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February 10, 2000

LICENSEE: Omaha Public Power District

FACILITY: Fort Calhoun Station, Unit No. 1

SUBJECT: MEETING SUMMARY - ABB-CE REPORT CEN-636, REV. O AND JAPANESE DATA FOR RT_{PTS} DETERMINATION

On January 6, 2000, a public meeting was held between NRC staff, Omaha Public Power District (OPPD), and OPPD's contractor, ABB-Combustion Engineering (ABB-CE), at the NRC headquarters offices in Rockville, Maryland. The purpose of the meeting was to discuss the staff's concerns with the OPPD submittal CEN-636, Revision 0, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials - Basis for Prediction of RT_{PTS} at Expiration of License." The staff had documented its concerns regarding CEN-636, Revision 0 in a letter dated November 30, 1999. Attachment 1 lists the meeting participants. The presentation slides used for the meeting are included as Attachment 2. Attachment 3 lists evaluation criteria for surveillance data applicable to Fort Calhoun. Attachment 4 lists items of staff concern regarding quality assurance of surveillance data.

At the beginning of the meeting, OPPD indicated that they plan to submit an application for license renewal to the staff at the end of 2002. Approval of a renewed license would extend Fort Calhoun's period of operation to 2033.

After introductions and a brief overview of the meeting, ABB-CE presented information on proprietary Japanese data from Mihama Unit 1. Mihama Unit 1 has the limiting Fort Calhoun weld wire heat combination (12008/27204) in their surveillance program. Preliminary results indicate that use of the Mihama data would lower the RT_{PTS} value for the limiting vessel weld, however, the data needs to be further verified and adjusted for temperature and chemical composition. During the presentation OPPD noted that, without the use of surveillance data and a reduced margin to calculate RT_{PTS} for the limiting weld, Fort Calhoun would exceed the pressurized thermal shock (PTS) screening criterion before the current license ends. OPPD also indicated that using the current rate of embrittlement, welds fabricated using tandem weld wire heat 27204/27204 would exceed the PTS screening criteria before the proposed license renewal period ends. They proposed to use surveillance data from Diablo Canyon Unit 1 and from a supplemental capsule in Palisades to calculate RT_{PTS} with a reduced margin to determine the rate of embrittlement for welds fabricated with tandem heat 27204/27204. One data point in the 27204/27204 data set is not credible; however, the measured value from the capsule is less than the predicted value.

OPPD discussed possible benefits of future initiatives such as improved embrittlement correlations and the PTS rule re-evaluation. These initiatives will probably not be completed early enough to be useful for the proposed application for license renewal in 2002. In order to ensure that all RPV welds are below the PTS screening criteria before the license renewal period ends, the licensee proposed to do the following:

DF01

1. Use the Mihama Unit 1 surveillance data for weld wire heat 12008/27204 to calculate a chemistry factor in accordance with Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," and use a reduced margin to calculate RT_{PTS} for the limiting weld in the Fort Calhoun vessel. Use of the Mihama data will include a temperature adjustment to account for differences in the Mihama and Fort Calhoun operating temperatures, a chemistry adjustment to account for differences in the Mihama surveillance weld and the best estimate chemistry for heat 12008/27204, and any other adjustments based on the difference in the nuclear environments of the Mihama and Fort Calhoun reactor pressure vessels (RPV). (See attachment 3).

2. Use the Diablo Canyon Unit 1 surveillance data for heat 27204/27204 and a supplemental capsule from Palisades to calculate a chemistry factor in accordance with RG 1.99, Revision 2, and use a reduced margin to calculate RT_{PTS} for the second most limiting weld in the Fort Calhoun vessel.

3. Evaluate the neutron fluence using a "fleet bias" in fluence. In addition, the staff requested that the licensee provide the impact of fluence on the RT_{PTS} value (using a biased or an unbiased value).

The licensee's presentation included responses to the staff's concerns with CEN-636, Revision 0. The report proposed to use surveillance data from Westinghouse designed plants with reactor pressure vessels fabricated by CE to demonstrate that the Fort Calhoun surveillance welds are credible and that the RT_{PTS} may be calculated using a reduced margin (reducing the sigma delta standard deviation by ½). Specifically, the licensee proposed to use a chemistry factor calculated from the RG 1.99, Revision 2 tables and the best-estimate chemistry for tandem weld wire heats 12008/27204 with a reduced margin. The licensee analyzed surveillance data from CE vessels that were fabricated about the same time as the Fort Calhoun vessel, and proposed the analyses as the basis for a reduced margin. The staff's major concern was that none of the data used as the basis for the evaluation had the weld wire heat combination of the limiting Fort Calhoun vessel weld. At the time the report was written, the licensee did not have the Mihama Unit 1 data. In light of the newly acquired data, the licensee agreed to withdraw the initial submittal and submit a new license amendment application by the end of March.

Since OPPD is proposing to use surveillance data from Diablo Canyon 1, Palisades and Mihama 1 to establish RPV integrity, OPPD must compare the nuclear environments for these RPVs to that of Fort Calhoun and must indicate when additional capsules from those RPVs will be withdrawn and evaluated by OPPD. (See Attachment 4).

Before the meeting concluded, the licensee agreed to provide revisions to the Fort Calhoun surveillance data in order for the staff to update the reactor vessel integrity database (RVID). The approximate time agreed for the submittal was one month from the date of the meeting.

February 10, 2000

The participants also agreed to have another meeting before the new CEN -636 report is submitted to the staff.

/RA/

L. Raynard Wharton, Project Manager, Section 2
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Attachments: 1. Meeting Attendees
2. Presentation Slides
3. Evaluation Criteria
4. Items of Staff's Concerns

cc w/atts: See next page

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JANUARY 6, 2000 MEETING WITH OMAHA PUBLIC POWER DISTRICT

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ATTACHMENT 1

OMAHA PUBLIC POWER DISTRICT
Fort Calhoun Nuclear Station

Meeting with NRC to Discuss Reactor Vessel Integrity Options for Current and Renewed License Terms

January 6, 2000

ATTACHMENT 2

Agenda

- Purposes & Objectives
- Fort Calhoun Station Reactor Vessel Integrity History (Materials & Fluence)
- Recent Evolutions & Present Status
- Options for RV Life Attainment/Extension
- Expanded Topics
- Closing Remarks

Purposes

- Respond to NRC Staff concerns regarding RT_{PTS} analyses for Fort Calhoun Station's (FCS) current operating license to 2013
- Describe the methodology and actions anticipated to assure FCS reactor vessel integrity (RVI) beyond a renewed license term to 2033

Objectives
Current License (2013)

