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AUTH.NAME	AUTHOR AFFILIATION
PLUNKETT,T.F.	Florida Power & Light Co.
RECIP.NAME	RECIPIENT AFFILIATION
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SUBJECT: Provides response to NRC request for addl info re low upper shelf toughness fracture analysis of reactor vessels for load level A & B Conditions.

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JUL 13 1992

L-92-193
10 CFR 2.790
10 CFR 50, App. G

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

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
TAC Nos. M68249 and M55042
Low Upper-Shelf Toughness Fracture Analysis
of Reactor Vessels for Load Level A & B Conditions
Response to Request for Additional Information

This letter provides Florida Power and Light Company's (FPL) response to a Nuclear Regulatory Commission (NRC) Request for Additional Information. Several questions were asked by the NRC during a telephone call on June 10, 1992, concerning B&W report 2118-P, as submitted by FPL letter L-92-02, dated February 4, 1992. Attachment #1 states the NRC questions, with Attachment #2 providing FPL's responses.

Should there be any questions, please contact us.

Very truly yours,

TFPlunkett

T. F. Plunkett
Vice President
Turkey Point Nuclear

TFP/GS

attachments

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

A001 /1

ATTACHMENT #1

REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING

LOW UPPER-SHELF TOUGHNESS FRACTURE ANALYSIS OF REACTOR VESSELS OF TURKEY POINT UNITS 3 AND 4 FOR LOAD LEVEL A & B CONDITIONS

1. Provide Table 4-6, which is missing from the submittal.

2. Provide actual Ramberg-Osgood parameters used in this submittal. Clarify the nature of the data. (i.e. reference? irradiated fluence? relevance to Turkey Point #3 and #4 application?)

3. Explain the relationship between the irradiated HSST and the J-R model curves in Figures 4-22 and 4-23.

4. Clarify the statement on Page 2-5: "The irradiated upper-shelf energy data exhibits an inconsistency between the data for SA-1101 and SA-1094.

5. Explain the effect of changing the standard condition for data normalization on your final results.

6. Provide reference and methodology in support of the statement estimated 50 ft-lb. CvUSE value at the end-of-life.

ATTACHMENT #2

RESPONSES TO THE NRC RAI
REGARDING

LOW UPPER-SHELF TOUGHNESS
FRACTURE ANALYSIS OF REACTOR VESSELS
OF TURKEY POINT UNITS 3 AND 4
FOR LOAD LEVEL A & B CONDITIONS

1. Provide Table 4-6, which is missing from the submittal.

Table 4-6 is given below:

Table 4-6 Compact Test Specimens of SA-1101 (HSST Data)

Specimen Number	Weld ID	Specimen Size	Fluence (10^{18} n/cm^2)	Test Temp(F)
62W-25	SA-1101	4T	14	350
62W-135	SA-1101	0.5T	0	392
62W-136	SA-1101	0.5T	0	300
62W-141	SA-1101	0.5T	0	350

Revise the second sentence in section 4.5. on page 4-10 as follows:

"............... Specimens of SA-1101 ~~and~~ weld material ~~of the same weld wire heat but different lot~~ used for J-R model comparison are listed in Table 4-6.

Additions to the text are shown as follows: an addition
Deletions to the text are shown as follows: ~~a deletion~~

2. Provide actual Ramberg-Osgood parameters used in this submittal. Clarify the nature of the data. (i.e. reference? irradiated fluence? relevance to Turkey Point #3 and #4 application?)

On page 5-4 insert the following paragraph as shown below:

(Beginning of Page 5-4)

base, the 62W series specimens show a CvUSE of 66 ft-lb unirradiated. There is no CvUSE data at a fluence of 18×10^{18} n/cm^2. CvUSE value at the end-of-life was estimated to be approximately 50 ft-lb. It is difficult to assign the characteristics of a J-resistance curve to a single value of absorbed energy which went through a different mode of deformation process. Nevertheless, if one accepts the preceding assumptions, the comparison is as shown in Figure 5-4 where another curve at CvUSE of 44 ft-lb is shown. The J-R curve based on a Cv value of 50 ft-lb agrees very well with that of fluence of 1.8×10^{19} n/cm^2. The strengths of SA-1101 weld metal are available from the HSST program (BAW-1975) and listed in Table 5-2. The values selected for this analysis are from Specimen 62W-11, since this was tested at 550F and has a fluence of 1.5×10^{19} n/cm^2. This fluence is less than the fluence of 1.8×10^{19} used in this analysis, however, the yield strength is not so sensitive to the final J calculation. For example, an increase of 10 ksi in the yield strength would cause to decrease the total J by 0.5% in this application. The strength values used for this analysis are

σ_y = 75.4 ksi and σ_{ult} = 91.7 ksi

The corresponding Ramberg-Osgood parameters using equations 3-15 and 3-16

α = 1.63 and n = 10.4

5.4. J Analysis

For criterion 1 in Section 3.1, an applied J needs to be calculated at a flaw size equal ....

