TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 19, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

By my letter to you dated September 22, 1983 we submitted a request to change the technical specifications of the Browns Ferry Nuclear Plant (TVA BFNP TS 191). The requested specifications enclosed with the letter revise the pressure-temperature limit curves for hydrostatic pressure test, reactor vessel heatup and cool down, and nuclear core operation. The enclosure to this letter provides additional information in support of TS 191 requested by your staff.

The enclosure includes information regarding chemical composition of reactor vessel material, end-of-life neutron fluence, and Charpy impact testing results. Please get in touch with us through the Browns Ferry Project Manager if additional information is needed.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Licensing

Subscribed and sworn to before
me this 19th day of October 1983.

Paulette H. White
Notary Public
My Commission Expires 9-5-84

Enclosure

cc: See page 2

Limited Dist.

A001
1/3

cc (Enclosure):
Mr. Charles R. Christopher
Chairman, Limestone County Commission
P.O. Box 188
Athens, Alabama 35611

Dr. Ira L. Myers
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. R. J. Clark
U.S. Nuclear Regulatory Commission
Browns Ferry Project Manager
7920 Norfolk Avenue
Bethesda, Maryland, 20814

ENCLOSURE

ADDITIONAL INFORMATION REGARDING REACTOR
VESSEL MATERIAL TOUGHNESS AND NEUTRON FLUENCE
IN SUPPORT OF AMENDMENT REQUEST TVA BFNP TS 191
BROWNS FERRY NUCLEAR PLANT
(DOCKET NOS. 50-259, -260, -296)

Attachment 1 - Reactor Vessel Beltline Material Toughness Data

Attachment 2 - GE letter dated February 9, 1971 - Information Regarding Neutron Fluence

Attachment 3 - Browns Ferry unit 1 - Analysis of Vessel Wall Neutron Dosimeter - (report by Southwest Research Institute)

Attachment 4 - Browns Ferry units 2 and 3 - Analysis of Vessel Wall Neutron Dosimeter (two reports by Southwest Research Institute)

Attachment 5 - Fabrication and Testing of Reactor Vessel Material (report by TVA's Singleton Materials Engineering Laboratory)

ATTACHMENT 1

Table 1

Reactor Vessel Beltline Material Toughness Data
Browns Ferry Nuclear Plant Units 1, 2, and 3

Component	Unit	Heat No.	Material Grade	Cu (%)[1]	NDTT (°F)	Minimum 50 ft lb/35 mil temp(°F) PMWD[4,6]	Minimum 50 ft lb/35 mil temp(°F) NMWD[5,6]	Initial RT_{NDT}[7] (°F)	Shift in RT_{NDT} for 40 EFPY 1/4T[8] (°F)	Final RT_{NDT} for 40 EFPY at 1/4T (°F)
Base Metal	1	C-2884	SA 302 GRB	0.12	0[2]	+34	+54[3]	0	32	32
HAZ	1	-	Weld	-	0[3]	+60	+80[3]	+20	32	52
WELD	1	-	Weld	0.10	+13[3]	-	+38	+13	26	39
Base Metal	2	A-0981	SA 302 GRB	0.14	-10[2]	-14	+14[3]	-10	39	29
HAZ	2	-	Weld	-	0[3]	-40	-28[3]	0	39	39
WELD	2	-	Weld	0.20	0[3]	-	+18	0	58	58
Base Metal	3	B-7267	SA 302 GRB	0.10	-20[2]	-36	-18[3]	-20	26	6
HAZ	3	-	Weld	-	0[3]	+20	+47[3]	0	26	26
WELD	3	-	Weld	0.11	0[3]	-	-22	0	29	29

[1]See letter from T. F. Henry, Combustion Engineering, Inc., to John Fox dated April 6, 1983

[2]General Electric Company letter from R. B. Beers to H. M. Bankus, "Reactor Vessel Surveillance Program", dated February 9, 1971 (copy attached).

[3]Estimated based on Branch Technical Position MTEB 5-2 (copy attached) and minimum valves from Charpy impact data provided by Singleton Laboratory (see attached memorandum from F. Van Meter to T. F. Ziegler dated August 11, 1983--L21 830812 112)

[4]Parallel to major working direction

[5]Normal to major working direction

[6]Determined by using minimum values from Charpy impact data provided by Singleton Laboratory.

