Duke Power Company
P.O. Box 1006
Charlotte, N.C. 28201-1006

HAL B. TUCKER
Senior Vice President
Nuclear Generation
(704) 382-4531

DUKE POWER

July 3, 1992

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

Subject: McGuire Nuclear Station Units 1 & 2
Docket Nos. 50-369,370
Catawba Nuclear Station Units 1 & 2
Docket Nos. 50-413,414
Oconee Nuclear Station Units 1, 2, & 3
Docket Nos. 50-269, 270, 287
Response to Generic Letter 92-01, Revision 1
Reactor Vessel Structural Integrity

Gentlemen:

On March 6, 1992, the NRC issued Generic Letter 92-01, Revision 1 to supercede the original version dated February 28, 1992. The subject generic letter was issued to obtain information from the licensees which will enable the NRC to assess the degree of compliance with regulatory requirements regarding reactor vessel integrity.

Duke Power has been in compliance with the regulatory program to maintain the structural integrity of the reactor vessels. Accordingly, Attachments A, B, and C to this letter provide the required information for Catawba, McGuire and Oconee Nuclear Stations, respectively. The attachment for Oconee Nuclear Station references a cover letter dated June 17, 1992 which was submitted to the NRC by the B & W Owners Group to address Oconee Units 1, 2, and 3 response to GL 92-01, Revision 1.

I declare under penalty of perjury that these statements are true and correct to the best of my knowledge.

Should you have any questions or require any additional information regarding this submittal, please contact Allison Jones at (704) 373-2026.

Add: D. McDONALD
K. WICKMAN
B. ELLIOT

A028
1/1

Very truly yours,

H. B. Tucker
Senior Vice President
Nuclear Generation

adj/gl9201

Attachments

cc: S. D. Ebneter
Regional Administrator, Region II

T. A. Reed, ONRR

R. E. Martin, ONRR

L. A. Wiens, ONRR

P. K. Vandoorn
Senior Resident Inspector (MNS)

W. T. Orders
Senior Resident Inspector (CNS)

P. E. Harmon
Senior Resident Inspector (ONS)

Kurt Cozens, NUMARC

bcc: A. V. Carr
W. M. Sample
R. C. Futrell
R. L. Gill, Jr.
R. O. Sharpe
M. E. Patrick
H. L. Atkins
D. R. Bradshaw
P. R. Herran
J. S. Forbes
B. L. Peele
G. D. Gilbert
T. P. Harrall
J. M. Davis
G. T. Smith
J. J. Nolin
D. E. Colson
J. D. Gilreath
L. J. Kunka
T. S. Barr - TTC
File: GC-801.01

ATTACHMENT A
CATAWBA NUCLEAR STATION'S RESPONSE TO GENERIC LETTER 92-01

The following is Catawba Unit's 1 & 2 response to Generic Letter 92-01.

__Question 1.__

Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

The Catawba Unit 1 reactor vessel surveillance program meets the requirements in ASTM E185-73 as described in WCAP-9734, referenced in paragraph 5.3.1.6 of the FSAR.

The Catawba Unit 2 reactor vessel surveillance program meets the requirements of ASTM E185-82, as described in WCAP-10868, referenced in paragraph 5.3.1.6 of the FSAR.

__Question 2.__

Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

a. **Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.**

   The Catawba Unit 1 & 2 reactor vessel beltline materials are predicted to maintain greater than 50 ft-lbs upper shelf energy through the end of license.

   Using the method described in Reg. Guide 1.99 rev. 2 paragraph C.1.2 each beltline material was examined for Catawba Unit's 1 & 2. All beltline materials had greater than 50 ft-lbs upper shelf at end of license. The chemistry and fluence documented in the most recent surveillance capsule reports (WCAP-10786 and WCAP-11941) were used for this verification.

b. **Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:**

The Catawba Unit 1 reactor vessel was built to the ASME B&PV code, Section III, 1971 Edition including addenda through the winter 1971 as documented in the reactor vessel QA data package and reported in paragraph 5.3.1.5 of the FSAR.

The Catawba Unit 2 reactor vessel was built to the ASME B&PV code, Section III, 1971 Edition including addenda through the winter 1972 as documented in the reactor vessel QA data package and reported in paragraph 5.3.1.5 of the FSAR.

Attached are all materials data necessary to answer questions 2b(1) - (6) for Unit 1.

(1) **the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;**

See attachment 2 pages 4-6 for Lower Shell forging material.
See attachment 2 pages 1-3 and attachment 3, section 3 of WCAP-9734 for Intermediate Shell forging.
See attachment 2 pages 7-9 and attachment 3, section 3 of WCAP-9734 for weld material.

(2) **the heat treatment received by all beltline and surveillance materials;**

See attachment 2.

(3) **the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;**

See attachment 2.

(4) **the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;**

See attachment 2.

(5) **the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material: and**

See attachment 2.

(6) **the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.**

Same as item 3.

<u>Question 3.</u>

Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

a. **How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 xF were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.**

Operating T-cold temperatures below 525F are not applicable to the Catawba Units. As shown in Catawba's FSAR figure 4-72 (see attachment 1) T-cold remains fairly constant at approximately 557F with T-avg ranging from 557F to 590.8F. Technical Specification 3.1.1.4 requires a minimum T-avg temperature for criticality of 551F. An LER search has been performed to determined if this tech spec has ever been violated. There were no violations discovered.

b. **How their surveillance results on the predicted amount of embrittlement were considered.**

With regard to the GL 88-11 response, dated November 28, 1988, neither Catawba Unit 1 nor Unit 2 had two sets of surveillance data available and therefore surveillance data was not used to determine the adjusted RTndt.

c. **If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.**

The measured increase in reference temperature for Catawba Units 1 & 2 has not exceeded the mean-plus-two standard deviations predicted by Reg. Guide 1.99 rev. 2. The measured decreases in charpy upper shelf energy has not exceeded the predicted percent USE decrease (See WCAP-10786 for Unit 1 and WCAP-11941 for Unit 2).

References:

Catawba Reactor Vessel DCP Data Package
containing: Rotterdam Material Test Reports

WCAP-9734 — Duke Power Company
Catawba Unit No. 1 Reactor Vessel
Radiation Surveillance Program
Date: June 10, 1980

WCAP-10786 — Analysis of Capsule Z From the Duke Power Company
Catawba Unit 1 Reactor Vessel Radiation Surveillance Program
Date: June 1987, Issued to the NRC by letter dated September 1, 1987.

Catawba's response to Generic Letter 88-11 dated November 28, 1988, Docket Nos. 50-413 and 50-414.

Catawba Reactor Vessel DDP Data Package
containing: CE Material Test Reports

WCAP-10868 — Duke Power Company
Catawba Unit No. 2 Reactor Vessel
Radiation Surveillance Program
Date: November 1985

WCAP-11941 — Analysis of Capsule Z From the Duke Power Company
Catawba Unit 2 Reactor Vessel Radiation Surveillance Program
Date: September 1988, Issued to the NRC by letter dated April 19, 1989.

ATTACHMENT 1

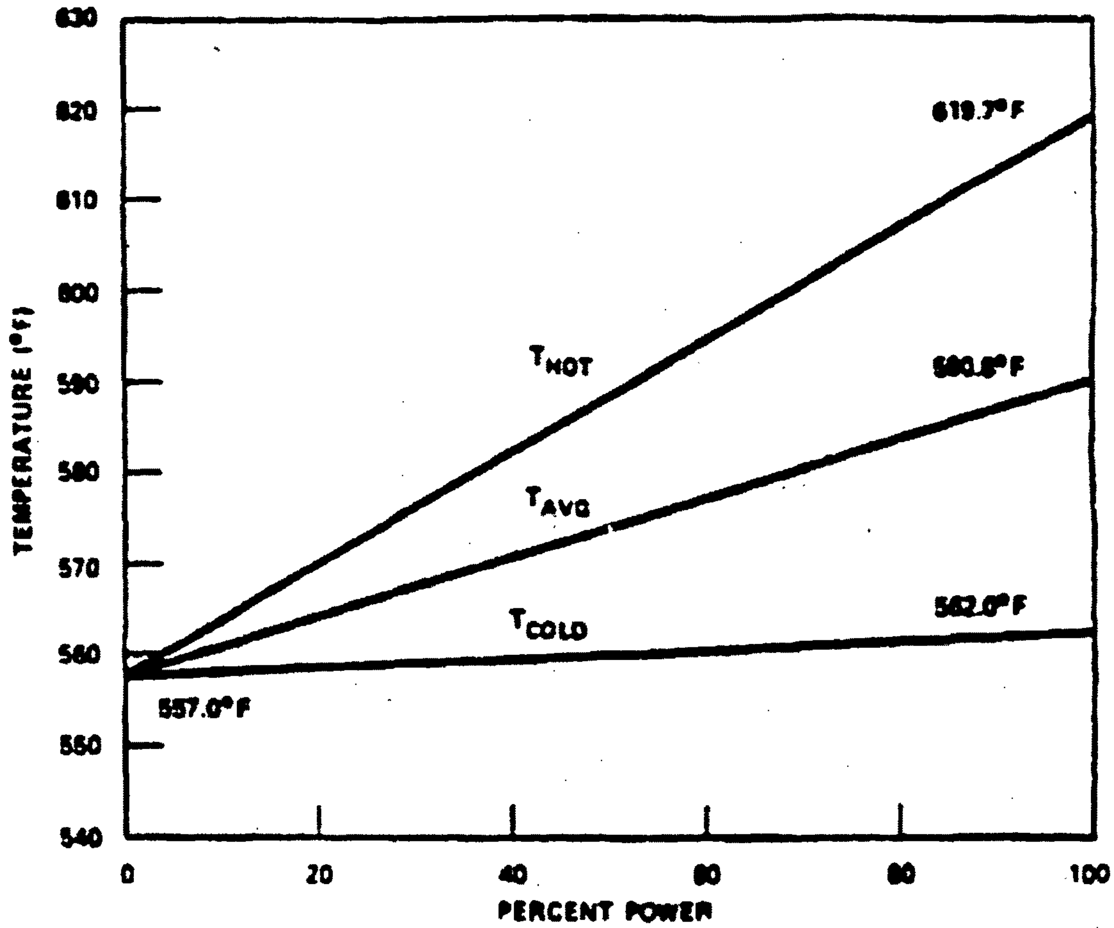

Figure 4-72
Reactor Coolant System Temperature - Percent Power Map

Attachment 2

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Piece Description: Intermediate Shell Forging

Supplier: Klockner-Werge AG

RDY Item Number	Heat Number(s)
Ring 05	411343

Composition:

Heat No.	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Co	
411343	0.21	0.78	0.004	0.006	0.28	0.89	0.38	0.6	0.08	0.01	Note 1
	0.02	0.73	0.014	0.004	0.35	0.84	0.38	0.53	0.1	0.02	Note 2
	0.207	0.72	0.002	0.014	0.234	0.85	0.418	0.63	0.08	0.01	Note 3

Note 1) Ref: RDY Material Certification, Catawba QA Vault DCP pkg.
Note 2) Ref: WCAP-9734, Table 4-1; (Original Analysis)
Note 3) Ref: WCAP-11527, Table 4-1; (Capsule Z, Specimen ML-87)

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
11Z-0307	940 C	860 C	2 Hr for surf.	WC	Austenizing - Quenching
11Z-0308	700 C	650 C	5.5 Hr for Surf.	FC - 450C/AC	Tempering
PWHT	620C	590C	21 Hr 14 Min	FC	Stress Relief
Ref: RDY Material Certification, Catawba QA Vault DCP pkg.					

Nil Ductility Temperature, Tndt

Tndt :	-40 F
Ref: WCAP - 9734	

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F):	-8 F
Ref:	Catawba Tech. Spec.

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils CVN tests).

Attachment 2

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Charpy V-notch Impact Test Results:
(Also see attached WCAP-9734, section 3 - Preirradiation Testing)

Charpy V-notch Impact Test Results for Heat No. 411333:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Lateral Expansion (mils)
4305K11	40	102	79
4305K12	40	103	79
4305K13	40	100	71
4305K21	40	87	71
4305K22	40	87	71
4305K23	40	58	51
4305K31	-130	5	8
4305K32	-112	8	8
4305K33	-94	6	8
4305K34	-76	8	8
4305K41	-58	14	12
4305K42	-40	14	18
4305K43	-22	58	51
4305K44	-4	18	20
4305K51	14	72	59
4305K52	32	81	67
4305K53	50	119	79
4305K54	68	116	87
4305K61	86	128	79
4305K62	104	138	91
4305K63	122	155	94
4305K64	140	150	94
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
Orientation: tangential direction with axis of notch axial			

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Additional Charpy V-notch Impact Test Results for Heat No. 411343:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
4305K1R	-4	43	59
4305K2R	-4	58	28
4305K3R	-4	54	39
4305K4R	-76	10	4
4305K5R	-76	17	16
4305K6R	-76	8	4
4305K7R	-148	5	4
4305K8R	-148	5	4
4305K9R	-148	5	4
4305K10R	60	86	67
4305K11R	60	83	63
4305K12R	60	81	63
4305K13R	176	138	99
4305K14R	176	137	91
4305K15R	176	136	91
4305K16R	104	115	83
4305K17R	104	118	75
4305K18R	104	112	83
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
Orientation: radial direction with axis of notch axial			

Additional Charpy V-notch Impact Test Results for Heat No. 411343:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
4305K71	-4	46	27
4305K72	-4	17	16
4305K73	-4	42	17
4305K74	-76	9	3
4305K75	-76	4	0
4305K76	-76	9	0
4305K77	-148	2	0
4305K78	-148	3	0
4305K79	-148	5	0
4305K80	60	118	75
4305K81	60	123	79
4305K82	60	92	63
4305K83	113	140	87
4305K84	113	130	91
4305K85	113	155	99
4305K86	176	153	87
4305K87	176	157	87
4305K88	176	153	94
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
Orientation: tangential direction with axis of notch axial			

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Attachment 2

Piece Description: Lower Shell Forging

Supplier: Klockner-Werge AG

RDY Item Number	Heat Number(s)
Ring 04	527708

Composition:

Heat No.	C	Mn	P	S	Cu	Ni	Cr	Mo	Si	Co
527708	0.21	0.72	0.008	0.007	0.04	0.83	0.33	0.55	0.33	0.01
Ref: RDY Material Certification, Catawba QA Vault DCP pkg.										