- Provide responses to NRC Staff questions on CEN-636 (Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials)
- Explain how OPPD will use the Mihama 1 surveillance weld data for FCS
- Obtain concurrence with methodology for analyses and application of uncertainties
- Obtain concurrence on one RAI during review and approval process

Objectives
License Renewal (2033)

- Obtain concurrence with methodology for RT_{PTS} analyses
- Obtain concurrence on application of uncertainties using current data, including Mihama 1 surveillance weld data
- Identify the importance of cooperative activities in progress that can result in beneficial rule changes
- Determine if NRC's schedule for approval of the results from these activities can support FCS license renewal submittal at the end of 2002

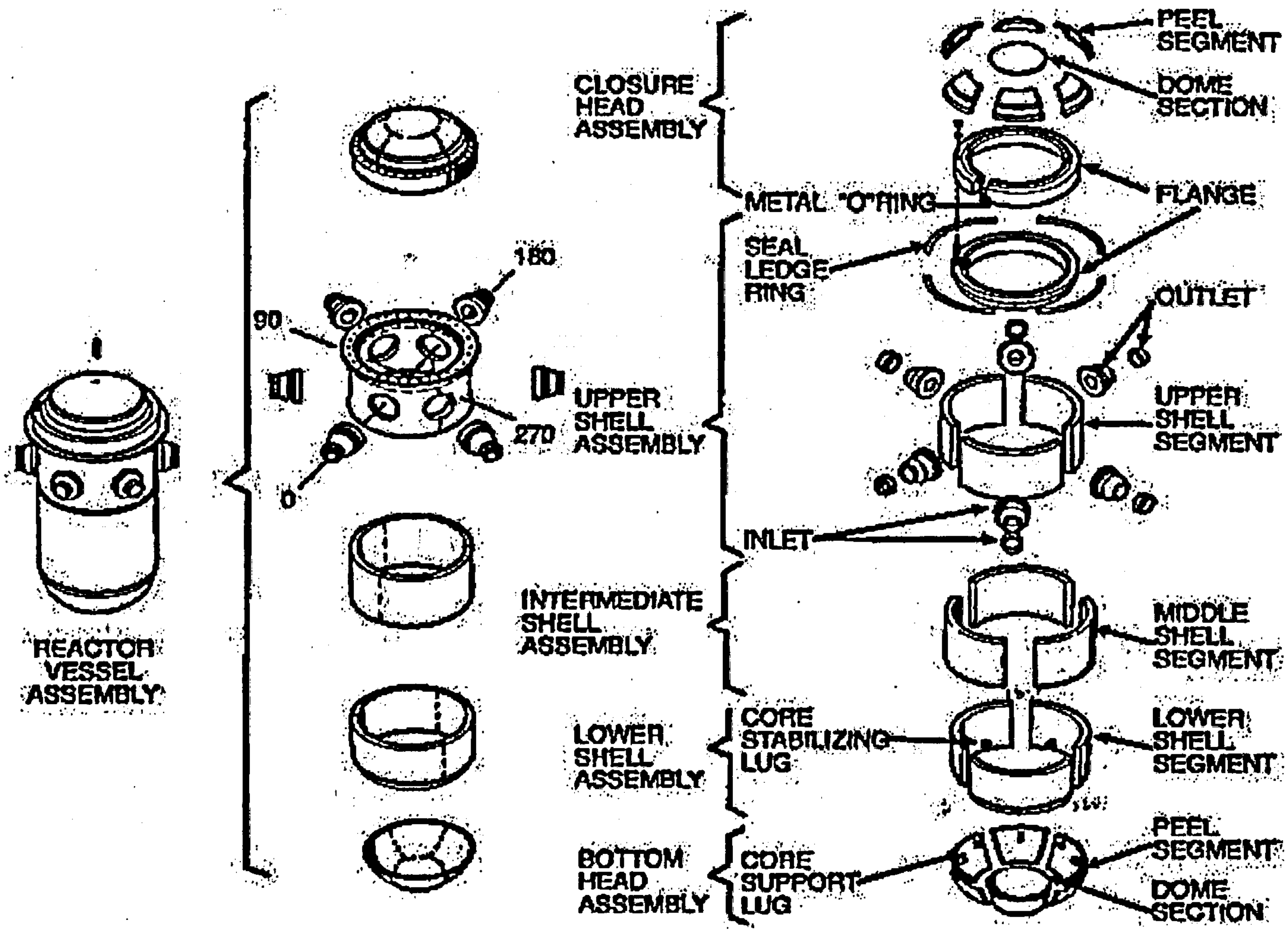
FCS RVI History: Materials

- FCS RV Fabricated 1967-1969 by CE
- Higher copper and nickel content of welds relative to newer vessels
- Axial 3-410A/C tandem-arc welds (60°,180°, & 300°) fabricated with weld wire heats 12008, 13253, & 27204. Fabrication records do not define specifics of weld composition. Most adverse combination conservatively assumed for entire weld.

FCS RVI History: Materials
(Continued)

- 27204/27204 combination was the most limiting in terms of chemistry factor for the 5-year Construction Period Recovery (Amendment No. 158 to Operating License DPR-40, 12/3/93)

Fort Calhoun Station Reactor Vessel



FCS RVI History: Materials
(Continued)

- CEOG Task 902 resulted in additional data showing limiting weld combination to be 12008/27204 vs. 27204/27204 (with a change in Chemistry Factor from 229°F to 231°F)

FCS RVI History: Fluence

- To achieve plant operation to 2013 without exceeding the 270°F screening criterion of 10 CFR 50.61, extreme low radial leakage fuel management was implemented in Cycle 14 with hafnium flux suppression rods and natural uranium assemblies to depress fast neutron flux near 3-410 welds
- Fluence Analysis using ENDF/B-IV cross-section library, benchmarked to PCA supported CPR

FCS RVI History: Fluence
(continued)

- Committed to re-perform with ENDF/B-VI when final R.G.-1025 issued
- ENDF/B-VI fluence analysis performed based on D.G.-1025/D.G-1053 (and submitted to NRC January 1998). 1999 review of FCS fluence analysis by the NRC rejected the use of the FCS plant specific bias, which was based on "benchmarking" the fluence analysis predictions to the W-225, W-265, and W-275 Surveillance Capsule results (i.e., actual fluence measurements translated to 1 $^1/_2$ inches to vessel wall)

FCS RVI History: Fluence
(continued)

- With application of this bias FCS is projected to reach the present 10 CFR 50.61 PTS screening criterion of 270°F for axial welds in April 2015
- Without application of this bias FCS is projected to reach the screening criterion in February 2009