[7]The greater of NDTT or C_V (50 ft lb/35 mils) - 60 for NMWD

[8]Estimated using [1] above, Regulatory Guide 1.99 Curves (copy attached) and neutron fluence determined from flux wire testing (see C16 790223 019 and L26 800318 883--copy attached).

POWER SYSTEMS

April 6, 1983

Mr. John Fox
Tennessee Valley Authority

Subject: Materials Analyses
Project No. 900051
Job No. 98120216
MML-83-72

Sample:	Unit 1, 2868 (P23079)	Base Unit 1, 2884 (P23080)	Base Unit 2, E6B (P23081)	Base Unit 3, 65E (P23082)	Weld Unit 2, EBM (D39368)	Weld Unit 3, 67J (D39369)	Weld Unit 1, Weld Plate (D39370)
C	.22	.27	.20	.18	.15	.16	.17
Mn	1.39	1.33	1.35	1.27	1.49	1.50	1.45
P	.007	.008	.007	.007	.010	.011	.012
S	.009	.015	.011	.007	.011	.012	.013
Si	.23	.21	.19	.22	.09	.10	.09
Ni	.52	.52	.55	.51	.33	.28	.30
Cr	.07	.09	.11	.08	.08	.08	.06
Mo	.48	.46	.49	.47	.49	.49	.48
V	.002	.003	.002	.002	.003	.003	.003
Co	.009	.010	.012	.010	.011	.010	.010
Cu	.09	.12	.14	.10 -	.20	.11	.10
Al	.032	.035	.025	.015	.007	.006	.010
B	<.001	<.001	<.001	<.001	<.001	<.001	<.001
W	<.01	<.01	<.01	<.01	<.01	<.01	<.01
As	.006	.009	.015	.010	.013	.008	.009
Sn	.006	.008	.013	.008	.009	.005	.006
Zr	<.001	<.001	<.001	<.001	<.001	<.001	<.001
N_2	.006	.007	.007	.005	.007	.006	.006

We certify that the information recorded above is an accurate copy of the data obtained from tests performed at the Metallurgical and Materials Laboratory.

COMBUSTION ENGINEERING, INC.

J. F. Henry

DATE 4-7-83

JFH/cel

Sworn and subscribed to before me this 7th day of April 1983

Constance D. Finch
Notary Public

My Commission Expires 3/6/84

ATTACHMENT 2

GENERAL ELECTRIC

NUCLEAR ENERGY DIVISION

ATOMIC POWER EQUIPMENT DEPARTMENT

FEB 11 1971

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIF. 95125
Phone (408) 297 3000, TWX NO. 910-338-0116

RECEIVED FEB 12 1971
DESIGN-MECHANICAL

Mr. R.H. Dunham
Chief Mechanical Engineer
Tennessee Valley Authority
200 Union Building
Knoxville, Tennessee

Letter No. BF 3693
February 11, 1971

cc: R. S. Craven, TVA, Chatt.

TVA Letter No. 2553
February 9, 1971

cc: M. O. Troy
R. R. Barris

Mr. H. M. Bankus
General Electric Company
1301 Hannah Avenue N. W.
Knoxville, Tennessee 37921

Subject: Reactor Vessel Surveillance Program
Reference: TVA to GE Letter 6599 and 3-3336

We have attached a marked-up copy of TVA Letter 6599 with the questions numbered. Our answers refer to the above numbers:

1) According to our QC records, the following is the data on the MK #58
2) shell plate material:
6)

	Heat No.	Plate No.	NDTT (°F)
TVA I	C-2884	6-139-19	0
	C-2868	6-139-20	0
	C-2753	6-139-21	-20
TVA II	A-0981	6-127-6	-10
	C-2467	6-127-16	-10
	C-2849	6-127-20	-10
TVA III	C-3201	6-145-1	-30
	C-3188	6-145-2	-30
	B-7267	6-145-6	-20

3) B&W cites the second sentence of Par. 3.3.1 ("The Buyer can furnish a plan which the Seller may use as a guide.") as permission to use Appendix B and Drawings 117B1549 and 117B1500 in lieu of preparing the required document. B&W stated that they followed these documents in preparing the surveillance specimen. B&W states that the surveillance samples will be taken from either MK #57 or MK #58 plate material.

4) Further correspondence with B&W has failed to obtain any further docu-
5) mentation or information beyond that described above. We are continuing to pursue this matter.