3. Explain the relationship between the irradiated HSST and the J-R model curves in Figures 4-22 and 4-23.

Attached clearer figures are to replace Figures 4-22 and 4-23. Since J_M is not used in this analysis, J_M-Δa curves are deleted. The text on pages 4-10 and 4-11 is modified as follows:

(Lower half of page 4-10)

4.5. Comparison Between J-R Model and SA-1101 Specimen Data

SA-1101 weld material is identified as the controlling welds in Turkey Point Units 3 and 4 reactor vessels. In the current J data bases, very few SA-1101 CT specimen test data are available. Specimens of SA-1101 ~~and~~ weld material ~~of the same weld wire heat but different lot~~ used for J-R model comparison are listed in Table 4-6.

In Figure 4-22, t~~T~~hree selected unirradiated SA-1101 J-R curves ~~and the 4T-CT irradiated specimen data J-R curves~~ are shown with a mean and a lower bounding J-R curves predicted by the model equation for unirradiated material (f = 0). It is obvious that SA-1101 is relatively tougher material in Mn-Mo-Ni/Linde 80 weld metal. ~~The predicted mean curve for unirradiated material is much lower than the irradiated material curve.~~ There is only one irradiated CT specimen of SA-1101 available from the HSST data set, i.e. 62W-25, a 4T CT specimen with fluence of 1.4×10^{19} n/cm^2, tested at 350 F. This material J-R curve was compared with the model predicted curves as shown in Figure 4-23. It is apparent that this model is very conservative for this application. There exists a factor greater than 3 on J values between the test data and the lower bounding model curve. ~~The same J-R data points are plotted against model curves with the same fluence as the 4T CT specimen in Figure 4-23.~~

