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RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. LIGHT WATER REACTORS BRANCH 4

SUBJECT: FORWARDS RESPONSES TO FIRST ROUND QUESTIONS ON OL APPLICATION,W/RESPONSES TO PAST DUE QUESTIONS WHICH HAVE NOW BEEN COMPLETED.INFO WILL BE INCORPORATED INTO FSAR AMEND WITHIN FOUR MONTHS. (see reports)

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Washington Public Power Supply System
A JOINT OPERATING AGENCY

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March 21, 1979
G02-79-45

Docket No. 50-397

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

Attention: Mr. S. A. Varga, Chief
Branch No. 4
Division of Project Management

Subject: WPPSS NUCLEAR PROJECT NO. 2
RESPONSES TO ROUND ONE
QUESTIONS - MTEB, CPB, AAB, ETSB, RAB, GSB, HMB

Reference: Letter, S. A. Varga (NRC) to N. Strand (WPPSS), "First Round Questions on the WNP-2 OL Application - MTEB, CPB, AAB, ETSB, RAB, GSB, HMB," dated December 8, 1978

Dear Mr. Varga:

Attached please find sixty (60) copies of responses to the referenced questions. The few open items from the question set are being carried forward and will be submitted at the earliest possible date.

B001
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1/60
CHANGE:
PM 2 ENCL 2 LTR

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Also included in this question set are some responses to past due questions which have now been completed. All these responses will be incorporated formally into the FSAR in an amendment within four months.

Very truly yours,

D L Renberger

D. L. RENBERGER
Assistant Director
Technology

DLR:OKE:cph

Attachment: Responses to Round 1 Questions (60)

cc: I. Littman - WPPSS, N.Y. - wo/att
JJ Verderber - B&R, N.Y. - "
JJ Byrnes - B&R, N.Y. - "
RC Root - B&R, Site - "
HR Canter - B&R, N.Y. - "
D. Roe - BPA, - "
E. Chang - GE, San Jose - w/att (4)
FA MacLean - GE, San Jose - w/att (1)
J. Ellwanger - B&R, N.Y. - w/att (5)
NS Reynolds - Debevoise & Liberman - w/att (1)
WNP-2 Files - w/att (1)

Subject: WPPSS Nuclear Project No. 2
Response to Round One Questions
MTEB, CPB, AAB, ETSB, RAB, GSB, HMB

STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

D. L. RENBERGER, Being first duly sworn, deposes and says: That he is the Assistant Director, Technology, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that he is authorized to submit the foregoing on behalf of said applicant; that he has read the foregoing and knows the contents thereof; and believes the same to be true to the best of his knowledge.

DATED March 14, 1979.

D L Renberger

D. L. RENBERGER

On this day personally appeared before me D. L. RENBERGER to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes therein mentioned.

GIVEN under my hand and seal this 14th day of March, 1979.

Reba B. Helgesen

Notary Public in and for the State of Washington
Residing at Richland

Responses to:

Materials Engineering Branch Questions
(Materials Intergity Branch)
(121.1 - 121.9)

Q 121.1
(5.2.3)

Provide a sketch of the WNP-2 reactor vessel, including the basic dimension, showing all longitudinal and circumferential welds, and all forgings and/or plates. Welds should be identified by a shop control number (such as a procedure qualification number), the heat of the filler metal, the type and batch of the flux, and the welding process. Each forging and/or plate should be identified by a heat number and a material specification.

Response:

See the attached figures and tables for the requested information for the beltline welds.