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
11Z-0933	925C	880C	3.67 Hr	WC	Austenizing- Quench
11Z-0334	670C	650 C	5.25 Hr	FC -395C/AC	Tempering
PWHT	620C	590C	13 Hr 49Min	FC	Stress Relief
Ref: RDY Material Certification, Catawba QA Vault DCP pkg.					

Nil Ductility Temperature, Tndt :

Tndt : -23 C (-13F)
Ref: RDY Material Certifications, Catawba's QA Vault DCP data pkg.

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F):	-13
Ref: Catawba Tech. Spec.	

Method of Reference Temperature (RTndt) Determination:
Based on drop weight data.

Attachment 2

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Charpy V-notch Impact Test Results:

Charpy V-notch Impact Test Results for Heat No. 527708:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
4304K11	40	102	79
4304K12	40	127	94
4304K13	40	110	83
4304K21	40	149	94
4304K22	40	157	91
4304K23	40	113	83
4304K31	-148	7.5	12
4304K32	-130	5.2	4
4304K33	-112	6.4	4
4304K34	-94	5.2	4
4304K41	-76	48.7	43
4304K42	-40	80	67
4304K43	-22	103	75
4304K44	-4	98	79
4304K51	14	87	71
4304K52	32	90	75
4304K53	50	103	79
4304K54	68	120	79
4304K61	104	142	91
4304K62	140	147	94
4304K63	158	145	99
4304K64	176	147	94
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
Orientation: tangential direction with axis of notch axial			

Attachment 2

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Additional Charpy V-notch Impact Test Results for Heat No. 527708:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Lateral Expansion (mils)
4304K71	-4	58	47
4304K72	-4	58	51
4304K73	-4	54	47
4304K74	-76	12	12
4304K75	-76	12	8
4304K76	-76	12	8
4304K77	-148	2	2
4304K78	-148	4	4
4304K79	-148	5	4
4304K80	60	126	83
4304K81	60	117	71
4304K82	60	107	71
4304K83	113	150	94
4304K84	113	136	91
4304K85	113	149	94
4304K86	176	155	99
4304K87	176	159	91
4304K88	176	151	91
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
Orientation: tangential direction with axis of notch axial			

Additional Charpy V-notch Impact Test Results for Heat No. 527708:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Lateral Expansion (mils)
4304K1R	-4	87	59
4304K2R	-4	82	67
4304K3R	-4	70	59
4304K4R	-76	10	8
4304K5R	-76	31	18
4304K6R	-76	20	24
4304K7R	-148	5	4
4304K8R	-148	3	4
4304K9R	-148	3	4
4304K10R	60	94	71
4304K11R	60	104	75
4304K12R	60	98	79
4304K13R	176	133	91
4304K14R	176	136	94
4304K15R	176	135	87
4304K16R	104	130	91
4304K17R	104	123	83
4304K18R	104	112	83
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
Orientation: radial direction with axis of notch axial			

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Piece Description: Beltline Weld Materials

Location	Heat No.	Weld Contr. #	Flux	Flux Lt. #
Intermediate to Lower Shell Girth (Seam W05)	895075	R747	Grau Lo	P46
Intermediate to Lower Shell Girth (W05 -Root Region)	899680	P710	Grau Lo	P23

Composition:

Weld Cont. #	C	Cu	P	S	Si	Ni	Cr	Mo	Mn	Co	
R747	0.069	0.05	0.010	0.01	0.22	0.7	0.05	0.58	1.97	--	Note (1)
	0.049	0.066	0.015	0.006	0.27	0.71	0.036	0.56	1.73	0.01	Note (2)
	0.068	0.031	0.010	0.009	0.208	0.74	0.006	0.58	1.97	0.01	Note (3)
P710	0.052	0.03	0.009	0.015	0.25	0.75	0.04	0.48	1.97	--	Note (4)

Note 1) Ref: RDY Weld Test Report, Lab No. R747
Note 2) Ref: WCAP-9734, (Original Analysis)
Note 3) Ref: WCAP-11527, Table 4-1; (Capsule Z, Specimen MW-85)
Note 4) Ref: RDY Weld Test Report, Lab. No. P710 dated 7/72

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
1	635C	595C	11 Hr 45 Min	Furnace Cooled	Total PWHT
Ref: RDY Material Certifications, Catawba's QA Vault DCP data pkg.					

Nil Ductility Temp. (Tndt) (F):

Location	Heat No.	Tndt	
Seam W05	895075	-76	Ref: WCAP-9734
Seam W05 HAZ	895075	-67	Ref: WCAP-9734

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F):	-51	(Weld Center)
RTndt (F):	-67	(HAZ)
Ref: WCAP-9734		

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils CVN tests).

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Charpy V-notch Impact Test Results:
(Also see attached WCAP-9734, section 3 - Preirradiation Testing)

Charpy V-notch Impact Test Results for Weld Control No. R747:

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
R747 1	10	41	35
R747 2	10	62	55
R747 3	10	57	55
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			

Charpy V-notch Impact Test Results for Weld Control No. R710 (root weld):

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
1	10	58	43
2	10	43	39
3	10	39	55
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			

Charpy V-notch Impact Test Results for HAZ of Ring 04 (HT No. 527708):

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
4345K19,20,21	68	152	87
4345K22,23,24	-76	74	43
4345K25	-184	31	20
4345K26	-148	28	16
4345K27	-112	87	55
4345K28	-40	125	75
4345K29	212	159	67
4345K30	-4	115	71
4345K31	32	137	87
4345K32	86	158	91
4345K33	122	193	99
4345K34	140	153	97
4345K35	158	187	91
4345K36	176	178	91
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			

Attachment 2

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Charpy V-notch Impact Test Results for weld sample C, ring 04/05

Impact No.	Temp (F)	Absorbed Energy (ft-lb's)	Laterial Expansion (mils)
4345K1,2,3	68	105	81
4345K4,5,6	-76	21	23
4345K7	212	127	87
4345K8	-148	3	4
4345K9	-112	9	12
4345K10	-40	45	39
4345K11	-22	58	47
4345K12	-4	55	43
4345K13	32	93	71
4345K14	86	113	91
4345K15	122	124	83
4345K16	140	144	99
4345K17	158	129	94
4345K18	176	130	87
Ref: RDY Material Certifications, Catawba QA Vault DCP data pkg			
All specimens taken at center of weld			

Attachment 2

Catawba Nuclear Station, Unit 1, RDY Vessel 30743

Piece Description: Surveilance Material

RDY Item Number	Heat Number(s)
Ring 05	411343
Weld Sample C (Ring 04/05)	895075

Surveilance Material Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
11Z-0310	1697 F	1679 F	3 Hrs 30 Min	WC	Austenizing- Quench
11Z-0311	1247 F	1220 F	6 Hrs	FC - 450C	Tempering
Ring 05	1165 F	1115 F	22 Hrs	FC	Stress Relief
Weld	1165 F	1115 F	15 Hrs	FC	Stress Relief
Ref: WCAP-9734, (Original Analysis)					

Nil Ductility Temperature, Tndt :

Tndt :	
Plate:	-40 F
Weld:	-76 F
HAZ	-67 F
Ref: WCAP-9734, (Original Analysis)	

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F) :	-8 F	(Ring 05)
	-51F	(Weld Metal)
Ref: Catawba Tech Spec		

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils CVN tests).

Charpy V-notch Impact Test Results:

See attached WCAP-9734, section 3 - Preirradiation Testing

ATTACHMENT 3

WESTINGHOUSE CLASS 3

WCAP-9734

DUKE POWER COMPANY
CATAWBA UNIT NO. 1
REACTOR VESSEL RADIATION
SURVEILLANCE PROGRAM

S. E. Yanichko

July 1980

APPROVED: T. R. Mager

T. R. Mager, Manager
Metallurgical and NDE Analysis

Work Performed Under DCP-106

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

ATTACHMENT 3
· WCAP 9734

SECTION 3
PREIRRADIATION TESTING

3-1. CHARPY V-NOTCH TESTS

Charpy V-notch impact tests were performed per ASTM E-23 with specimens from the vessel intermediate shell forging 05. Specimens of both axial and tangential orientations were tested at various test temperatures in the range —100 to 210°F yielding a full Charpy V-notch transition curve in both orientations (tables 3-1 and 3-2 and figures 3-1 and 3-2). Tests were also performed on weld metal and HAZ metal at various temperatures from —200 to 210°F. The results are reported in tables 3-3 and 3-4 and figures 3-3 and 3-4.

The specimens were tested on a Sontag SI-1 impact machine which is inspected and calibrated every 12 months. Charpy V-notch impact specimens of known energy values, supplied by the Watertown Arsenal, are used for the calibration.

3-2. TENSILE TESTS

Table 3-5 and figures 3-5, 3-6, and 3-7 give results of tensile tests (per ASTM E-8 and E-21 test criteria) from vessel intermediate shell forging 05 and from the weld metal Specimens from the shell forging were tested at room temperature, 300°F, and 550° F in both the axial and tangential directions.

An Instron TT-C tensile testing machine was used with the standard Instron gripping devices. A Baldwin-Lima-Hamilton Class B-1 extensometer and chart recorder provided a full stress-strain curve for each specimen. The chart recorder was calibrated to the Class B-1 extensometer. The measurement and control of speeds in the tests conformed to ASTM A370-68 (Mechanical Testing of Steel Products). The Instron TT-C and the Baldwin-Lima-Hamilton extensometer are calibrated by test equipment which has been certified by the National Bureau of Standards. A typical stress-strain curve is shown in figure 3-8.

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3-3. DROPWEIGHT TESTS

The nil-ductility transition temperature (T_{NDT}) was determined for forging 05 and the core region weld metal and heat-affected zone by dropweight tests (ASTM E-208) performed at Rotterdam Dockyard Co. The following results were obtained:

Material	T_{NDT} (°F)
Forging 05	—40
Weld Metal	—76
HAZ	—67

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TABLE 3-1

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE CATAWBA UNIT NO. 1 REACTOR
PRESSURE VESSEL INTERMEDIATE SHELL
FORGING 05 (TANGENTIAL ORIENTATION)

Test Temperature (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
—100	12	0	1
—40	15	0	10.5
—40	11	0	8
0	15	23	12
0	50	27	37
0	17	33	15
10	17	42	14
10	24	33	19.5
20	85	45	58
20	85	48	62
20	75	52	51
75	126	73	82
75	116	66	80
120	139	81	85
120	130	81	86
210	158	100	94
210	177.5	100	83
210	168	100	88

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TABLE 3-2

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE CATAWBA UNIT NO. 1 REACTOR
PRESSURE VESSEL INTERMEDIATE
SHELL FORGING 05 (AXIAL ORIENTATION)

Test Temperature (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
—100	2	0	0
—40	8	0	5.5
—40	16	0	11
—15	48	25	34
0	27	20	19
0	35	30	25
0	51	30	38
20	60	34	44
20	56	40	44
20	69	45	52
75	81	52	60
75	89	61	67
120	111	82	77
120	119	81	79.5
150	135	100	90
210	137	100	88
210	132	100	91
210	133	100	87

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TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE CATAWBA UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

Test Temperature (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
—100	12	13	6
—60	13	28	9.5
—60	15	33	11.5
—40	37	33	26
—40	26	28	18
—16	40	42	31
—16	60	37	45.5
—16	54	50	39
5	44	54	38
25	91	72	66
25	98.5	87	74
75	119	87	86
75	110	95	77
120	132	100	90.5
120	119	100	87
210	133	100	90
210	130	100	90
210	124	100	89

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TABLE 3-4

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE CATAWBA UNIT NO. 1 REACTOR PRESSURE
VESSEL CORE REGION WELD
HEAT-AFFECTED-ZONE MATERIAL

Test Temperature (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
—200	13	0	4
—170	35	33	22
—150	96	80	54
—100	82	38	43
—40	74	54	48
—40	108	50	58
— 7	135	81	74
— 7	110	77	60
— 7	127	81	73
25	159	100	85
25	147.5	89	81
75	150	100	83
75	138.5	93	80.5
120	154	100	84
120	162	100	83
210	180	100	81.5
210	159	100	85
210	166	100	83