Recent Evolutions & Current Status

- To gain margin available under Reg. Guide 1.99, Rev.02, Position 2.1, an analysis was performed (CEN-636) and submitted in November 1999
- NRC would accept this approach only with data that included 27204/12008
- December 1999: FCS received feedback and additional information from Kansai Electric Company (Japan) for the Mihama 1 plant which has the 12008/27204 weld heat combination in their surveillance program, with 3 capsules removed and evaluated along with pre-irradiation characterization

Recent Evolutions & Current Status
(continued)

- OPPD continues to receive and evaluate the Mihama data, and believes that this resolves the NRC concern on having specific data applicable to FCS in CEN-636
- OPPD plans to continue with a CEN-636 reanalysis and submittal of the revised report

Options for RV Life Attainment/Extension

- Position 2.1 analysis of Mihama 1 weld data, as discussed on previous slides. [Expected benefit of 21.5°F]
- Perform "fleet bias" assessment for fluence analysis and apply to FCS [Expected benefit of 3°F to 5°F, additive]

Options for RV Life Attainment/Extension

(continued)

- Demonstrate that FCS will not exceed the PTS screening criteria via the updated and improved embrittlement correlation. Continue participation in ASTM E10.02 for approval of ASTM E900 (NUREG/CR-6551). Support NRC subsequent adoption into Reg. Guide 1.99, Rev.03 and 10 CFR 50.61 [Expected Benefit of 30°F to 50°F, parallel path]

Options for RV Life Attainment/Extension
(continued)

- Participation in EPRI MRP for industry initiatives and focal point for interaction with NRC on PTS Screening Criteria Re-evaluation [Expected Benefit of 30°F to 60°F, parallel path]
- Perform R.G. 1.154 analysis for FCS parallel to the NRC PTS Screening Criteria Re-evaluation Project. [Expected Benefit of 30°F to 60°F, parallel path]

Options for RV Life Attainment/Extension

(continued)

- Pursue and apply Master Curve Approach /Direct Measurement of Fracture Toughness for PTS and other RV issues

Expanded Topics

- Available surveillance data
- Mihama 1 surveillance data
- Staff concerns with CEN-636
- Significance of ASTM E900 to Fort Calhoun

FCS Reactor Vessel

Identification of Reactor Vessel Plates and Welds in the Fort Calhoun Reactor Vessel Beltline

Plate or Weld Identification	Plate or Weld Electrode Heat No.	Weld Flux Type and Lot No.	Chemistry Factor (°F)[a]
Plate D4802-1	C2585-3	N/A	82.2
Plate D4802-2	A1768-1	N/A	65
Plate D4802-3	A1768-2	N/A	73.1
Plate D4812-1	C3213-2	N/A	83
Plate D4812-2	C3143-2	N/A	65
Plate D4812-3	C3143-3	N/A	65
Surveillance Plate	A1768-1	N/A	72.0[c]
2-410 A/C	51989	Linde 124, #3687	89.03
3-410 A/C	12008 & 13253 (T)[b]	Linde 1092, #3774	208.68
3-410 A/C	13253 (T)[b]	Linde 1092, #3774	189.05
3-410 A/C	12008 & 27204 (T)[b]	Linde 1092, #3774	231.06
3-410 A/C	27204 (T)[b]	Linde 1092, #3774	226.81
9-410	20291	Linde 1092, #3833	188.41
Surveillance Weld	305414	Linde 1092, #3947 and #3951	194

FCS Beltline Welds

Weld Seam	Heat No.	Chemistry Factor
2-410 A/C	51989	89.03
3-410 A/C	12008/13253	208.68
3-410 A/C	13253	189.05
3-410 A/C	12008/27204	231.06
3-410 A/C	27204	226.81
9-410	20291	188.41

Available Surveillance Data

- FCS difficult to analyze because of complex makeup of 3-410 welds and lack of detailed fabrication data
- Numerous other applicable surveillance data available

Available Surveillance Data

(Continued)

<u>Weld Heat</u>	<u>Credible Surveillance Source</u>
13253	D.C. Cook 1, Salem 2
12008, 13253	Test Reactor,EPRI/CRIEPI
27204	FCS Suppl. Capsule, Palisades Suppl. Capsule, Diablo Canyon 1
12008, 27204	Mihama 1