7) No special handling tools are provided for removal of the surveillance
8) specimen holders from the reactor vessel. The normal pool tool accessories (J-hook, L-hook, actuating pole, etc.), plus the in-core detector cutter are sufficient to remove, handle and prepare for shipment. Shipment is necessary because testing of the individual specimen requires the services of a radiation-laboratory. It is left up to each customer to select the laboratory and contract for the particular services they desire. Shipment is most easily accomplished by separating the specimen basket (intact with capsules containing the individual specimen) from the surveillance holder pole. This is done with use of the in-core detector cutter. The basket is deposited in a small lead shipping cask and sent to the laboratory for processing. The remaining pole and handle assembly may be cut up and discarded as rad-waste.

9) GE would be willing to provide testing and evaluation service on irradiated
10) surveillance samples. Actual contract conditions will have to be negotiated through our Chattanooga office.

11) The maximum calculated integrated neutron dose (1 mev) on the vessel wall
12) is 3.80×10^{17} NVT. This exposure is expected to occur at an elevation
13) three feet above the bottom of the active core at azimuthal positions
14) opposite the corner elements. The azimuthal positions are approximately 25°, 65°, 115°, 155°, 205°, 245°, 295° and 335°.

The calculated dose on the surveillance baskets at azimuthal locations of 30°, 120°, and 200° are equal and equivalent to

$$3.07 \times 10^{8} \frac{\text{neutrons}}{\text{cm}^2 - \text{sec}} \text{ or } 3.9 \times 10^{17} \text{ NVT}$$

at rated power for 40 years. These values are based on the same calculation type as the maximum fluence base. The axial variation is assumed to be proportional to Figure 1.

The above data applies to all three units.

15) As stated in NEDO-10115, Para. 5.7: "Because the boiling water reactor is a constant-temperature device, no special temperature monitoring devices are required." This has been confirmed by temperature monitoring devices which had been installed in earlier plants.

16) Two copies of NEDO-10115 are enclosed.

This completes our response to TVA Letter 6599 and 3-3336.

R. B. Beers
Project Manager
TVA Project

JRO:kn
Enclosure

CUSTOMER: General Electric CUSTOMER ORDER NO: 205-55577 DATE: 5-25-70

JOB NO: 610-0127-51 Unit #1 TYPE OF MATERIAL: Mn-Mo-Ni SPECIFICATION: Code Case 1335 SA 302, Gr B

WK NO: 58A COMPONENT SERIAL NUMBER: 6-139-19 THICKNESS: 6 1/8" Min.

CHEMICAL ANALYSIS HEAT NO.	C	Mn	P	S	Si	Cr	Ni	Mo	Cu / Nb	Co / Ta	Al / V	Sn	Fe
Lukens C2884-2	.22	1.33	.015	.016	.23		.53	.48					

SPECIMEN CODE NO.	HEAT NO.	DROP WEIGHT - ASTM E208 TEST TEMPERATURE +10°F.	0°F.	-10°F.	NDT TEMP.	CHARPY - V NOTCH IMPACT TEST TEMPERATURE DEG. F	FT. - LB.	LATERAL EXPANSION	% SHEAR
6-139-19T	C2884-2	NF,NF	F	F	0°F.	+10	33, 55, 34	.020, .038, .026	15, 20, 15
6-139-19B		NF,NF				+10	50, 44, 50	.036, .028, .027	35, 20, 25
		Height - 3' Weight - 100		Lbs.			240 Ft.-Lb. Energy Load		

SPECIMEN CODE	HEAT NO.	TENSILE STRENGTH P.S.I.	YIELD POINT P.S.I.	% ELONGATION IN 2 INCHES	REDUCTION OF AREA %	180° BEND TEST	[illegible]	GRAIN SIZE
6-139-19T	C2884-2	88,250	69,500	26.6	68.2	Good	Homogeneity	6 and
6-139-19B		92,250	69,750	24.2	69.3		OK	Finer

HEAT TREATMENT: 1675-1725°F. for 6½ Hrs./Brine Quench
1600-1650°F. for 6½ Hrs./Brine Quench
1175-1225°F. for 6½ Hrs./Brine Quench

I HEREBY CERTIFY THAT THE [illegible]

CERTIFICATE OF TEST

CUSTOMER: General Electric | CUSTOMER ORDER NO: 205-55577 | DATE: 5-25-70

LOT NO: 610-0127-51 Unit #1 | TYPE OF MATERIAL: Mn-Mo-Ni | SPECIFICATION: Code Case 1335-4 SA 302, GR. B

HEAT NO: 58A | COMPONENT SERIAL NUMBER: 6-139-20 | THICKNESS: 6 1/8" Min.