Figure 4-22 J-R Model and Unirradiated HSST Data Comparison

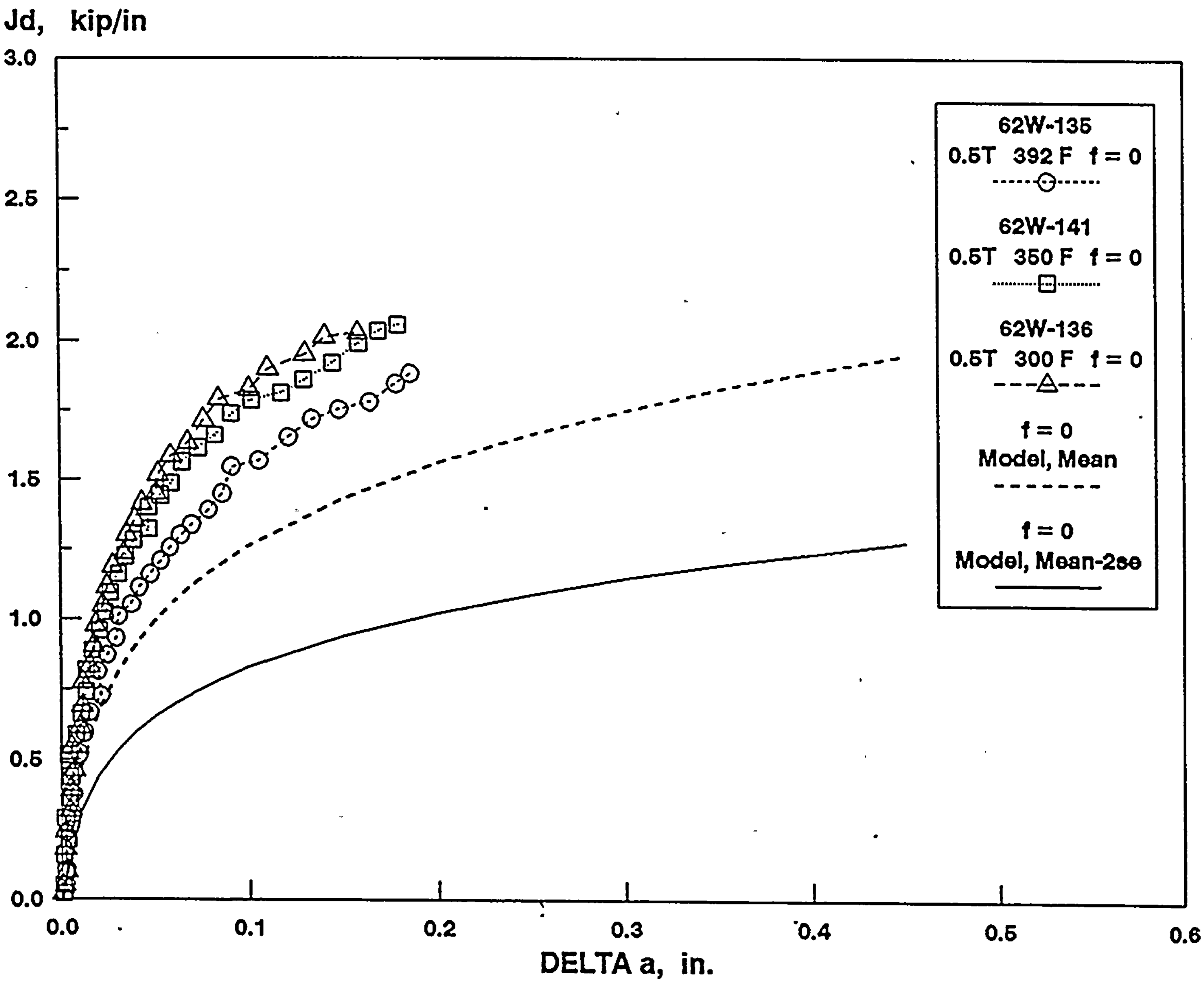
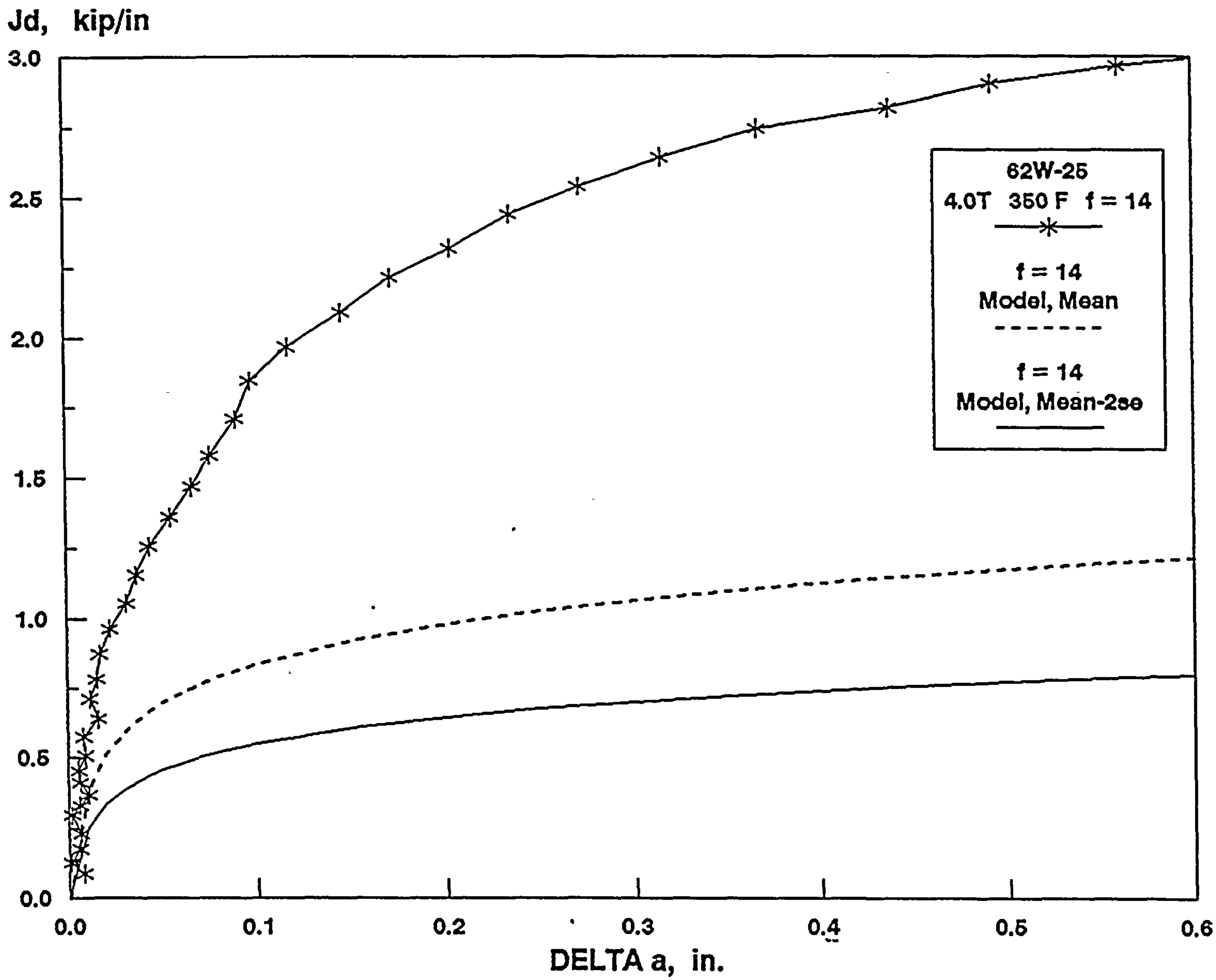