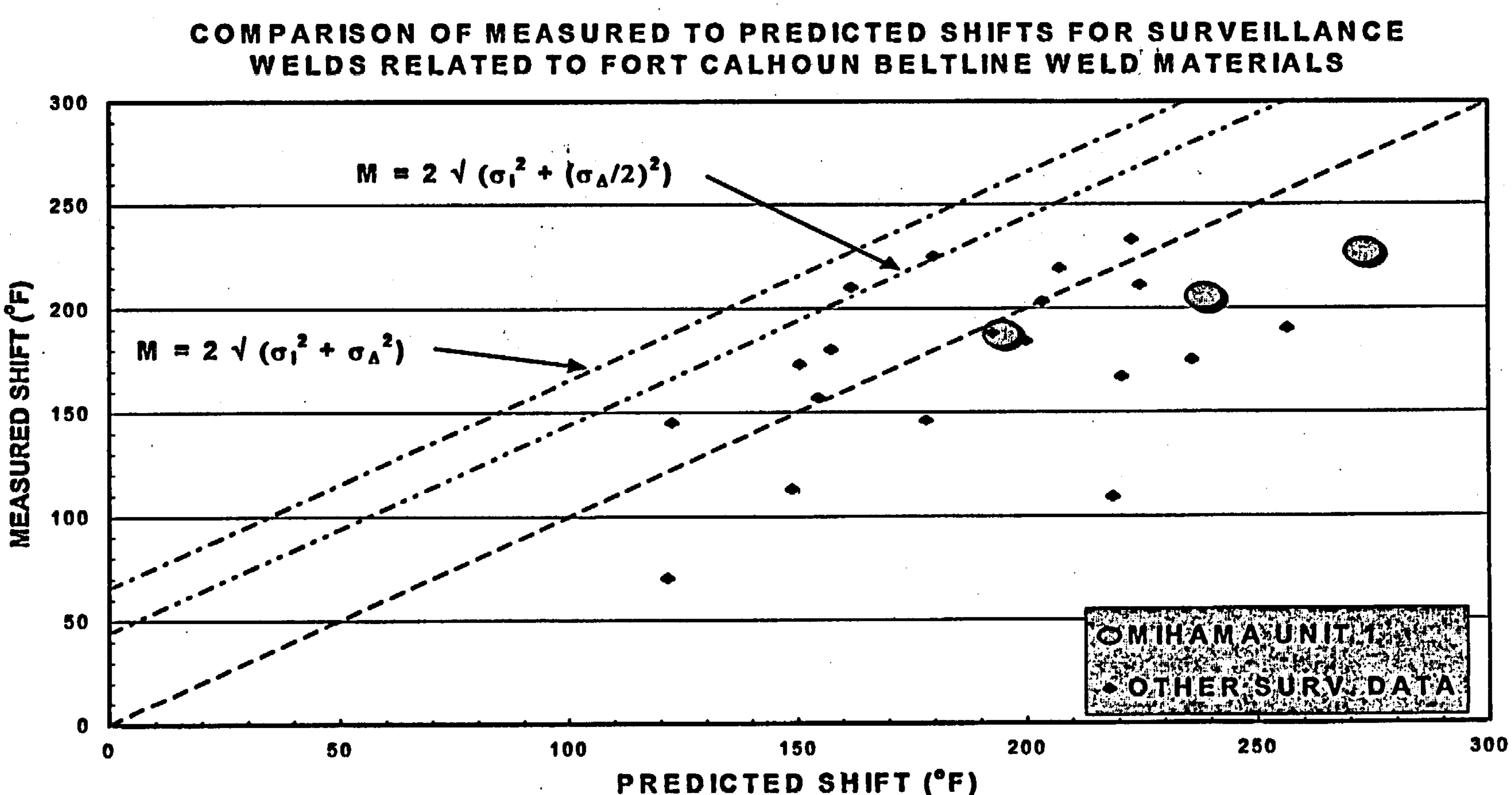
Mihama 1 Surveillance Data

- Action taken to contact Kansai resulted in data from Mihama 1 surveillance program
- Needed in recognition of "credibility" issue
- Significance to FCS: Mihama 1 surveillance weld is heats 12008 & 27204

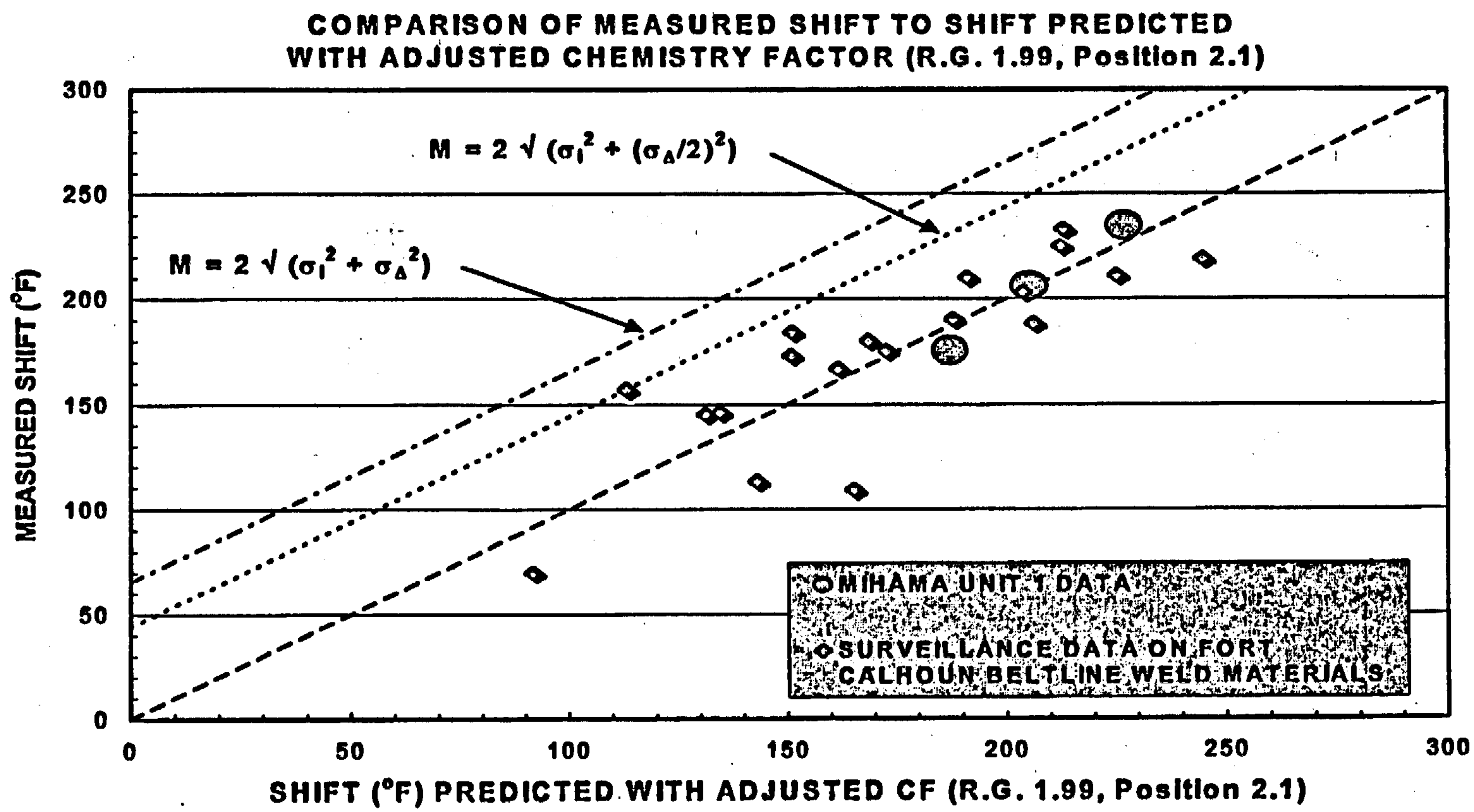
Mihama 1 Surveillance Data Evaluation Status

- Verified weld source via CE fabrication records
- Three capsules with exposures to 0.6, 1.2 and 2.1 E 19 n/cm^2 (E>1MeV)
- Have requested details to facilitate Position 2.1 analysis

Comparison of Measured vs. Predicted Shift



Comparison of Measured vs. Predicted Shift with CF Adjusted



Staff Concerns with CEN-636

- *Data credibility issue- need precise combination to use surveillance data from another plant; i.e., need data from heats 12008 & 27204*
- Response:
  - See analysis of Mihama 1 data
  - Used precise combinations for assumed vessel welds to show credibility

Staff Concerns with CEN-636

(Continued)

- Response to data credibility issue (continued):
  - Used set of similar combinations to show credibility for equivalent weld materials
  - Did not claim CF credit, only σ_{Δ} credit
  - Responsive to 10 CFR 50.61, pp.(b)(3) to consider "information... to improve the accuracy…"

Staff Concerns with CEN-636

(Continued)

- *Treatment of data scatter and predictability-exclusion of data without rigorous justification*
- Response:
  - CEN-636 analysis methods are consistent with those used in a similar Calvert Cliffs submittal and accepted by the NRC, exclusive of T-cold adjustment.