ATTACHMENT 3
WCAP 9734

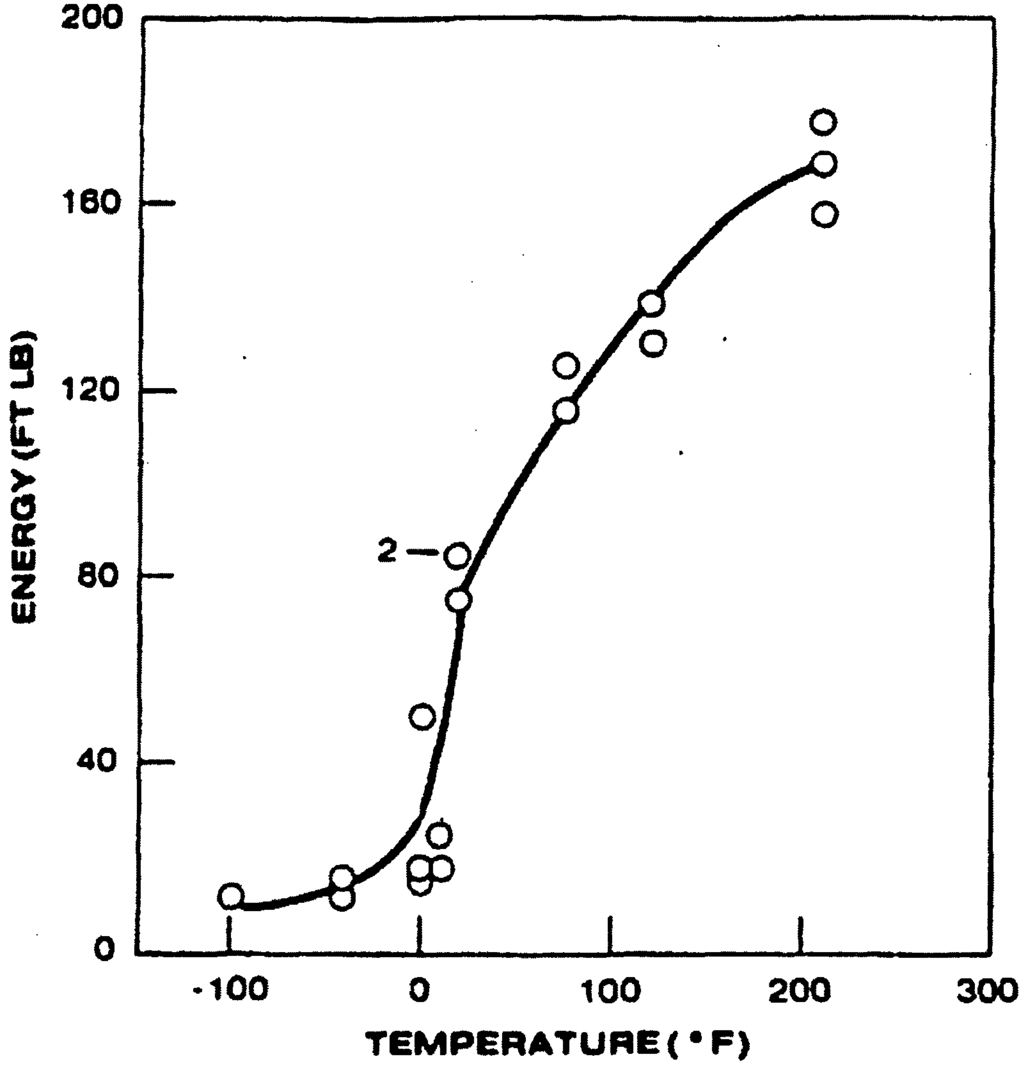

FIGURE 3-1 PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE CATAWBA UNIT NO. 1 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL FORGING 05 (TANGENTIAL ORIENTATION)

ATTACHMENT 3
WCAP 9734

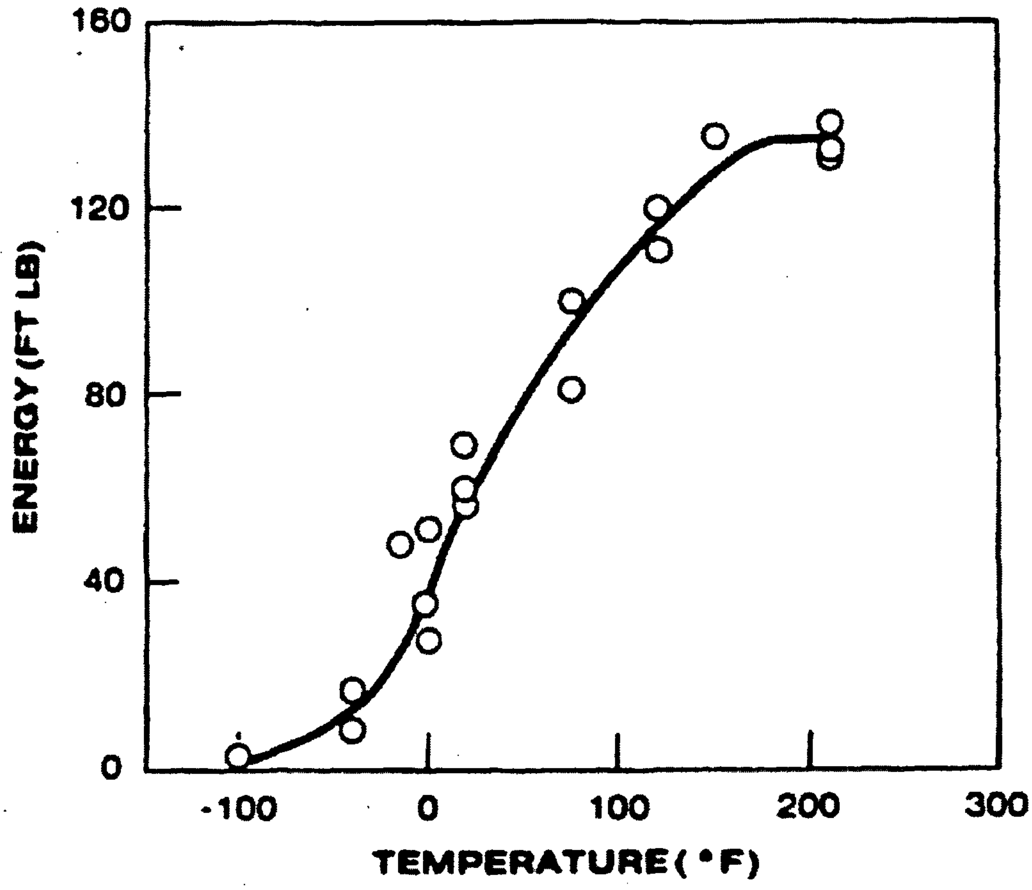

FIGURE 3-2 PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE CATAWBA UNIT NO. 1 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL FORGING 05 (AXIAL ORIENTATION)

ATTACHMENT 3
WCAP 9734

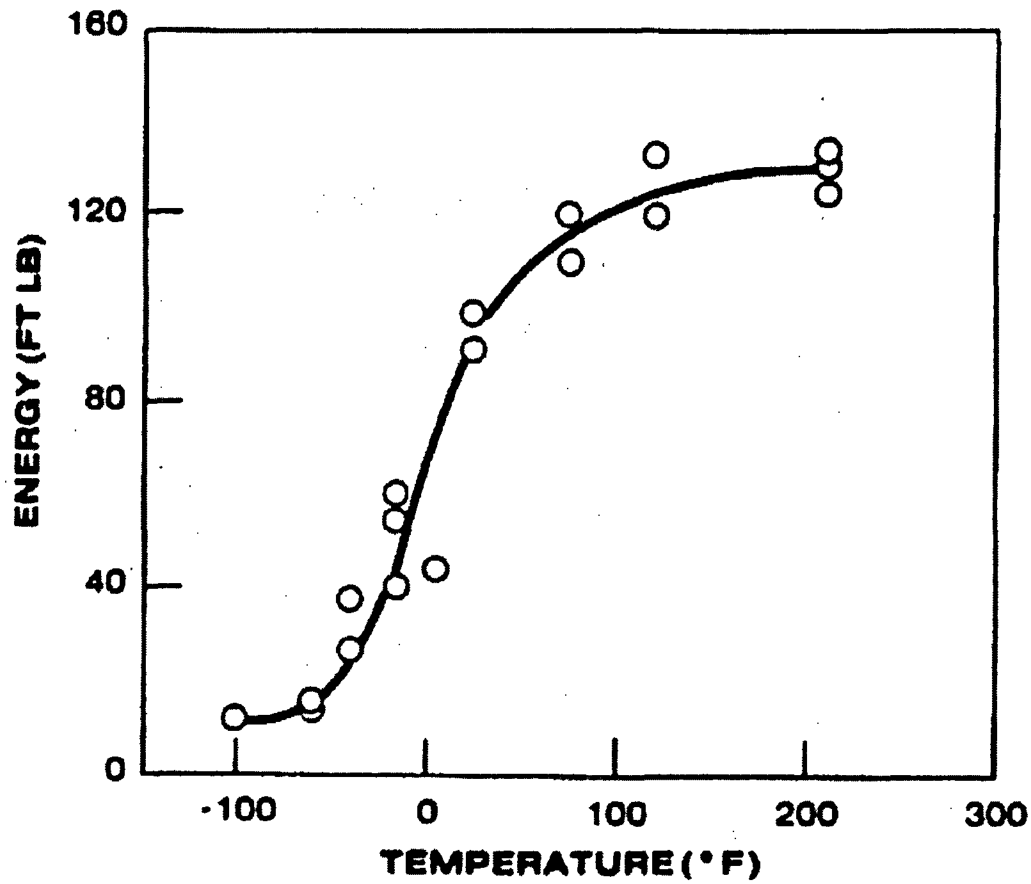

FIGURE 3-3 PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE CATAWBA UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

ATTACHMENT 3
WCAP 9734

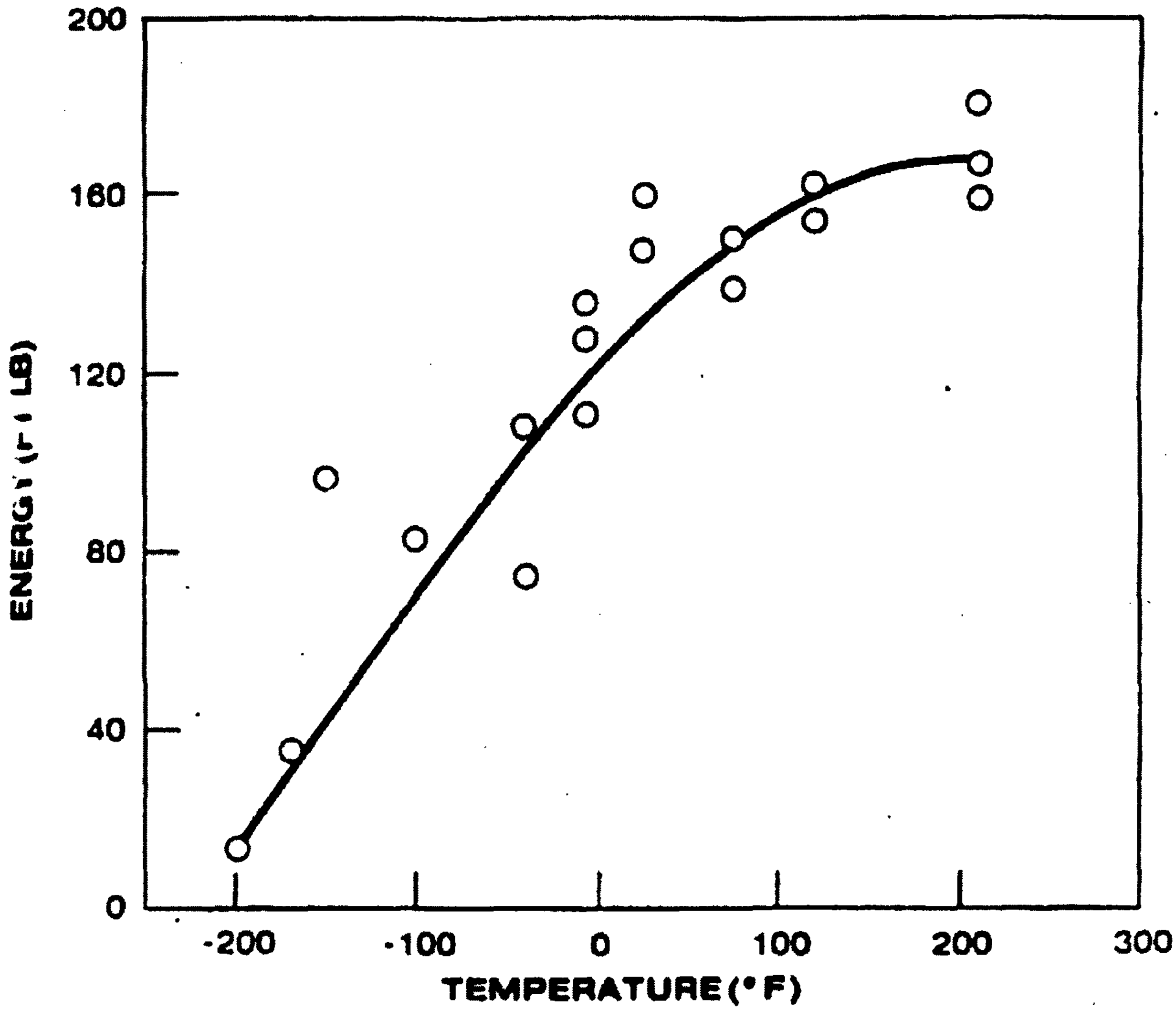

FIGURE 3-4 PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE CATAWBA UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD HEAT-AFFECTED-ZONE MATERIAL

ATTACHMENT B
McGUIRE NUCLEAR STATION'S RESPONSE TO GENERIC LETTER 92-01

The following is McGuire Unit's 1 & 2 response to Generic Letter 92-01.