Figure 4-23 J-R Model and Irradiated HSST Data Comparison



4. Clarify the statement on Page 2-5: "The irradiated upper-shelf energy data exhibits an inconsistency between the data for SA-1101 and SA-1094.

To respond to this comment, the revised pages of 2-3, 2-4, 2-5, and 2-6 are attached.

Additions to the text are shown as follows: an addition
Deletions to the text are shown as follows: ~~a deletion~~

2.3.6. Overview of Surveillance Data

The available surveillance data from Turkey Point Units 3 and 4 are presented in Table 2-9. The summary presented are the ~~critical~~ Charpy impact energy and shift data ~~needed to evaluate~~ for the

chemistry for the weld metal. However, both of these weld metals were fabricated with the same weld wire (Mn-Mo-Ni Wire/Heat No. 71249) but different lots of Linde 80 weld flux and, therefore, it is expected that the two weld metals would behave in a similar manner. ~~Based on the evaluation of the individual weld metals it would indicate that a significant difference existed between the two weld metals.~~ In Figure 2-3 the RT_{NDT} shift data for SA-1094 appears to be significantly higher than those of SA-1101. An evaluation of the two weld metals based on Regulatory Guide 1.99, Position 2, and using the SA-1101 weld metal data for the calculation of the mean curve is shown in Figure 2-4. This demonstrates that the SA-1094 weld metal properties are within the two standard deviations of the surveillance data for the SA-1101 weld metal.

The upper-shelf energy data for SA-1094 and SA-1101 weld metals are compared to the correlation developed for Mn-Mo-Ni/Linde 80 submerged-arc weld metals in BAW-1803, Revision 1,[45] and are presented in Figure 2-5. These data also include two test reactor data points for SA-1101. All the data points for SA-1101 group around the 0.30% copper line which indicates a similar response of the weld metals to neutron radiation. It is significant that the data point for the SA-1094 weld metal is displaced about 0.10 percentage points from the similar data for SA-1101. Unlike~~,~~ the data for the Charpy shift data shown in Figure 2-3, the data in Figure 2-5 is not adjustable but represents the measured values. ~~Based on these observations it is possible that~~ The reasons for the differences ~~observed~~ in the data observed in Figure 2-5 may be ~~is~~ attributable to unidentified parameters which are not ~~readily obvious~~ clear from the

In conclusion, the Charpy 30 ft-lb shift temperature data is representative of ~~that which~~ what would be expected based on fluence and chemical composition. The irradiated upper-shelf energy data ~~exhibits an inconsistency between the data for SA-1101 and SA-1094. It appears that there is an inconsistency in the SA-1094 weld metal data that cannot be identified.~~ of SA-1094 seems to be significantly lower than the SA-1101 data. There appears to be no obvious reason for this behavior except that this data point may be an outlier. The variation is not important to the analysis because it is estimated that both materials will decrease below 50 ft-lbs prior to end-of-life. Since the controlling material in the actual reactor vessel is SA-1101 weld metal, the analysis of the reactor vessel will be based on SA-1101 weld metal properties.

2.4. Evaluation of Reactor Vessel End-of-Life Material Properties

The reactor vessels end-of-life fracture properties were evaluated to determine the need to implement the appropriate actions as defined in 10CFR50, Appendix G. The weld metals were identified as the most neutron radiation sensitive materials in the reactor vessels. The reactor vessel weld end-of-life (32 EFPY) fracture toughness properties for Turkey Point Units 3 and 4 are presented in Tables 2-10 and 2-11. These data show that the upper-shelf energy of weld metal SA-1101 (controlling weld metal for both reactor vessels) is estimated to decrease to a value less than 50 ft-lbs prior to the end of the reactor vessel design life. This decrease is estimated to occur only when the Regulatory Guide 1.99, Revision 2, formulation is used to make the estimate and not that included in BAW-1803, Revision 1. The data used to develop the correlation in BAW-1803, Revision 1, is based entirely on reactor vessel surveillance program data and thereby is the best correlation currently available.