Staff Concerns with CEN-636

(Continued)

- Response to treatment of data scatter and predictability (continued):
  - data scatter and predictability will be addressed in the analysis of the Mihama 1 data
  - Use of the Mihama 1 data necessitates further evaluation of the next most-limiting weld composition (27204/27204)

Staff Concerns with CEN-636

(Continued)

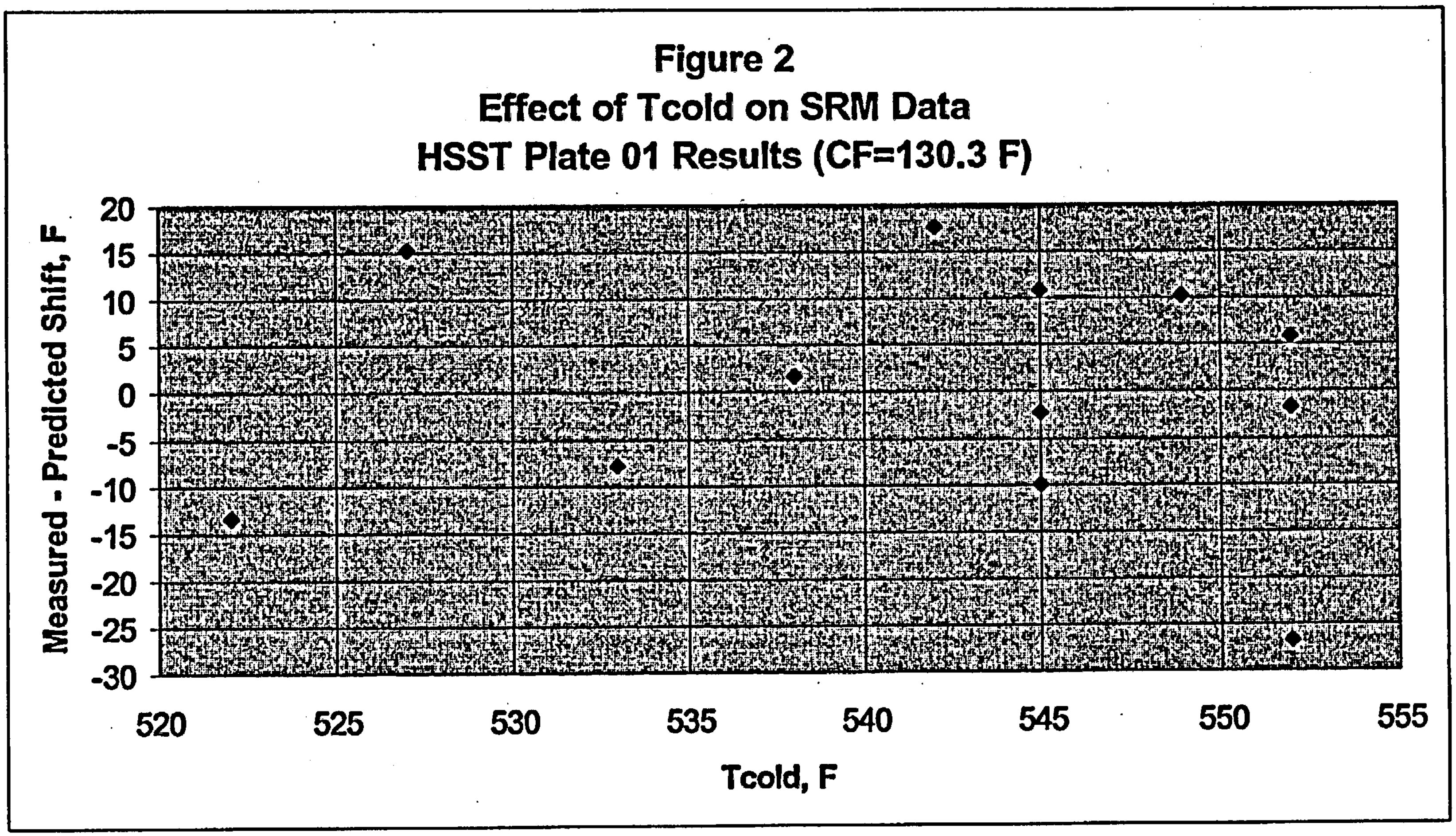
- *Treatment of irradiation environment- data are from W plants and were irradiated at 533 °F - 557 °F vs. Fort Calhoun at 527 °F - 538 °F*
- Response:
  - Surveillance data for FCS are a time averaged value of 527°F - 538°F, and current value is 543°F. Time-averaged value of approximately 540°F.

Staff Concerns with CEN-636

(Continued)

- Response to treatment of irradiation environment (continued):
  - Most relevant W data were at at 537°F - 542°F
  - CEN-636 showed no discernible effect on HSST 01 over range 522°F - 552°F; two FCS measurements included

Effect of T_{cold} on SRM Data



Staff Concerns with CEN-636

(Continued)

- Response to treatment of irradiation environment (continued):
  - Use of NUREG/CR-6551 correlation with temperature term is comparable to RG 1.99 without temperature term

Staff Concerns with CEN-636

(Continued)

- Response to treatment of irradiation environment (continued):
  - Conclusion: data and analysis of CEN-636 presently justify that no temperature correction is needed. Will clarify in Revision 1 of CEN-636.

Staff Concerns with CEN-636

(Continued)

- *Two of nine data points from 12008 and 27204 are not credible so can not use to reduce margin*
- Response:
  - Use of Mihama 1 data resolves issue

Staff Concerns with CEN-636
(Continued)

- *Differences between CEN-636 and RVID*
- Response: Proposed changes identified. Will formally document in letter

- *Need to discuss immediate and future plans for addressing RT_{PTS} for the Fort Calhoun reactor vessel*
- Response: Previously discussed

Significance of E900 to FCS

- Cooperative NRC/Industry effort to replace RG 1.99 RT_{NDT} shift prediction technique
- Development of ASTM E900 to incorporate prediction technique into consensus standard
- Latest balloted version of E900 contains equation from NUREG/CR-6551

Significance of E900 to FCS

E900 Approach:

