initiating event frequencies with plant-specific data where available.
- Loss of Off-site Power (LOOP) initiating event frequencies were modeled as their constituent parts (Grid, Plant and Weather- Centered events). Consequential LOOP and degraded grid conditions were also included in the PRA model. These frequencies were also updated using industry data collected by EPRI.
- Updating the latent, dynamic and recovery human reliability analysis (HRA) using the EPRI HRA Calculator software.
- Implemented the Westinghouse 2000 RCP seal modeling, including NRC SER recommendations.
- Updating the Thermal-Hydraulics analysis used to develop core uncovery times associated with seal LOCA scenarios.
- Updating the model and associated documentation to reflect WOG and Peer review comments. Remaining category A & B F&Os (documentation) from the WOG Peer Review were incorporated into the update documentation as well as other documentation issues identified during that process.

An Independent Industry Peer review of the Revision 3 changes associated with the RCP seal LOCA model, T-H analyses associated with seal LOCA scenarios, LOOP model changes (discussed above) and quantification process was completed. This review was completed based on ASME PRA Standard. No category A or B F&Os were identified by the peer review and other F&Os items were resolved and incorporated into Revision 3B of the model.

The following is a list of peer reviewer credentials:

1. A utility peer reviewer that has over 20 years of engineering experience with at least nine years of PRA modeling and evaluation. At the time of the independent review he was a utility PRA engineer providing PRA support services including model updates, external event model development, on-line risk monitor development, Human Reliability Analysis (HRA), PRA training, and Peer Reviewer for Prairie Island.
2. A utility peer reviewer that has over 30 of engineering experience with at least 25 years of PRA modeling and evaluation. At the time of the independent review he was a utility PRA Supervisor providing PRA support services including model updates, external event model development, on-line risk monitor development, Human Reliability Analysis (HRA), PRA training, and Peer Reviewer for three plants in the USA and one in Korea. He was a guest lecturer at MIT for the course "PRA for Managers."
3. An industry consultant that has 30 years of experience in areas of engineering analysis, system reliability analysis, safety analysis, Probabilistic Risk Assessment (PRA), project

management, design engineering, and power plant operation. He is a registered professional engineer in the states of California and North Carolina and a member of the American Society of Mechanical Engineers (ASME). At the time of the independent review he was an independent consultant providing PRA support services including model updates, external event model development, on-line risk monitor development, ILRT Extensions, Risk-Informed ISI, SAMA Support, Risk-Informed Tech Specs, and PRA training.

4. An industry consultant that has 20 years of experience in Probabilistic Risk Assessment (PRA). At the time of the independent review she was an independent consultant providing PRA support services including model updates, external event model development, on-line risk monitor development, Risk-Informed ISI, SAMA Support, Risk-Informed Tech Specs, and PRA training.

This current version of the CPSES model is used in support of the RI-ISI process. There are no outstanding A or B category F&Os from the WOG peer review process or from any of the other third party independent reviews.

As part of the Westinghouse Owners Group (WOG) industry participation in the MSPI, the WOG performed a cross comparison and assessment of monitored components and PRA results used in the implementation of NEI-99-02 for establishing Mitigating Systems Performance Indicators (MSPI). This cross comparison was to be done across the entire fleet of Westinghouse and Combustion Engineering designed plants. The cross comparison has been given significant importance due to an NEI/NRC agreement to substitute the cross comparison as a vehicle for resolving PRA quality issues relevant to MSPI before implementation. The results of that effort identified Comanche Peak as presenting potential outliers in two areas which were subsequently resolved. Candidate outliers were established based on the plants Birnbaum value being either relatively low or high for those in its "group" and/or the observed presence of large component asymmetries. The information provided to the industry peers and NRC established an understanding of the reasons for those risk importance measures being considered as potential outliers. That information was reviewed and accepted and the technical adequacy of the Comanche Peak PRA was found to be acceptable for generation of risk based MSPI metrics. There are no open items associated with the WOG cross comparison and assessment effort.

2. <u>NRC Question:</u>

Partially as a result of the re-performance of the RI-ISI analysis, and partially due to your inclusion of 4 NPS Class 2 Auxiliary Feedwater piping into the RI-ISI scope, Table 1 indicates 12 additional inspection locations in the CPSES 2 FWS system and 9 additional inspection locations in the CPSES 2 AFW system proposed for the second interval, relative to the proposed inspection locations in those systems from the original RI-ISI submittal. All 21 of these locations are in High Consequence segments, susceptible only to the flow-accelerated corrosion (FAC) damage mechanism (DM). Due to your Generic Letter 89-08 augmented inspection program for FAC you were able to place the segments of these welds into Risk Category 4 (medium) (as opposed to Risk Category 1 (high)), requiring inspection of 10% (rather than 25%) of these welds in each of the two systems.

a. Please describe the type of non-destructive examination(s) you intend to perform on these 21 welds, and whether or not you intend to credit the Generic Letter 89-08-

related augmented inspection program examinations toward the completion of RI-ISI-required inspections of these 21 locations.

b. If you intend to credit the Generic Letter 89-08-related augmented inspection program examinations toward the completion of RI-ISI-required inspections for these 11 locations, please explain your rationale for doing this, given the nature of the examinations performed for FAC-susceptible locations.

TXU Response:

Ultrasonic examinations will be performed on the 12 FWS and 9 AFW welds during the second interval of Unit 2. These examinations are identified in the latest revision of the Unit 2 ISI Program Plan.

The principal concern discussed in Generic Letter 89-08 is managing localized wall thinning caused by erosion/corrosion or FAC such that the pressure boundary materials are not allowed to degrade to a point where the operating and transient forces cause the pressure boundary material to experience stress levels that exceed the applicable code design values.

No credit is given with regards to Generic Letter 89-08 towards the completion of these 21 required examinations. FAC examinations at CPSES monitor local wall thicknesses of pipe on a grid system for a determined population of components, following the guidance provided in EPRI document NSAC-202-L.

3. **NRC Question:**

In your original RI-ISI submittal, dated February 15, 2001, you stated that a deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for the potential for thermal stratification, cycling and striping (TASCS). In your response to NRC Request for Additional Information, dated July 20, 2001 (corrected date), you clarified that the methodology for assessing TASCS in the CPSES RI-ISI program is identical to the methodology described in the Electric Power Research Institute (EPRI) letter to NRC dated March 28, 2001, and indicated that you will update the RI-ISI program based on the final EPRI Materials Reliability Program (MRP) guidance as warranted.

a. Please confirm that, upon issuance, TXU will update the CPSES RI-ISI program to incorporate NRC-approved final MRP guidance on thermal fatigue management for assessing TASCS.

TXU Response:

After completion of the assessment of thermal fatigue and TASCS per the requirements of MRP-146, these results will be reviewed and incorporated into the Unit 1 and 2 ISI Program Plans.