Reactor vessel surveillance program capsule data from Turkey Point Unit 3 and the available test reactor data for weld metal SA-1101 are presented in Figure 2-5. These data indicate that it is reasonable to expect weld metal SA-1101 upper-

shelf energy to decrease to below 50 ft-lbs at or near to the end of design life of the reactor vessels.

The ~~data from~~ predicted Charpy impact energy levels by 1) Regulatory Guide 1.99, Revision 2, 2) the upper-shelf correlation ~~data~~ model from BAW-1803, Revision 1 and 3) the data obtained from the available surveillance capsules and test reactor research programs are summarized in Table 2-12. The ~~data~~ predicted Charpy impact energy values developed using the Regulatory Guide procedures demonstrate~~s~~ just how out-mo~~tive~~ded the technique is for estimating irradiated upper-shelf energies. By contrast, the ~~correlation~~ prediction based on the model ~~lower-bound~~ from BAW-1803, Revision 1, compares very favorably with the limited surveillance program data and the data points obtained from test reactor research programs. These data could be interpreted to justify that the upper-shelf energies will not decrease below the required 50 ft-lb. However, the uncertainty in the data precludes defining the relationship with absolute certainty either to upper-shelf energy level and/or reactor vessel life. Therefore, the prudent approach is to evaluate the reactor vessel as if the 50 ft-lb requirement was violated.

5. Explain the effect of changing the standard condition for data normalization on your final results.

This set of standard conditions is selected based on an approximate median value of each variable in the J-R data base. If one selects another set, it should produce similar results, unless the selections are based on remote or infrequent conditions. This normalization is intended for verification of the model. It shows how good the model is. Once the model is determined to be acceptable, then, the normalization has no direct impact on the final analysis.

6. Provide reference and methodology in support of the statement estimated 50 ft-lb. CvUSE value at the end-of-life.

The statement appeared on top part of page 5-4 is based on Table 2-12 and Figure 2-5 of this report where the Charpy energy prediction model from BAW-1803, Revision, yields 50 ft-lb when Cu% of 0.26 and fluence of 1.8 $x10^{19}$ n/cm^2 are used.

Table 4-2. Chemical Composition of Weld Metals in Data Base Used to Develop Correlation Models

Item	Weld ID	Chemical Composition, w%								
		C	Mn	P	S	Si	Cr	Ni	Mo	Cu
1	WF-193A	0.09	1.49	0.016	0.016	0.51	0.06	0.59	0.39	0.28
2	WF-182-1	0.09	1.69	0.014	0.013	0.41	0.15	0.63	0.40	0.21
3	SA-1263	0.09	1.47	0.019	0.024	0.49	0.13	0.57	0.39	0.22
4	SA-1036	0.08	1.41	0.012	0.016	0.59	0.09	0.56	0.36	0.23
5	SA-1101	0.08	1.56	0.019	0.008	0.59	0.16	0.54	0.38	0.21
6	SA-1094	0.10	1.44	0.014	0.011	0.50	0.14	0.60	0.36	0.30
7	SA-1526	0.09	1.53	0.013	0.017	0.53	0.08	0.68	0.42	0.35
8	WF-233	0.10	1.45	0.021	0.015	0.42	0.08	0.68	0.44	0.27
9	WF-25	0.09	1.58	0.015	0.016	0.54	0.09	0.67	0.42	0.35
10	WF-67	0.08	1.55	0.021	0.016	0.58	0.10	0.60	0.40	0.22
11	SA-1585	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21
12	WF-70	0.09	1.63	0.018	0.009	0.54	0.11	0.59	0.40	0.42
13	WF-112	0.08	1.47	0.016	0.015	0.54	0.07	0.59	0.40	0.32
14	SA-1135	0.08	1.45	0.011	0.013	0.49	0.08	0.59	0.38	0.27
15	WF-209-1	0.11	1.55	0.022	0.010	0.65	0.09	0.58	0.39	0.36
16	WF-292	0.13	1.47	0.009	0.011	0.61	0.09	0.62	0.45	0.03
17	SA-1118	0.08	1.29	0.013	0.014	0.61	0.09	0.57	0.39	0.32