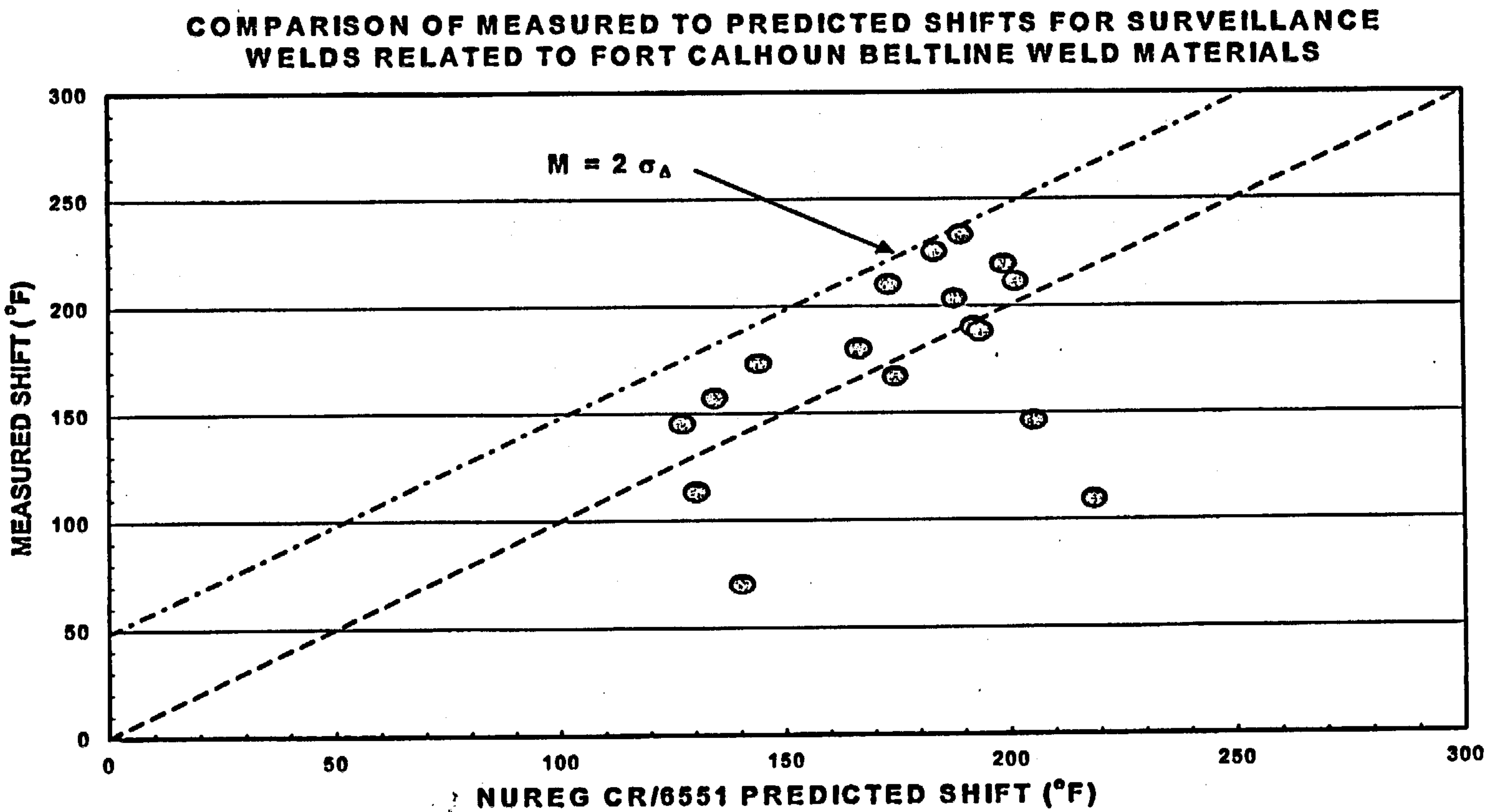
- Broad base of scrubbed surveillance data established high confidence in trends
- Future use of surveillance data to avoid subjective guidelines
- Surveillance data to be used only to indicate anomalous behavior

Significance of E900 to FCS

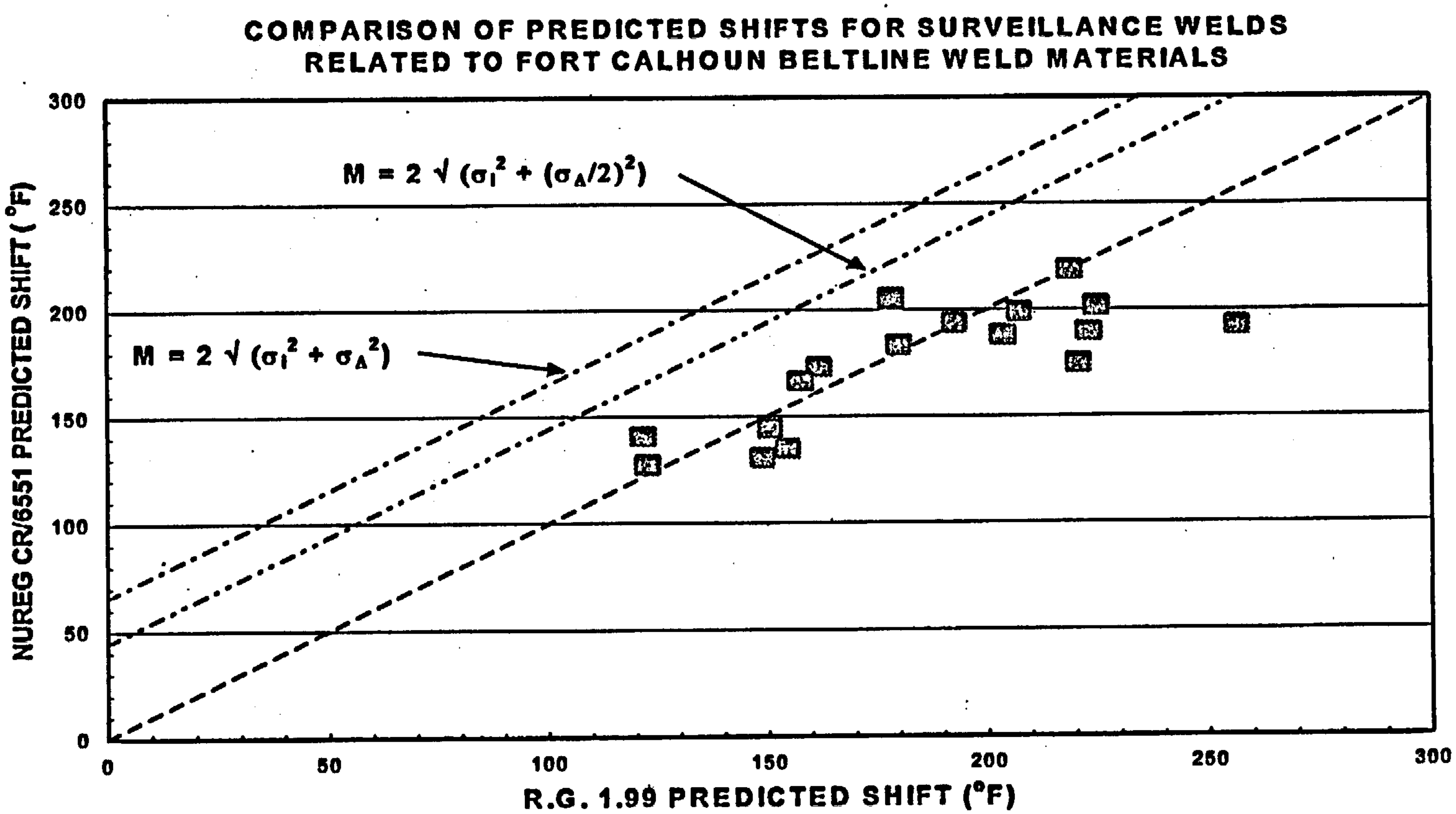
NUREG/CR-6551 Prediction:

- Correlation including time, temperature, fluence, P, Cu and Ni dependence
- RT_{PTS} prediction for FCS vessel is significantly less than with RG 1.99
- RT_{PTS} prediction for 2033 is below present screening criterion of 270°F

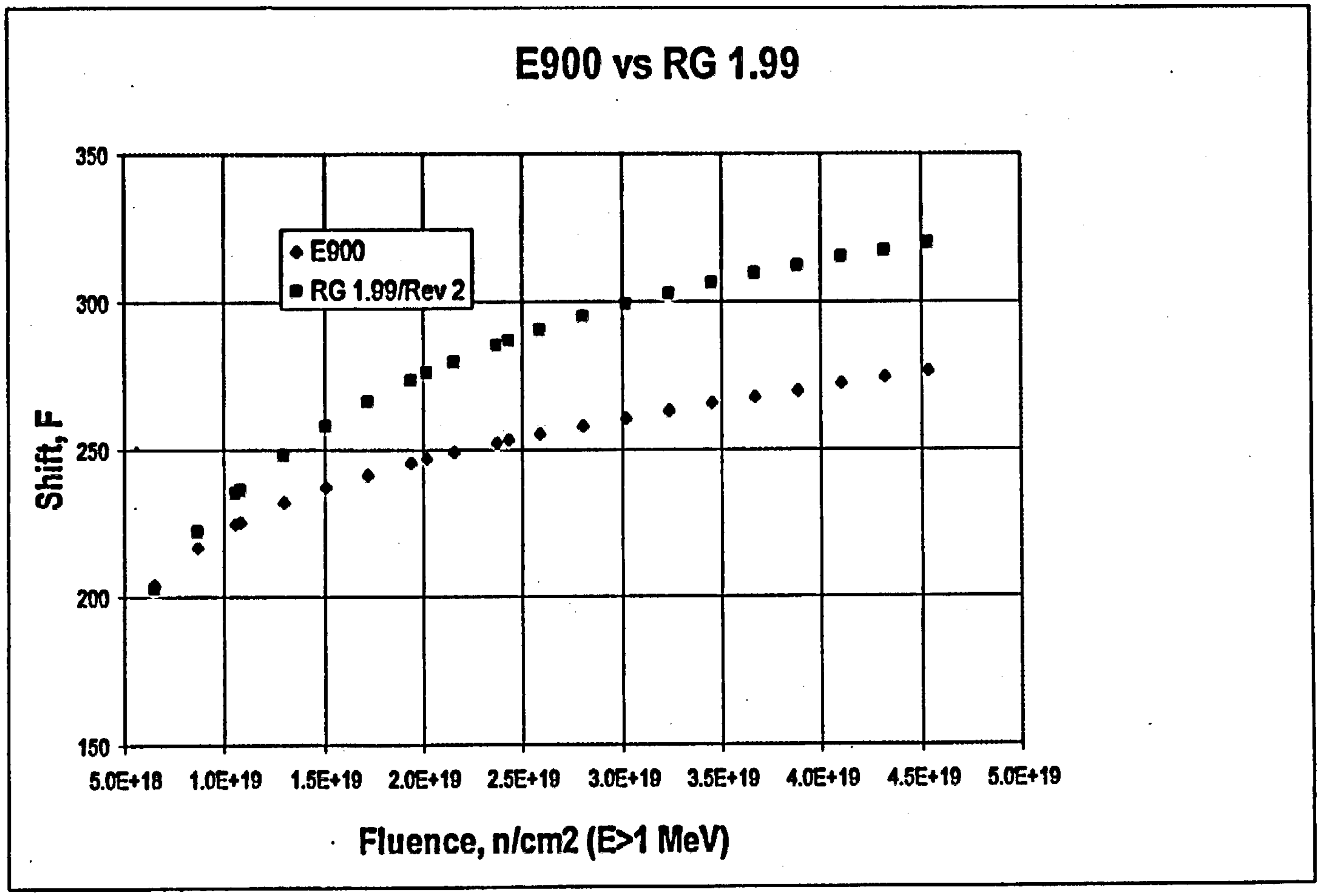
Comparison of Shifts for FCS Beltline Weld Materials