CHEMICAL ANALYSIS	HEAT NO.	C	Mn	P	S	Si	Cr	Ni	Mo	Cu / Nb	Co / Ta	Al / V	Sn	Fe
Lukens	C2868-2	.21	1.38	.013	.015	.22		.48	.45					

SPECIMEN CODE NO.	HEAT NO.	DROP WEIGHT - ASTM E208 TEST TEMPERATURE -10°F.	0°F.	+10°F.	NDT TEMP.	CHARPY - V NOTCH IMPACT TEST TEMPERATURE DEG. F	FT. - LB.	LATERAL EXPANSION	% SHEAR
6-139-20T	C2868-2	F	F	NF,NF	0°F.	+10	46, 55, 25	.035, .040, .020	35, 15, 35
6-139-20B				NF,NF		+10	59, 56, 56	.042, .041, .051	15, 10, 20
		Height - 3' Weight - 100		Lbs.			240 Ft.-Lb. Energy Load		

SPECIMEN CODE	HEAT NO.	TENSILE STRENGTH P.S.I.	YIELD POINT P.S.I.	% ELONGATION IN 2 INCHES	REDUCTION OF AREA %	180° BEND TEST	MACRO	GRAIN SIZE
6-139-20T	C2868-2	86,500	68,000	27.3	69.9	Good	Homogeneity OK	6 and Finer
6-139-20B		86,500	64,750	26.6	70.4			

HEAT TREATMENT: 1675-1725°F. for 6½ Hrs./Brine Quench
1600-1650°F. for 6½ Hrs./Brine Quench

I HEREBY CERTIFY THAT THE ABOVE RES... ARE CORRECT & SPECIFIED IN THE ORDER

ATTACHMENT 3

RWC

C16 790223 919

J. G. Dewease, Power Plant Superintendent, Browns Ferry Nuclear Plant

H. J. Green, Chief, Nuclear Generation Branch, 727 EB-C

FEB 28 1979

SOUTHWEST RESEARCH INSTITUTE - FINAL REPORT ON ANALYSIS OF THE VESSEL WALL NEUTRON DOSIMETER FROM BROWNS FERRY UNIT 1 PRESSURE VESSEL

Attached are four copies of the completed report on the analysis of the Browns Ferry unit 1 vessel wall dosimeter. This analysis was performed to comply with the requirements of Browns Ferry Surveillance Instruction (SI) 4.6.A.4. The results of the analysis do not meet the acceptance criteria of the above-mentioned SI which states that the experimentally-determined neutron fluence shall be less than the maximum calculated neutron fluence. The experimentally-determined neutron fluence for the end-of-core cycle 1 at the inside diameter of the vessel was 5.64×10^{17} cm^{-2}, while the calculated value of neutron fluence was 1.37×10^{17} cm^{-2}. We are therefore required to perform an analysis to determine the effect of increased fluence on vessel-wall embrittlement and RT_{NDT}.

Regulatory Guide 1.99 stipulates that the fluence value at the 1/4T (thickness) vessel wall location be used as the basis for determining the shift in RT_{NDT}. Figure 1 of the same document presents a means to determine the shift in RT_{NDT} based on known fluence levels and material copper content. In a letter from A. L. Vest, of General Electric Company, to me dated December 19, 1977, the curves of Figure 1 were extrapolated downward to allow determination of transition temperature shifts at the lower generating flux values for boiling water reactors.

A letter dated August 23, 1977, from J. E. Gilleland to A. Schwencer, of NRC, established an upper limit for plate and weld metal copper content of 0.30 weight percent. Utilizing this copper content, the projected 1/4T fluence value of 2.62×10^{16} cm^{-2} per EFPY (derived from the product of $0.67 \times 3.91 \times 10^{16}$ cm^{-2}/EFPY taken from the SwRI analysis), and the extrapolated Figure 1 of Regulatory Guide 1.99, the following chart is generated.