Question 1.

Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

The McGuire Unit 1 reactor vessel surveillance program was developed to meet the requirements in ASTM E185-73 as described in WCAP-9195. However, the McGuire Unit 1 reactor vessel surveillance capsules were built to the applicable standard during the time of fabrication, ASTM E185-70. ASTM E185-70 required that the weld be fabricated from core region base material with the highest nil-ductility transition (NDT) temperature. ASTM E185-73 requires that the surveillance weldment be fabricated using the limiting core region base material selected on the basis of chemical composition (CU and P), initial RT ndt, and upper shelf charpy impact energy. Because of the change in surveillance material selection procedures, the McGuire Unit 1 surveillance weldment does not contain base metal from the limiting core region plate as required by ASTM E185-73. Therefore, the weld heat-affected-zone specimens used in the surveillance program are representative of core region plate B5012-2 rather than of limiting plate B5012-1.

McGuire Unit 2 reactor vessel surveillance program was developed to meet the requirements of ASTM E185-73, as described in WCAP-9489.

Both McGuire Units have higher lead factors than the 3.0 stated in ASTM E185-73. McGuire Unit 1 lead factors are presently between 4.76 and 5.33. McGuire Unit 2 lead factors are presently between 4.06 and 4.76. The later edition of Appendix H deleted the lead factor limits. An exemption to Appendix H was issued for the McGuire Units per NUREG-0422 Supplement No. 2, issued in March 1, 1979.

The latest surveillance withdrawal schedule per ASTM 185-82, recommended in our last surveillance capsule report (WCAP-12354), which will reduce exposure time to the capsule has not yet been incorporated in the Technical Specifications for McGuire Unit 1. A new withdrawal schedule will be submitted to the NRC by October 1992, meeting the ASTM 185-82 recommended withdrawal schedule. At the same time a request to remove the withdrawal schedule from the Technical Specifications per GL 91-01 will be submitted.

Question 2.

Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

a. **Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.**

The McGuire Unit 1 & 2 reactor vessel beltline materials are currently predicted to maintain greater than 50 ft-lbs upper shelf energy through the end of life.

Using the method described in Reg. Guide 1.99 rev. 2 paragraph C.1.2 each beltline material was examined for McGuire Unit's 1 & 2. All beltline materials had greater than 50 ft-lbs upper shelf at the end of license. The chemistry and fluence documented in the most recent surveillance capsule reports (WCAP-12354 and WCAP-12556) were used for this verification.

b. **Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:**

The McGuire Unit 1 reactor vessel was built to the ASME B&PV code, Section III, 1971 Edition including addenda through the summer 1971 as documented in the reactor vessel QA data package and reported in paragraph 5.2.4.1 of the FSAR.

The McGuire Unit 2 reactor vessel was built to the ASME B&PV code, Section III, 1971 Edition including addenda through the winter 1971 as documented in the reactor vessel QA data package and reported in paragraph 5.2.4.1 of the FSAR.

Attached are all materials data necessary to answer questions 2b(1) - (6).

(1) **the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;**

See attachment 2 for plate and forging material.
See attachment 3 and 4, section 3 of WCAP-9195 and WCAP-9489 for surveillance material.
See response to I&E Bulletin 78-12 for additional charpy and drop weight test on beltline weld material.

(2) **the heat treatment received by all beltline and surveillance materials;**

See attachment 2.

(3) **the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;**

See attachment 2.

(4) **the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;**

See attachment 2.

(5) **the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material: and**

See attachment 2.

(6) **the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.**

See attachment 2.

Question 3.

Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

a. **How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 xF were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.**

   Operating T-cold temperatures below 525F are not applicable to the McGuire Units. As shown in McGuire's FSAR figure 5.3.7 (see attachment 1) T-cold remains fairly constant at approximately 557F with T-avg ranging from 557F to 588F. Technical Specification 3.1.1.4 require a minimum T-avg temperature for criticality of 551F. An LER search indicated this tech spec was violated once when the minimum NC system temperature of 536.7 F was reached during a 79 minute event. This event was documented in the Duke Power Company, McGuire Nuclear Station's Reportable Occurrence Report No. 370/83-62, dated December 1, 1983.

b. **How their surveillance results on the predicted amount of embrittlement were considered.**

   With regard to the GL 88-11 response, neither McGuire Unit 1 nor Unit 2 had two sets of surveillance data available and therefore surveillance data was not used to determine the adjusted RTndt. Reg. Guide 1.99 rev.2, regulatory position 1 was used to determine the adjusted RTndt for each McGuire unit in WCAPS 12354 AND 12556.

c. **If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.**

The measured increase in reference temperature for McGuire Units 1 & 2 has not exceeded the mean-plus-two standard deviations predicted by Reg. Guide 1.99 rev. 2. The measured decreases in charpy upper shelf energy has not exceeded the predicted percent USE decrease except for the weld metal in capsule U, which was 5% greater than predicted by Reg. Guide 1.99 rev. 2. Capsule U was McGuire Unit 1's first capsule examined in 1985. Since then the same weld metal has been examined from capsule X in 1989 and the percent decrease was 11% less than predicted by Reg. Guide 1.99 rev. 2. The upper shelf energy for the weld metal in unit 1 is exhibiting a more than adequate shelf level to the end of license. The intermediate shell longitudinal seam's upper shelf is 71.5 ft.lbs. and the lower shell longitudinal seam's upper shelf is 60.3 ft.lbs. at the end of license.

References:

McGuire Nuclear Station
Docket Nos. 50-369, -370
Response to GL 88-11
Date: November 21, 1988

NRC IE BULLETIN 78-12, Response by
letter dated June 29, 1979

Duke Power Company, McGuire Nuclear Station
Reportable Occurrence Report No. 370/83-62
Report Date: December 1, 1983

References Continued:

McGuire Reactor Vessel DAP Data Package
containing: CE Material Test Reports
Lukens Steel Co. Test Certificates

WCAP-9195 Duke Power Company
William B. McGuire Unit No. 1
Reactor Vessel Radiation Surveillance Program
Date: November 1977

WCAP-10786 Analysis of Capsule U From the Duke Power Company
McGuire Unit 1 Reactor Vessel Radiation Surveillance Program
Date: February 1985, Issued to the NRC by letter dated April 5, 1985.

WCAP-12354 Analysis of Capsule X From the Duke Power Company
McGuire Unit 1 Reactor Vessel Radiation Surveillance Program
Date: August 1989, Issued to the NRC by letter dated November 17, 1989.

McGuire Reactor Vessel DBP Data Package
containing: Rotterdam Material Test Reports

WCAP-9489 Duke Power Company
William B. McGuire Unit No. 2
Reactor Vessel Radiation Surveillance Program
Date: May 1979

WCAP-11029 Analysis of Capsule V From the Duke Power Company
McGuire Unit 2 Reactor Vessel Radiation Surveillance Program
Date: January 1986, Issued to the NRC by letter dated April 2, 1986.

WCAP-12556 Analysis of Capsule X From the Duke Power Company
McGuire Unit 2 Reactor Vessel Radiation Surveillance Program
Date: April 1990, Issued to the NRC by letter dated August 30, 1990.

ATTACHMENT 1

5.3.7 FIGURES

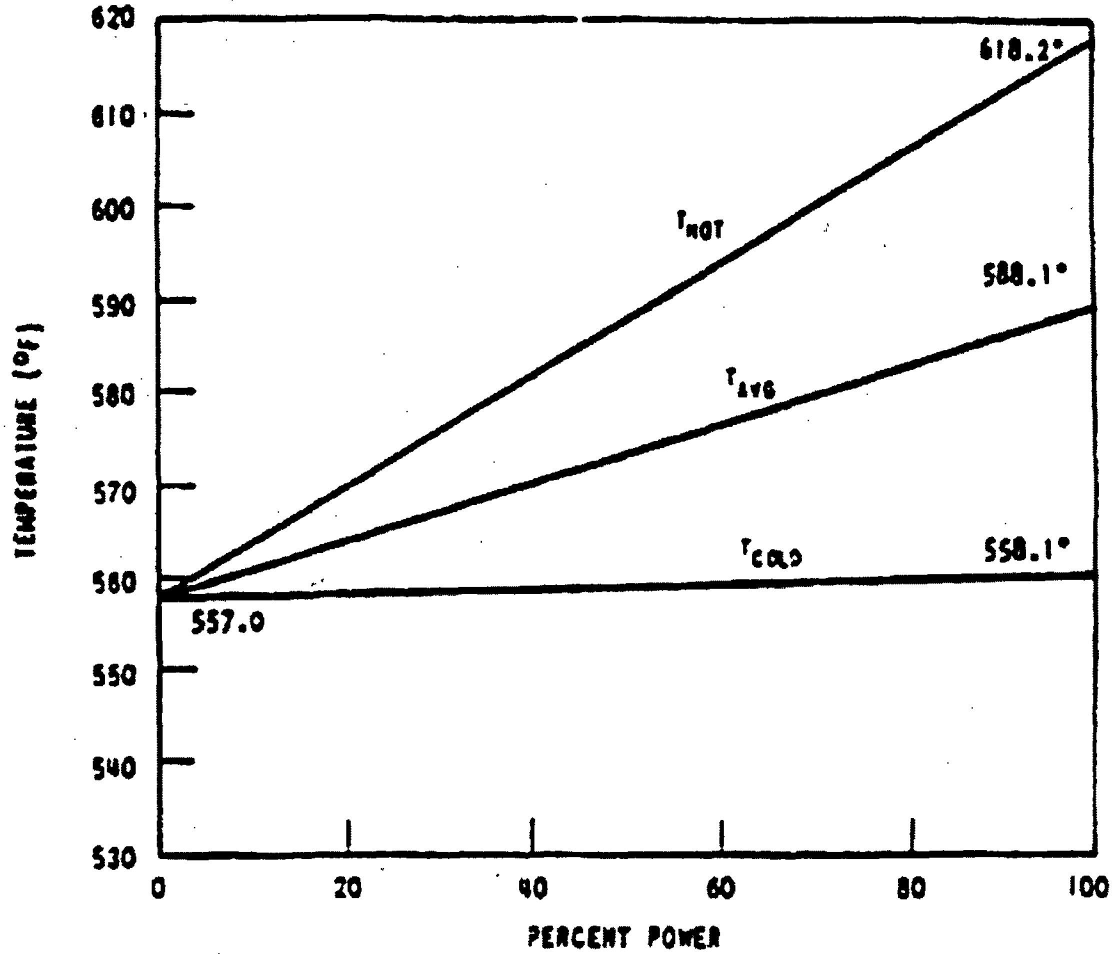

Figure 5-19.
Relationship Between Reactor Coolant System Temperature and Power

Attachment 2

McGuire Nuclear Station, Unit 1, CE Vessel 2167

Piece Description: Intermediate (Middle) Shell Segments

CE Code Number(s)	Heat Number(s)
B-5012-1	C4387-2
B-5012-2	C4417-3
B-5012-3	C4377-2

Supplier: Lukens Steel Company

Composition:

Heat No:	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Co
C4387	0.22	1.28	0.009	0.018	0.22	0.58	--	0.53	--	--
C4387-2	0.21	1.28	0.010	0.016	0.23	0.60	--	0.57	0.13	--
(Supplemental Analysis of Plate B5012-1 (Heat C4387-2): Cu=0.087 per WCAP-9195										
C4417	0.22	1.31	0.012	0.018	0.24	0.60	--	0.55	--	--
C4417-3	0.25	1.33	0.011	0.016	0.25	0.62	--	0.57	0.13	--
C4377	0.23	1.35	0.015	0.015	0.23	0.65	--	0.52	--	--
C4377-2	0.22	1.36	0.013	0.015	0.24	0.66	--	0.54	0.10	--
Ref: WCAP-12354 ,Table A-1 & Lukens Steel Co. Test Certificates (McGuire QA Vault, DAP Data Pkg Pg's 00323 &00324)										

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
1	1650 F	1550 F	4 Hrs	Water (WC)	Austenizing - Quenching
2	1250 F	1200 F	4 Hrs	Air Cool (AC)	Tempering
3	1175 F	1125 F	40 Hrs	Furnace Cool (FC)	Stress Relief
Ref: CE Materials Test Reports in McGuire QA Vault					

Nil Ductility Temp. (Tndt) (F):

Heat No. C4387-2	-30	(McGuire QA Vault, DAP Data Pkg. Pg. 00325)
Heat No. C4417-3	0	(McGuire QA Vault, DAP Data Pkg. Pg. 00326)
Heat No. C4377-2	-20	(McGuire QA Vault, DAP Data Pkg. Pg. 00327)
Ref: CE Materials Test Reports in McGuire QA Vault		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E -208)

Attachment 2

McGuire Nuclear Station, Unit 1, CE Vessel 2167

Reference Transition Temperature (RTndt) (F):

Heat No. C4387-2	(Material Code: B5012-1)	34
Heat No. C4417-3	(Material Code: B5012-2)	0
Heat No. C4377-2	(Material Code: B5012-3)	-13
Ref: WCAP -12354 (McGuire Unit 1 Reactor Vessel Radiation Surveillance Program Analysis of Capsule X, Table A-1)		

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).