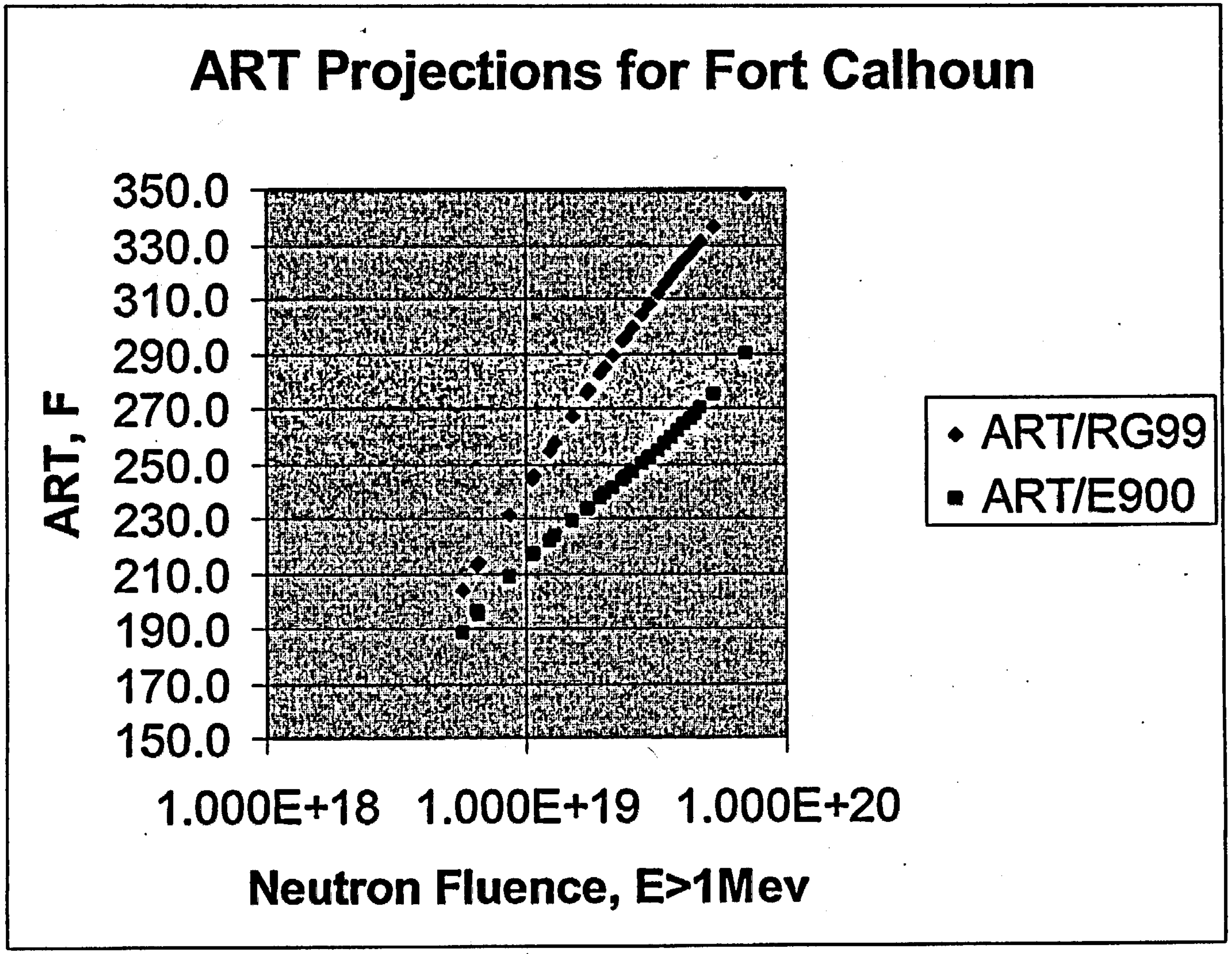
Comparison of Predicted Shifts for Surveillance Welds



ASTM E900 vs. R.G. 1.99, R2



FCS ART_{NDT} Projections



Derived CFs for Surveillance Welds

Vessel	Heat	RG 1.99 CF	Pos. 2.1 CF
Cook 1	13253	189.1	142.8
DC-1	27204	226.8	<217
DC-2	12008/21935	208.6	209
McG-1	12008/20291	200.4	146.2
Salem 1	13253	189.1	202.4
Mih-1	12008/27204	231.06	[ ]

205 Preliminary number

Summary

- Application of Mihama 1 surveillance data expected to satisfy Staff concerns on CEN-636
- Present and future surveillance data on 27204 will address next most limiting weld

Summary
(Continued)

- ASTM E900 prediction tool shows Fort Calhoun vessel will not exceed present PTS screening criterion until approximately 45 years after 2033
- NRC initiative on PTS screening criteria is expected to show that risk of vessel failure for FCS is acceptable beyond 2033

Summary

(Continued)

- Master Curve Approach demonstrates that CE fabricated welds have substantially more toughness margin than indicated by current RT_{NDT} approach
- One or more of the preceding conservatisms can be used to demonstrate vessel integrity beyond 2033 for the FCS reactor vessel

Closing Remarks

- OPPD has recognized the importance of RVI and has taken steps to prevent it from limiting the life of the RV and provide the option of a 20 year life extension:
  - Development and implementation of extreme low radial leakage fuel management

Closing Remarks

(Continued)

- Participation in industry groups to increase the total industry knowledge base associated with these issues:
  - CE Reactor Vessel Group: material properties and data improvements
  - NUMARC RVI AHAC: industry steering committee
  - CEOG RVWG & MWG: best estimate weld chemistry, other RV materials research, and fracture toughness activities
  - ASTM E10.02 (with interactions with ASTM E10.05): revised embrittlement correlation
  - EPRI MRP: industry steering group for RVI initiatives and focal point for NRC interface and interactions

Closing Remarks

(Continued)

- Use of the Mihama 1 surveillance data combined with completion by the industry and NRC of other initiatives previously identified should demonstrate integrity of the FCS reactor vessel beyond 2033
- OPPD will submit Revision 1 of CEN-636 within 60 days of final receipt/confirmation/release of the Mihama 1 surveillance data (approximately 3/31/00)

EVALUATION OF WELD DATA FROM THE JAPANESE MIHAMA UNIT 1 REACTOR (KANSAI ELECTRIC POWER COMPANY)

In order to determine if the weld surveillance data from the Mihama Unit 1 reactor vessel is applicable to the Fort Calhoun vessel, the following information will need to be evaluated:

1) Unirradiated and irradiated Charpy data for tandem weld wire heat 12008/27204

2) Irradiation temperature of the capsule based on PWR cold leg

3) Neutron flux of capsules

4) Gamma heating of capsules

5) Neutron spectrum of capsules

6) Chemistry of surveillance data

ATTACHMENT 3

QUALITY ASSURANCE CONCERNS

The licensee stated that the Mihama Unit 1 data is currently being verified. The NRC staff needs clarification on how the licensee is addressing/will address quality assurance (QA) of the data.

Items to be addressed:

A. Standard for calibration of Charpy test machine, temperature, measurement of equipment

B. Method of determining chemical composition (% Cu, % Ni) of surveillance test specimens

C. Accuracy of measurements of items A. and B.

D. If Japanese standards are different than American standards, compare the standards

ADDITIONAL ITEM ON SURVEILLANCE DATA

Describe the surveillance program (number of capsules and withdrawal schedules) if available for Mihama Unit 1 with regard to weld wire heat 12008/27204. In addition, describe the Diablo Canyon Unit 1, Palisades (supplemental capsule) and Fort Calhoun (supplemental capsule) surveillance programs with regard to weld wire heat 27204/27204. Since Fort Calhoun proposes to use surveillance data from other units for the Fort Calhoun vessel integrity calculation, monitoring of the host units will become part of Fort Calhoun's 10 CFR 50, Appendix H reactor vessel surveillance program.

ATTACHMENT 4