J. G. Dewease

SOUTHWEST RESEARCH INSTITUTE - FINAL REPORT ON ANALYSIS OF THE VESSEL WALL NEUTRON DOSIMETER FROM BROWNS FERRY UNIT 1 PRESSURE VESSEL

EFPY	1/4T Fluence (cm^{-2})	Shift In RT_{NDT} (° F)
1	2.62×10^{16}	0
2	5.24×10^{16}	20
3	7.86×10^{16}	25
4	1.05×10^{17}	29
5	1.31×10^{17}	33
6	1.57×10^{17}	36
7	1.83×10^{17}	38
8	2.10×10^{17}	41
9	2.34×10^{17}	43
10	2.62×10^{17}	45
11	2.88×10^{17}	48
12	3.14×10^{17}	50
13	3.41×10^{17}	52
14	3.67×10^{17}	54
15	3.93×10^{17}	56
16	4.19×10^{17}	58
17	4.46×10^{17}	60
18	4.72×10^{17}	61
19	4.98×10^{17}	63
20	5.24×10^{17}	65
21	5.50×10^{17}	66
22	5.76×10^{17}	68
23	6.03×10^{17}	69
24	6.29×10^{17}	71
25	6.55×10^{17}	73
26	6.81×10^{17}	74
27	7.07×10^{17}	75
28	7.34×10^{17}	76
29	7.60×10^{17}	78
30	7.86×10^{17}	79

J. G. Dewease

SOUTHWEST RESEARCH INSTITUTE - FINAL REPORT ON ANALYSIS OF THE VESSEL WALL NEUTRON DOSIMETER FROM BROWNS FERRY UNIT 1 PRESSURE VESSEL

EFPY	1/4T Fluence (cm^{-2})	Shift in RT_{NDT} (° F)
31	8.12×10^{17}	80
32	8.38×10^{17}	81
33	8.65×10^{17}	82
34	8.91×10^{17}	83
35	9.17×10^{17}	85
36	9.43×10^{17}	86
37	9.69×10^{17}	87
38	9.96×10^{17}	88
39	1.02×10^{18}	89
40	1.05×10^{18}	90

The operator's curves for unit 1 will not be affected by the above-referenced beltline curve shift in RT_{NDT} until the eighth effective full power year of operation since the nonbeltline region curve is dominant until the beltline curve shifts by 40° F.

The predicted end-of-core life adjusted reference temperature at the 1/4T position determines the specimen capsule removal schedule in accordance with 10 CFR 50, Appendix H. Page 22 of the attached report contains a table which depicts two different specimen capsule removal schedules based on the initial RT_{NDT} of the beltline region plates. The RT_{NDT} for the unit 1 reactor vessel beltline region is obtained as follows.

A letter dated February 9, 1971, from R. B. Sears, of GE, to H. M. Bankus, of GE, specifies that the highest NDTT value for unit 1 reactor vessel beltline region plates is 0° F. This value is derived from drop-weight tests performed by the vessel manufacturer. Copies of the test certificates for the limiting heats (C2868 and C2884) show the drop-weight test, as well as the single-temperature Charpy V-notch impact

J. G. Deweese

SOUTHWEST RESEARCH INSTITUTE - FINAL REPORT ON ANALYSIS OF THE VESSEL WALL NEUTRON DOSIMETER FROM BROWNS FERRY UNIT 1 PRESSURE VESSEL

test results (longitudinally-oriented specimens). Section B.1.1.(4) of Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants," specifies how to determine the RT_{NDT} corresponding to the plate material if limited Charpy V-notch tests were performed at a single temperature on specimens oriented in the longitudinal direction. It states that if the minimum absorbed energy is less than 45 ft/lbs., the RT_{NDT} is estimated as 20° F above the test temperature. Therefore, RT_{NDT} = 10° F (test temperature) + 20° F = 30° F.

Since the above value of RT_{NDT} is greater than 12° F, the end-of-core life adjusted reference temperature at the 1/4T position will exceed 100° F. This dictates a specimen capsule withdrawal schedule conforming to Section II.C.3.b, 10 CFR 50, Appendix H. Since the 50° F shift in RT_{NDT} occurs at approximately the 12 EFPY point, which exceeds the one-fourth service life value of 10 EFPY, the first specimen capsule removal should occur at the 10 EFPY point. (Note that the table on page 22 of the attached report is incorrect for this estimate.) Note however that the results of the specimen tests are not utilized to predict the shift in RT_{NDT}, but to verify, if needed, Figure 3.6-2 of the Browns Ferry unit 1 technical specifications.

Copies of applicable sections of all referenced documents are attached.

H. J. Green

BDD:CAM:RWC:DB
Attachments
cc: ARMS PP, G23 EB-C