Charpy V-notch Impact Test Results:

Temperature (F)	CE code #	Run	-40	10	40	110	160	212
Absorbed Energy (ft-lb's)	B-5012-1	1	17	32	92	108	143	139
" "	B-5012-1	2	19	52	74	122	145	136
" "	B-5012-1	3	12	42	68	--	--	--
Lateral expansion (mils)	B-5012-1	1	12	25	68	75	86	89
" "	B-5012-1	2	16	38	52	77	87	85
" "	B-5012-1	3	9	31	50	--	--	--
Absorbed Energy (ft-lb's)	B-5012-2	1	12	38	78	99	132	--
" "	B-5012-2	2	17	80	55	110	137	--
" "	B-5012-2	3	11	70	86	125	142	--
Lateral expansion (mils)	B-5012-2	1	13	31	59	71	87	--
" "	B-5012-2	2	15	59	44	79	89	--
" "	B-5012-2	3	10	4	66	85	88	--
Absorbed Energy (ft-lb's)	B-5012-3	1	15	78	71	127	153	--
" "	B-5012-3	2	35	60	101	123	153	--
" "	B-5012-3	3	16	61	104	129	156	--
Lateral expansion (mils)	B-5012-3	1	12	60	55	84	88	--
" "	B-5012-3	2	26	46	72	75	91	--
" "	B-5012-3	3	14	47	73	86	90	--
Ref: CE Materials Test Reports in McGuire QA Vault								
Orientation: Parallel to rolling direction and notch perpendicular to plate surface								

Attachment 2

McGuire Nuclear Station, Unit 1, CE Vessel 2167

Piece Description: Lower Shell Segments

CE Code Number(s)	Heat Number(s)
B-5013-1	C4315-1
B5013-2	C4374-2
B5013-3	C4371-2

Supplier: Lukens Steel Company

Composition:

Heat No.	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Co
C4315	0.22	1.32	0.011	0.018	0.28	0.56	--	0.58	--	--
C4315-1	0.23	1.32	0.009	0.018	0.28	0.58	--	0.57	0.14	--
C4374	0.23	1.32	0.009	0.015	0.23	0.50	--	0.53	--	--
C4374-2	0.24	1.3	0.01	0.015	0.25	0.52	--	0.55	0.10	--
C4371	0.23	1.41	0.011	0.014	0.28	0.54	--	0.55	--	--
C4371-2	0.23	1.43	0.01	0.014	0.27	0.55	--	0.56	0.10	--
Ref: WCAP-12354 ,Table A-1 & Lukens Steel Co. Test Certificates (McGuire QA Vault, DAP Data Pkg Pg's 00329, 00331 &00333)										

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench
1	1650 F	1550 F	4	Water
2	1250 F	1200 F	4	Air Cool
3	1175 F	1125 F	40	Furnace Cool
Ref: CE Materials Test Reports in McGuire QA Vault				

Nil Ductility Temp. Tndt (F):

Heat No. C4315-1	-10	(CE Job No: V-70334-001; DAP Data Pkg. Pg. 00330)
Heat No. C4374-2	-10	(CE Job No. V-70334-005; DAP Data Pkg. Pg. 00332)
Heat No. C4371-2	0	(CE Job No. V-70334-009; DAP Data Pkg. Pg. 00334)
Ref: CE Materials Test Reports in McGuire QA Vault		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Attachment 2

McGuire Nuclear Station, Unit 1, CE Vessel 2167

Reference Transition Temperature (RTndt) (F):

Heat No. C4315-1	(Material Code: B5013-1)	0
Heat No. C4374-2	(Material Code: B5013-2)	30
Heat No. C4371-2	(Material Code: B5013-3)	15
Ref: WCAP -12354 (McGuire Unit 1 Reactor Vessel Radiation Surveillance Program Analysis of Capsule X. TABLE A-1)		

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).

Charpy V-notch Impact Test Results:

Temperature (F)	CE code #	Run	-40	10	40	110	160
Absorbed Energy (ft-lb's)	B-5013-1	1	9	48	69	95	132
" "	B-5013-1	2	28	45	65	105	130
" "	B-5013-1	3	12	48	53	97	125
Laterial expansion (mils)	B-5013-1	1	6	35	50	70	90
" "	B-5013-1	2	22	38	47	77	87
" "	B-5013-1	3	12	35	40	69	84
Absorbed Energy (ft-lb's)	B-5013-2	1	15	53	58	118	143
" "	B-5013-2	2	22	49	67	128	148
" "	B-5013-2	3	14	56	75	119	136
Laterial expansion (mils)	B-5013-2	1	12	41	43	79	87
" "	B-5013-2	2	19	37	49	83	92
" "	B-5013-2	3	12	43	54	81	85
Absorbed Energy (ft-lb's)	B-5013-3	1	24	40	48	97	128
" "	B-5013-3	2	21	51	51	100	134
" "	B-5013-3	3	14	38	50	85	130
Laterial expansion (mils)	B-5013-3	1	17	29	32	68	86
" "	B-5013-3	2	15	38	38	64	87
" "	B-5013-3	3	12	25	37	60	85
Ref: CE Materials Test Reports in McGuire QA Vault							
Orientation: Parallel to rolling direction, notch perpendicular to plate surface							

Attachment 2

McGuire Nuclear Station, Unit 1, CE Vessel 2167

Piece Description: Beltline Weldments

Location	Heat No.	Number	Flux	Flux Lot No.
Intermediate Shell Long. Seams	20291/12008	M1.22	Linde 1092	3833\ 3854
Intermediate to Lower Shell Girth	83640	G1.39	Linde 0091	3490
Lower Shell Long. Seam	21935/12008	M1.32	Linde 1092	3889
Lower Shell Long. Seam	21935/33A277	M1.33	Linde 1092	3889
Lower Shell Long. Seam	305424	M1.34	Linde 1092	3889

Composition:

Weld	(Original Analysis)				(Capsule U analysis, Specimen DW-15)					
Control #	Ni	Cu	S	P		Ni	Cu	P	S	
M1.22	0.88	0.21	0.008	0.011		0.91	0.20	0.01	--	
G1.39	0.20	0.05								
M1.32	0.87	0.20								
M1.33	0.68	0.21								
M1.34	0.64	0.30								
Ref: WCAP-12354, Table 4-1 & Table A-1										

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench
1	1175	1125	40 Hrs	Furnace Cooled
Ref: CE Materials Test Reports in McGuire QA Vault				

Nil Ductility Temp. (Tndt) (F):

Location	Heat No.	Tndt
Intermediate Shell Long. Seams	20291/12008	-60
Intermediate to Lower Shell Girth	83640	-70
Lower Shell Long. Seam	21935/12008	--
Lower Shell Long. Seam	21935/33A277	--
Lower Shell Long. Seam	305424	--
Ref: WCAP-12354 Table A-1		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

Location	Heat No.	RTndt	
Intermediate Shell Long. Seams	20291/12008	-50	Note (1)
Intermediate to Lower Shell Girth	83640	-70	Note (1)
Lower Shell Long. Seam	21935/12008	-56	Note (2)
Lower Shell Long. Seam	21935/33A277	-56	Note (2)
Lower Shell Long. Seam	305424	-56	Note (2)
Ref: WCAP -12354			

Method of Reference Temperature (RTndt) Determination:
Note (1) : RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).
Note (2) : Generic Mean per RG 1.99 Rev 2

Attachment 2

McGuire Nuclear Station, Unit 1, CE Vessel 2167

Piece Description: **Surveillance Material**

Description	CE Code Number(s)	Heat Number(s)
Inter. Shell Plate	B5012-1	C4387-2
Test Plate B	(B5012-2/B5012-3)	Plates: C4417-3 / C4377-2
Weldment (Seam 2-442B)	(Inter. Shell Long. Seam)	Weld: 20291/12008
		(flux- Linde 1092, Lot 3854)

Supplier: Lukens Steel Company

Composition:

Heat No.		S	P	C	Si	Ni	Cr	Mo	Cu	Mn
C4387		0.018	0.009	0.22	0.22	0.58	--	0.58	--	1.28
C4387-2		0.016	0.010	0.21	0.23	0.60	--	0.57	0.13	1.26
Westinghouse Analysis of Plate B5012-1, Cu= 0.087, Ref. WCAP-9195, Table A-2										
20291/12008		0.008	0.011	0.10	0.24	0.88	0.04	0.55	0.21	1.36
Analysis of Capsule U Specimen DW-15; Cu= 0.20, Ni= 0.91 Ref. WCAP-10786										

Heat Treatment:

Plate:	1550/1650F (4 hrs) Water Quench
	1200/1250F (4 hrs) Air Cooled
	1125/1175F (40 hrs) Furnace Cooled
Weld:	1175/1125 (40 Hrs) Furnace Cooled
Ref: CE Material Test Reports	

Nil Ductility Temperature , Tndt (F):

Plate: Heat No. C4387-2	-30	
Weld: Heat No. 20291/12008	-60	(-50 F in HAZ)
Ref: CE Vessel Weld Test Report , Job No. V70333-017 (3/2/70) Located in McGuire QA Vault, DAP Cert. pkg. page 000181		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

Plate:	Heat No. C4387-2 (Code B5012-1)	34
Weld:	Heat No. 20291/12008 (Weld Control Number M1.22)	-50
Ref: WCAP -12354 , Analysis of Capsule X, . TABLE A-1		

(Based on Actual Data)

Attachment 2

McGuire Nuclear Station, Unit 2, RDY Vessel 30664

Piece Description: Intermediate Shell Forging

RDY Item Number	Heat Number(s)
Ring 05	526840

Material Specification: ASME SA-508 Cl. 2

Composition:

Heat No.	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Co
526840	0.19	0.76	0.012	0.014	0.26	0.85	0.38	0.56	0.16	0.008
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg										

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
11Z-0283	925 C	880 C	3.5 Hrs	WC	Austenizing- Quenching
11Z-0284	670 C	650 C	7.5 Hrs	FC - 450C / AC	Tempering
PWHT	620 C	595 C	22.1Hrs	FC	Stress Relief
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg					

Nil Ductility Temperature, Tndt

Tndt :	-20 C	(-4F)
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg, Pg 329		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F):	-4
Ref: WCAP-12556, Table A-1	

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).

Charpy V-notch Impact Test Results:

Temperature (F)	Heat No.	Run	-148	-76	-4	60	113	176
Absorbed Energy (ft-lb's)	526840	1	6.3	40.3	39.7	137	160	144
" "	526840	2	7.5	42.5	86.4	146	152	140.5
" "	526840	3	6.3	43.2	85.8	122	148	137.1
Lateral expansion (mils)	526840	1	4	35	35	91	79	83
" "	526840	2	4	32	71	91	94	91
" "	526840	3	6	35	67	79	87	87
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg								
Orientation: tangential direction with axis of notch, axial								

Attachment 2

McGuire Nuclear Station, Unit 2, RDY Vessel 30664

Piece Description: Lower Shell Forging

RDY Item Number	Heat Number(s)
Ring 04	411337-11

Material Specification: ASME SA-508 Cl. 2

Composition:

Heat No.	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Co
411337-11	0.19	0.7	0.004	0.007	0.28	0.88	0.38	0.6	0.15	0.015
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg										

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
11Z-0310	925 C	860 C	4 Hrs	WC	Austenizing - Quench
11Z-0311	670 C	650 C	8 Hrs	FC-450 C /AC	Tempering
PWHT	620 C	590 C	18.3 Hrs	FC	Stress Relief
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg.					

Nil Ductility Temperature, Tndt :

Tndt :	-35 C	(-30F)
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg, Pg 319		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F):	-30
Ref: WCAP-12556, Table A-1	

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).

Charpy V-notch Impact Test Results:

Temperature (F)	Heat No.	Run	-148	-76	-4	60	113	176
Absorbed Energy (ft-lb's)	411337-11	1	5.8	33.4	107	124	147	152.6
" "	411337-11	2	8	44.9	68.5	136	152	145.7
" "	411337-11	3	7.5	19.6	69.7	133	153	146.9
Lateral expansion (mils)	411337-11	1	4	20	83	83	94	91
" "	411337-11	2	8	35	55	83	91	91
" "	411337-11	3	8	16	55	79	91	91
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg								
Orientation: tangential direction with axis of notch axial								

Attachment 2

McGuire Nuclear Station, Unit 2, RDY Vessel 30664

Piece Description: Beltline Weld Materials

Location	Heat No.	Weld Contr. #	Flux	Flux Lt. #
Intermediate to Lower Shell Girth (Seam W05)	895075	R747	Grau Lo	P46
Intermediate to Lower Shell Girth (W05 -Root Region, 27mm depth)	899680	P710	Grau Lo	P23
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg.				

Composition:

Weld Cont. #	C	Cu	P	S	Si	Ni	Cr	Mo	Mn	
R747	0.069	--	0.010	--	0.22	0.7	0.05	0.56	1.97	Note (1)
	0.055	0.031	0.016	0.015	0.29	0.73	0.03	0.55	1.81	Note (2)
	0.070	0.03	0.005	0.004	0.089	0.66	0.03	--	1.66	Note (3)
P710	0.052	0.03	0.009	0.015	0.25	0.75	0.04	0.46	1.97	Note (4)

Note 1) Ref: RDY Weld Test Report, Lab. No. R747
Note 2) Ref: WCAP-11029, Table 4-1; (Original Analysis)
Note 3) Ref: WCAP-11029, Table 4-1; (Capsule V, Specimen DW-30)
Note 4) Ref: RDY Weld Test Report, Lab. No. P710 dated 7/72

Heat Treatment:

HT No.	Max Temp	Min Temp	Time	Quench	Description
1	620C	590C	15Hrs 10Min.	Furnace Cooled	Total PWHT
Ref:	RDY Material Certification, McGuire DBP data pkg., pg 150				

Nil Ductility Temp. (Tndt) (F):

Location	Heat No.	Tndt	Reference
Seam W05	895075	-78	WCAP-11029, Table A-1
Seam W05 HAZ	895075	-76	WCAP-11029, Table A-1

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F):	-68	(Weld Center)
RTndt (F):	-76	(HAZ)
Ref: WCAP-11029, Table A-1		

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).

Attachment 2

McGuire Nuclear Station, Unit 2, RDY Vessel 30664

Piece Description: Surveillance Material

RDY Item Number	Heat Number(s)
Ring 05	526840
Weld Sample (Ring 04/05)	895075

Surveilance Material Heat Treatment:

Material	Max Temp	Min Temp	Time	Quench	Description
Ring 05	1697 F	1688 F	3.5 Hrs	WC	Austenizing- Quench
Ring 05	1238 F	1229 F	7.5 Hrs	FC - 842F/AC	Tempering
Ring 05	1165 F	1115 F	22 Hrs	FC	Stress Relief
Weld	1165 F	1115 F	15 Hrs	FC	Stress Relief
Ref:	WCAP-9489, Appendix A-1				

Nil Ductility Temperature, Tndt :

Plate:	(See Ring 05 data Report , Page 5)	
Weld:	-60 C	(-76F)
Ref: RDY Material Certifications, McGuire QA Vault DBP data pkg. Pg 156 & 157		

Method of Nil-Ductility (Tndt) Determination:
Drop Weight Test (ASTM E-208)

Reference Transition Temperature (RTndt) (F):

RTndt (F) :	-4	(Ring 05)
	-68	(Weld Metal)
Ref: WCAP-12556, Table A-1		

Method of Reference Temperature (RTndt) Determination:
RTndt based on actual data (50 Ft-Lb/ 35 Mils using T-L direction CVN tests).

ATTACHMENT 3

WESTINGHOUSE CLASS 3

WCAP-9195

DUKE POWER COMPANY
WILLIAM B. McGUIRE UNIT NO. 1
REACTOR VESSEL RADIATION
SURVEILLANCE PROGRAM

J. A. Davidson
S. E. Yanichko

November 1977

APPROVED: J. N. Chirigos, Manager
Structural Materials Engineering

Work Performed Under DAP-106

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

ATTACHMENT 3
WCAP 9195

3.3. DROPWEIGHT TESTS

The nil-ductility transition temperature (NDTT) was determined for plate B5012-1 and the core region weld metal and heat-affected zone by dropweight tests (ASTM E 208) performed at Combustion Engineering, Inc. The following results were obtained:

Material	NDTT (°F)
Plate B5012-1	-30
Weld Metal	-60
HAZ	-50

ATTACHMENT 3
WCAP 9195

TABLE 3-1
PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 1 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE B5012-1 (LONGITUDINAL DIRECTION)

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
-50	7	9	3
-50	10	9	5
0	12.5	20	10
0	38	25	26
30	42	34	32
30	65	48	46
50	56	38	21
75	77	55	55
75	86.5	58	64
100	120	85	82
100	102	80	72
150	146	100	91
150	133	100	87
175	140	100	86
175	141	100	89
210	139.5	100	87
210	135	100	89
210	144	100	90

ATTACHMENT 3
WCAP 9195

TABLE 3-2

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 1 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE B5012-1 (TRANSVERSE DIRECTION)

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
-40	23	14	13
-40	10	9	4.5
-40	20	14	12
0	33	25	27
0	33	20	23
0	35	25	28
30	49	34	36
30	41	30	28
30	35	29	34
80	57	52	45
80	46	43	40.5
80	33	25	23
110	79	92	62
110	68	59	53
110	69	65	52.5
210	103	100	80.5
210	98	100	77
210	103	100	80

ATTACHMENT 3
WCAP 9195

11,970-7

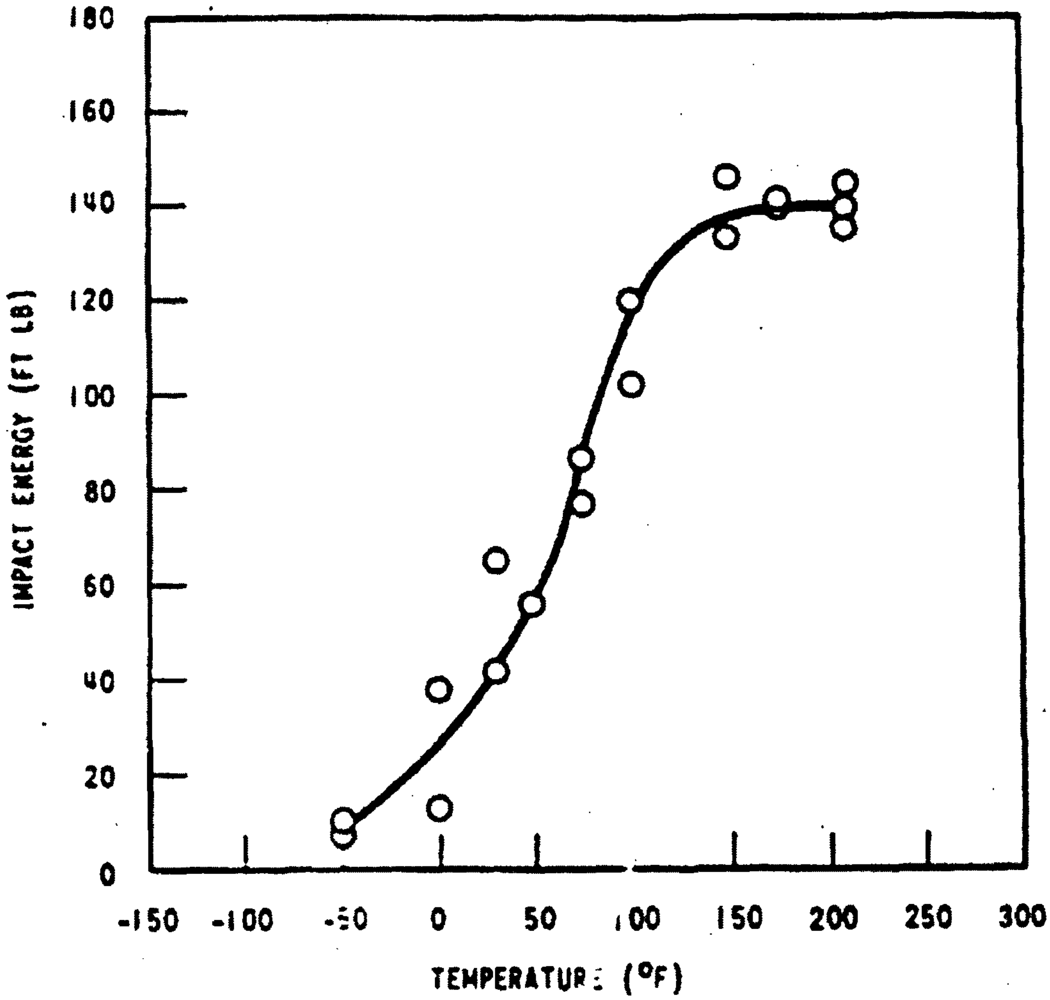

Figure 3-1. Preirradiation Charpy V-notch Impact Energy for the William B. McGuire Unit No. 1 Reactor Pressure Vessel Intermediate Shell Plate B5012-1 (Longitudinal Direction)

ATTACHMENT 3
WCAP 9195

11,370-8

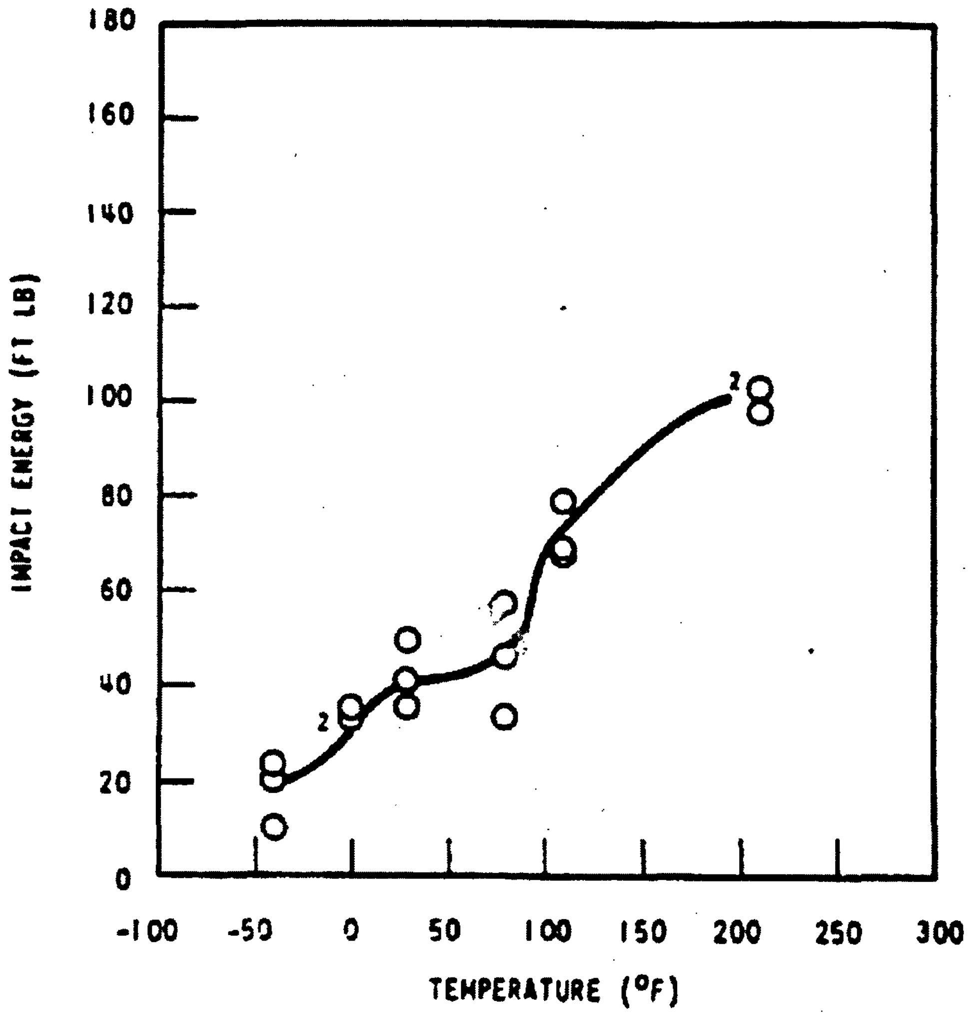

Figure 3-2. Preirradiation Charpy V-notch Impact Energy for the William B. McGuire Unit No. 1 Reactor Pressure Vessel Intermediate Shell Plate B5012-1 (Transverse Direction)

ATTACHMENT 3
WCAP 9195

TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
-100	4.5	6	1
-100	4.2	6	1.5
-35	11.5	23	11
-35	30	27	26
0	15	35	17
0	31	40	29
0	65	47	53
25	75	60	61
25	60	47	49
50	74.5	60	63
50	58	55	47
100	105	93	84
100	96	85	80
150	105	92	86
150	113	100	89
210	113	100	87
210	112	100	84
210	110	100	86

. ATTACHMENT 3
WCAP 9195

TABLE 3-4
PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD-HEAT-AFFECTED-ZONE MATERIAL

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
-100	13	20	7
-100	21	18	11
-50	24	32	14
-50	39.5	37	27
-25	58	50	36
-25	35	37	26
10	90	65	61
10	65	48	45
50	122	80	76
50	74	65	54
50	63.5	70	46
100	126	100	79
100	83	90	69
110	104	94	73
150	132	100	86
150	106	100	76
210	131	100	81
210	102	100	79

ATTACHMENT 3
WCAP 9195

11,970-9

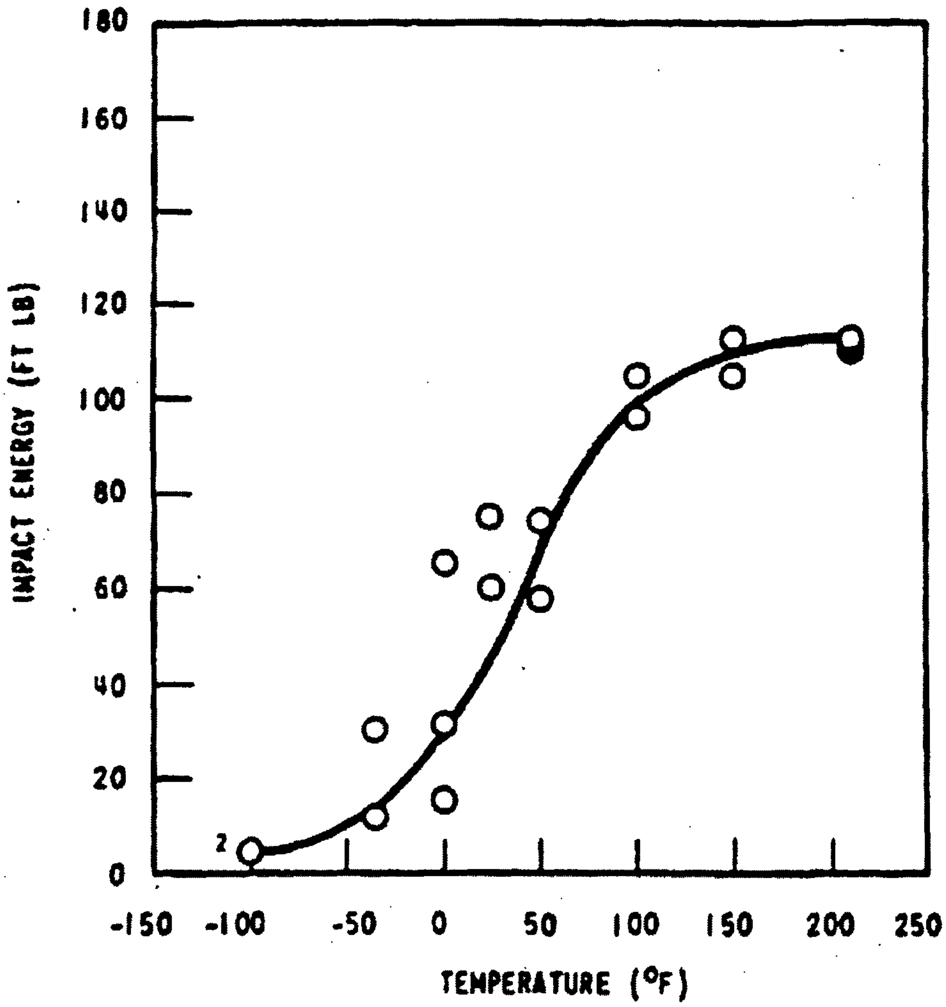

Figure 3-3. Preirradiation Charpy V-notch Impact Energy for the William B McGuire Unit No. 1 Reactor Pressure Vessel Core Region Weld Metal

·ATTACHMENT 3
·WCAP 9195

11,970-10

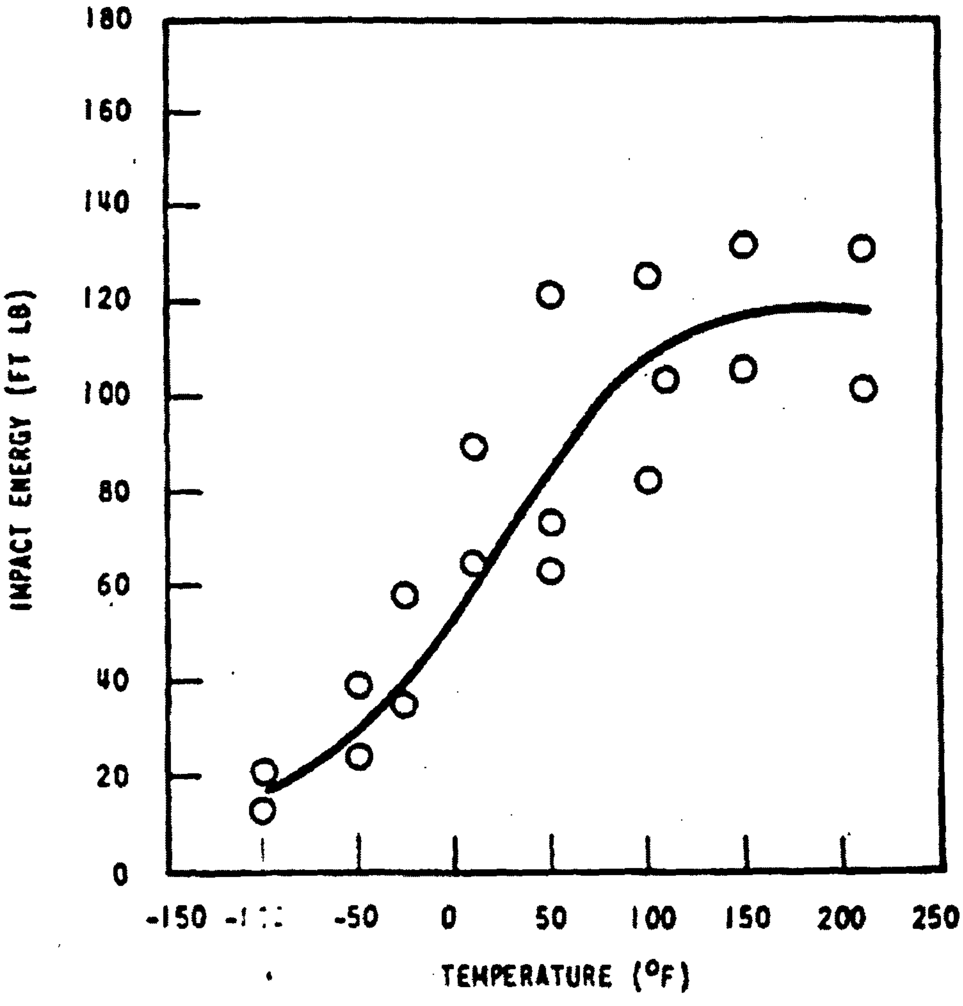

Figure 3-4. Preirradiation Charpy V-notch Energy Curve for the William B. McGuire Unit No. 1 Reactor Pressure Vessel Core Region Weld Heat-Affected-Zone Material

ATTACHMENT 4

WESTINGHOUSE CLASS 3

WCAP-9489

DUKE POWER COMPANY
WILLIAM B. McGUIRE UNIT NO. 2
REACTOR VESSEL RADIATION
SURVEILLANCE PROGRAM

May, 1979

K. Koyama

J. A. Davidson

APPROVED: J. N. Chirigos, Manager
Structural Materials Engineering

Work Performed Under DBP-106

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P.O. Box 355
Pittsburgh, Pennsylvania 15230

.ATTACHMENT 4
. WCAP 9469

3.3 DROPWEIGHT TESTS

The nil-ductility transition temperature (NDTT) was determined for the intermediate shell forging 05, the core region weld metal, and heat-affected zone by drop-weight tests (ASTM E-208) performed at the Rotterdam Dockyard Company.

The following results were obtained:

Material	NDTT (°F)
Forging 05 (HT, 526840)	-4
Weld Metal	-76
HAZ	-76

·ATTACHMENT 4
WCAP 9489

TABLE 3-1

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 2 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL FORGING 05 (TANGENTIAL ORIENTATION)

Test Temp. (^{o}F)	Impact Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
-125	2.5	0	5
-100	2.0	0	0
-100	3.5	0	0
- 70	44	10	30
- 70	69	19	51
- 50	70	23	54
- 30	93	45	67
- 30	56	40	41.5
0	96	51	69
0	91.5	50	63.5
25	143.5	70	83
25	113	59	78
70	152	100	90
70	159.5	100	90
125	158	100	93
125	148	100	86
210	156.5	100	97
210	152	100	77

ATTACHMENT 4
WCAP 9489

TABLE 3-2

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 2 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL FORGING 05 (AXIAL ORIENTATION)

Test Temp. (^{o}F)	Impact Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
-100	5.5	2	1
-100	7.5	2	1
- 50	21	15	12
- 25	29	15	20
- 25	28.5	15	18
0	34	30	26
25	49	45	40
25	53	45	39
56	68	54	47
56	63	54	46
56	67	54	47
100	71.5	90	54
100	74	92	61
100	72	94	55
140	93	100	71
140	92.5	100	64
210	95	100	67
210	97	100	67

ATTACHMENT 4
WCAP 9489

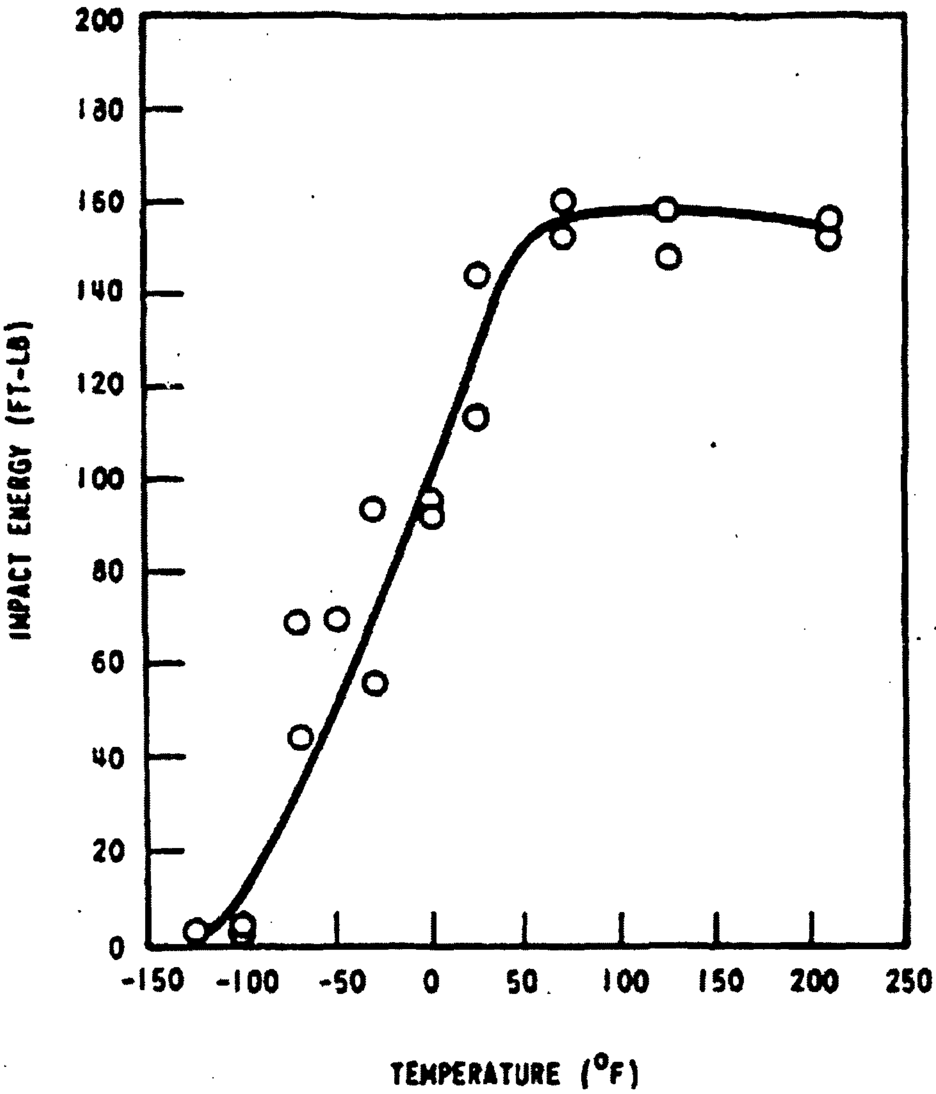

Figure 3-1. Preirradiation Charpy V-notch Impact Energy for the William B. McGuire Unit 2 Reactor Pressure Vessel Intermediate Shell Forging 05 (Tangential Orientation)

ATTACHMENT 4
WCAP 9489

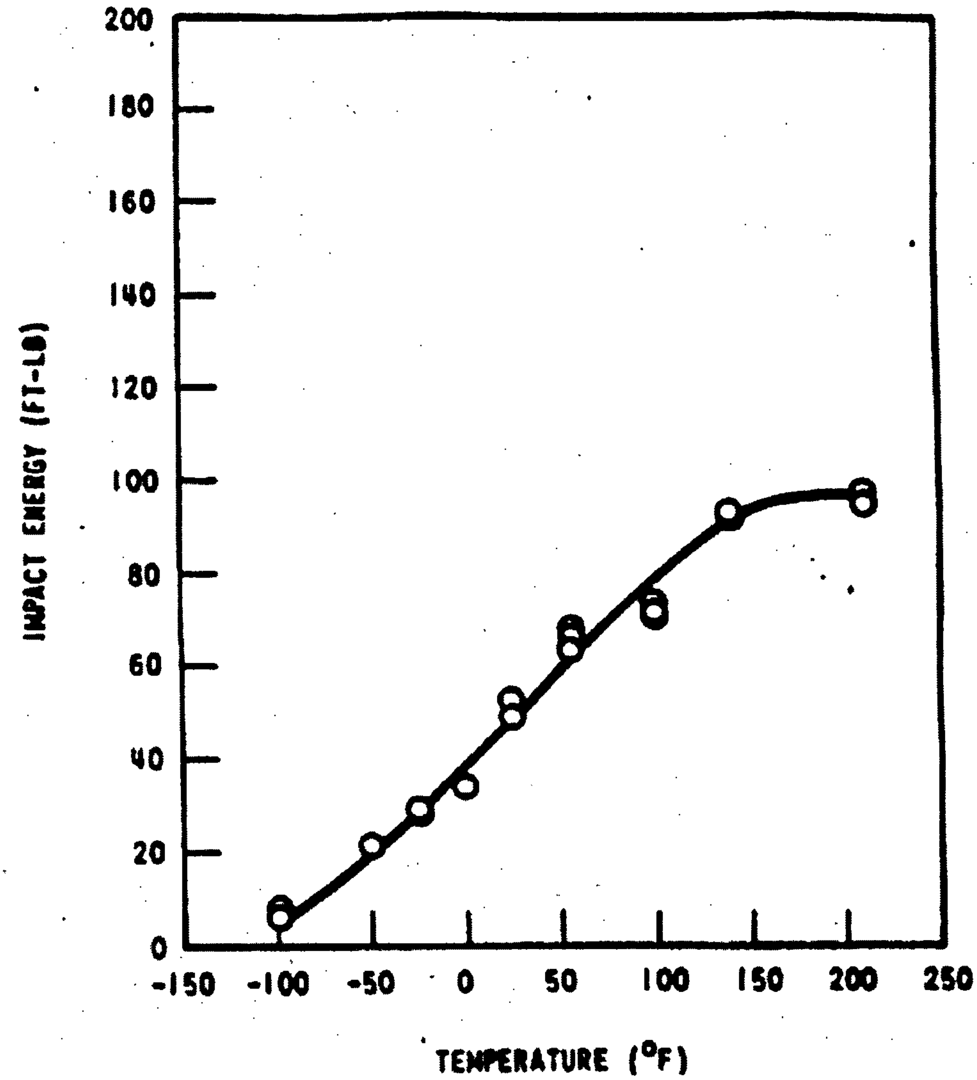

Figure 3-2. Preirradiation Charpy V-notch Impact Energy for the William B. McGuire Unit No. 2 Reactor Pressure Vessel Intermediate Shell Forging 05 (Axial Orientation)

ATTACHMENT 4
· WCAP 9489

TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE WILLIAM B. McGUIRE UNIT NO. 2 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

Test Temp. (°F)	Impact Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
-150	1	0	0
-150	1	0	0
- 75	20	30	15.5
- 75	16.5	29	9.5
- 35	34	46	31
- 35	52	54	40
- 16	50	65	40
- 16	57	65	46
- 16	59.5	65	47
25	93	81	67
25	83.5	77	64
71	112	75	82
71	110	79	80
125	125	98	91
125	124	99	92
210	140	100	96
210	132.5	99	96.5
275	144.5	100	98

·ATTACHMENT Y
WCAP 9489

TABLE 3-4

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE
WILLIAM B. McGUIRE UNIT NO. 2 REACTOR PRESSURE
VESSEL CORE REGION WELD HEAT-AFFECTED ZONE MATERIAL

Test Temp. (^{o}F)	Impact Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
-150	3	0	1.5
-125	6.5	0	0
-125	3.5	0	0
-100	29	18	16
- 75	73	56	48
- 75	34	47	26
- 50	43	20	28
- 16	66	69	43
- 16	67	64	48.5
- 16	78	77	51
32	82	91	56
32	67	87	58
100	107	100	69
100	81	100	56
150	101	100	70
210	116	100	66
210	121	100	67
250	99	100	69

ATTACHMENT 4
WCAP 9489

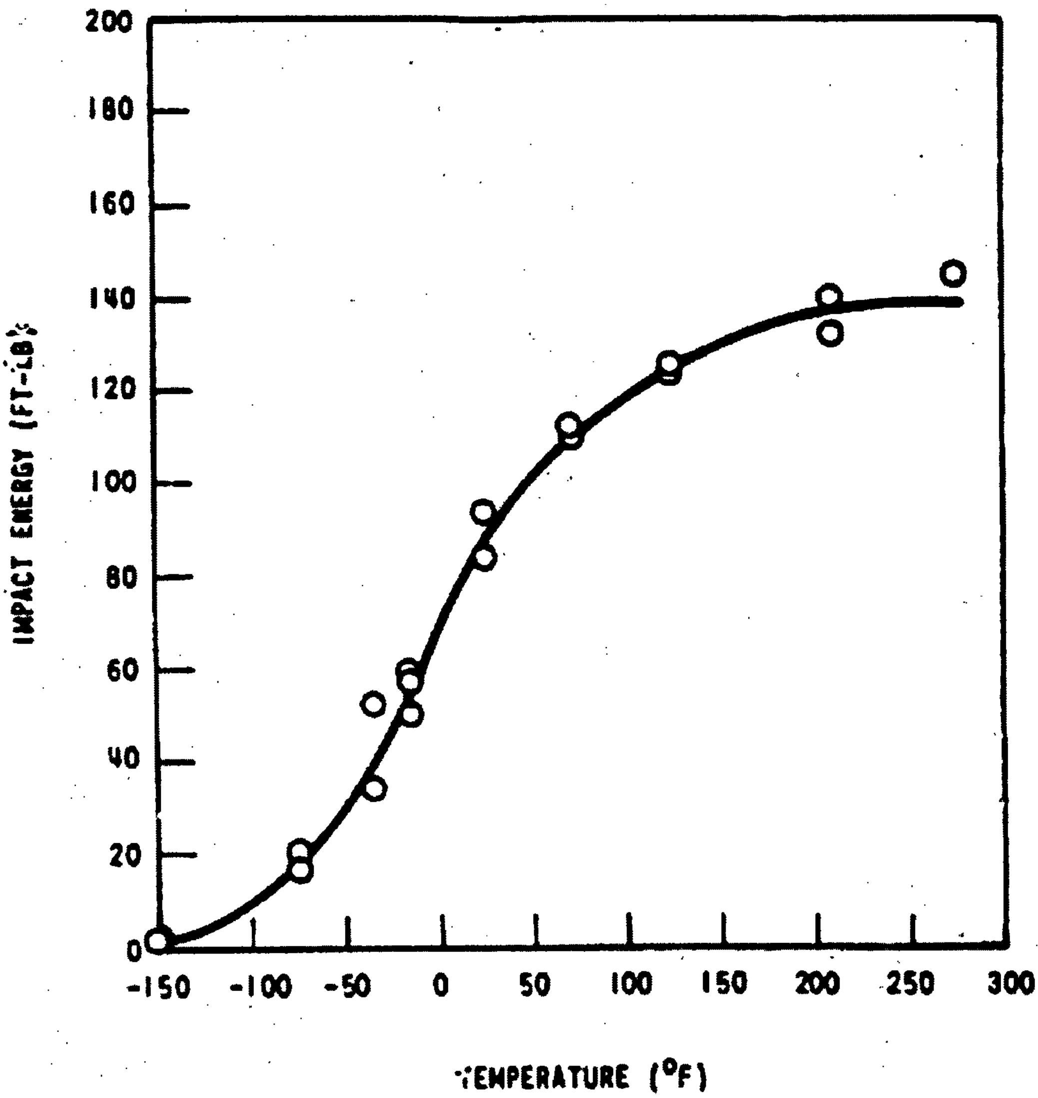

Figure 3-3. Preirradiation Charpy V-notch Impact Energy for the William B. McGuire Unit No. 2 Reactor Pressure Vessel Core Region Weld Metal

ATTACHMENT 4
WCAP 9489

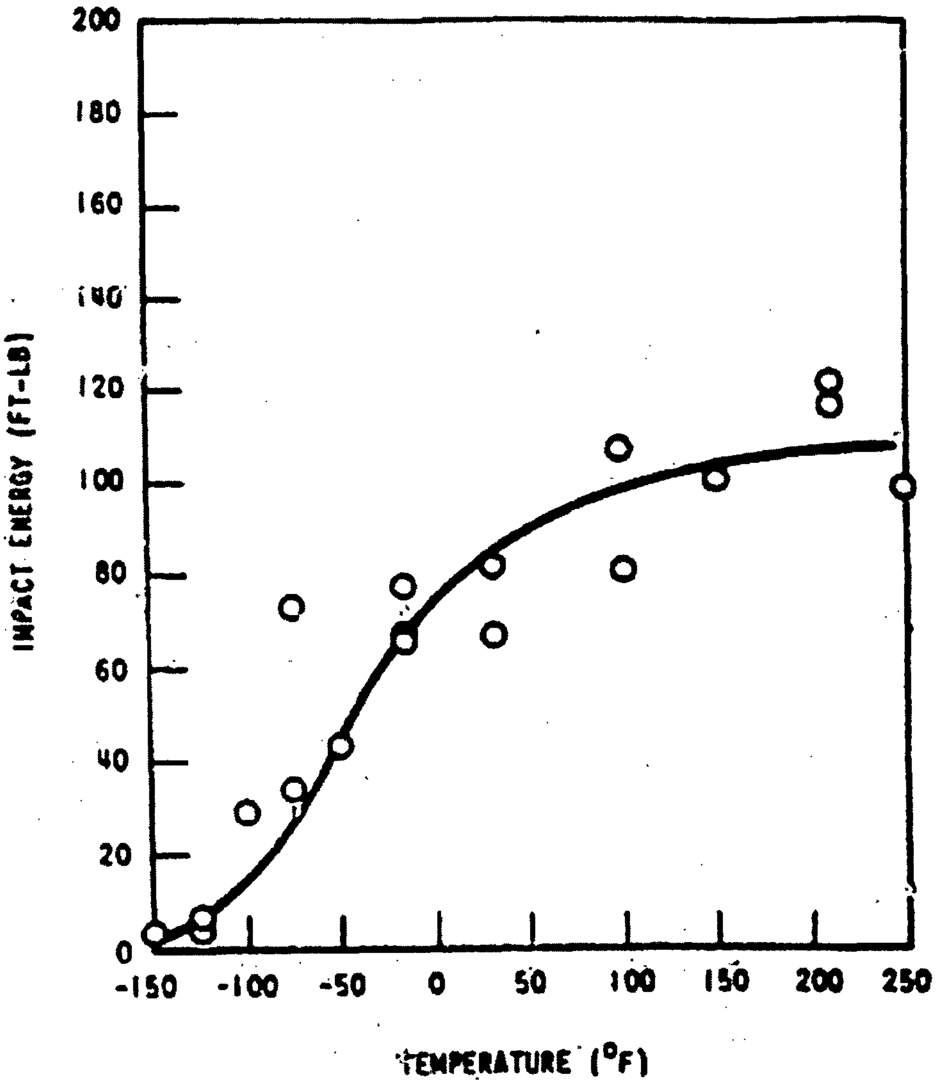

Figure 3-4. Preirradiation Charpy V-notch Impact Energy for the William B. McGuire Unit 2. Reactor Pressure Vessel Core Region Weld Heat-Affected-Zone Material

ATTACHMENT C
OCONEE NUCLEAR STATION'S RESPONSE TO GENERIC LETTER 92-01

THE B&W OWNERS GROUP

Arkansas Power & Light Company — ANO-1
Duke Power Company — Oconee 1, 2, 3
Florida Power Corporation — Crystal River 3
GPU Nuclear Corporation — TMI-1
Sacramento Municipal Utility District — Rancho Seco
Toledo Edison Company — Davis Besse
Tennessee Valley Authority — Bellefonte 1, 2
B&W Nuclear Technologies

Working Together to Economically Provide Reliable and Safe Electrical Power

Suite 525 • 1700 Rockville Pike • Rockville, MD 20852 • (301) 230-2100

June 17, 1992
OG-1036

US Nuclear Regulatory Commission
Washington, DC 20555

Attn: Document Control Desk

Subject: NRC Generic Letter 92-01 To Holders of Operating Licenses or Construction Permits For Nuclear Power Plants

Attachment: BAW-2166 B&W Owners Group Response To Generic Letter 92-01

Gentlemen:

The document BAW-2166, is hereby submitted on behalf of the B&W Owners Group Reactor Vessel Working Group. This report provides the information requested in NRC Generic Letter 92-01 for the following plants:

Arkansas Nuclear One Unit 1
Crystal River 3
Davis Besse
Ginna
Oconee 1, 2, 3
Point Beach 1 and 2
Surry 1, 2
Three Mile Island 1
Turkey Point 3, 4
Zion 1, 2

Utility Owners of these plants may reference this letter and the attachment BAW 2166, in their docketed responses.

Very truly yours,

DLHowell
for James H. Taylor
James H. Taylor
Manager
Licensing Services

JHT/DLH/mcl

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