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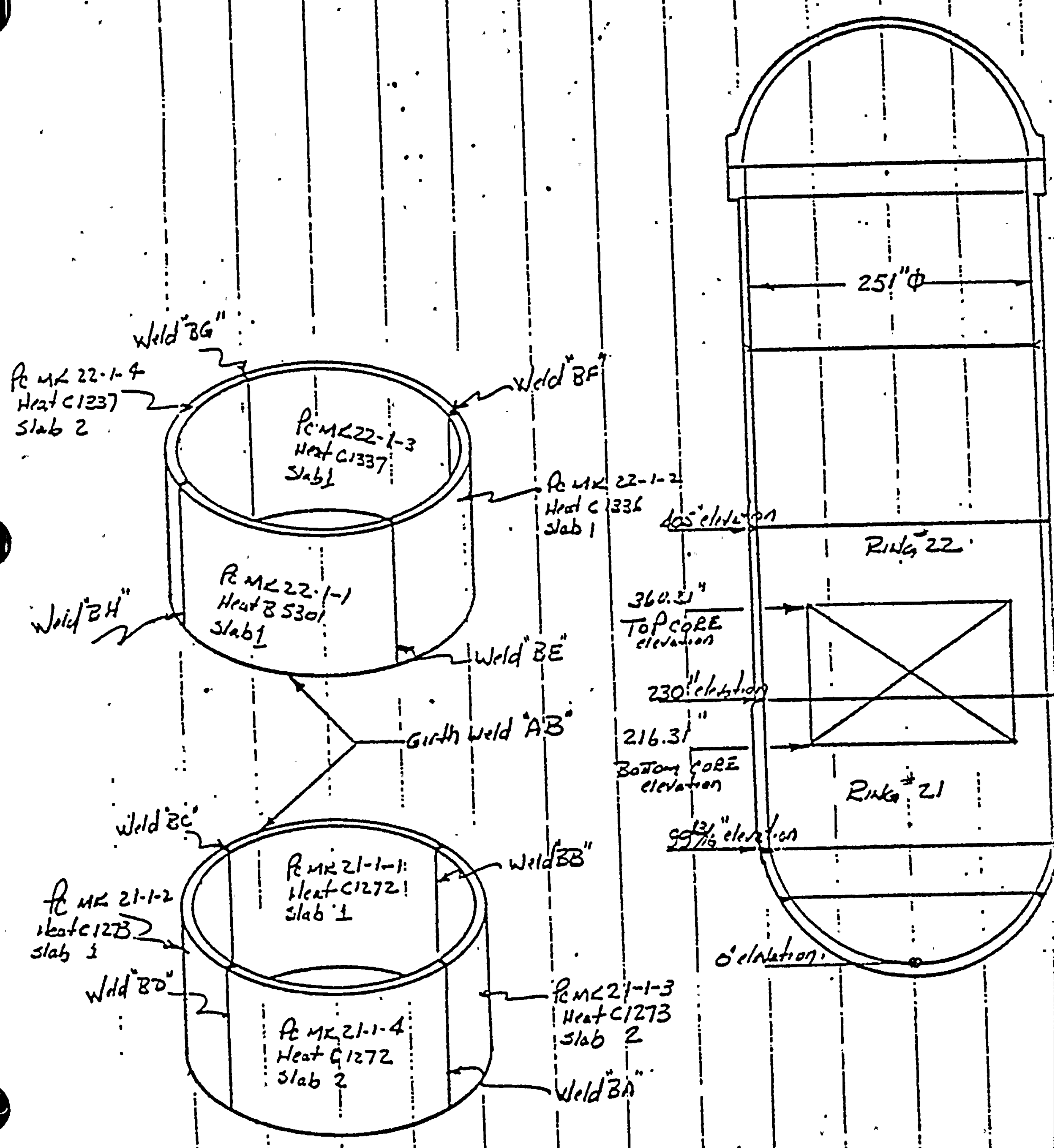
HANFORD 2

BELT LINE MATERIAL IDENTIFICATION.



Hanford-2

PLATE MATERIAL

	HEAT	SLAB
Ring # 21		
Pc MK 21-1-1	C1272	1
Pc MK 21-1-2	C1273	1
Pc MK 21-1-3	C1273	2
Pc MK 21-1-4	C1272	2
Ring # 2		
Pc MK 22-1-1	B5301	1
Pc MK 22-1-2	C1336	1
Pc MK 22-1-3	C1337	1
Pc MK 22-1-4	C1337	2

Hanford-2

Weld Material

Weld ID	Type	Heat	Lot
AB-Girthweld	E8018NM	4924L4871	A422B27AF
	RACO1NMM	5P6756	0342
	RACO1NMM	3P4955	0342
	E8018NM	04T931	A423B27AG
Ring 21			
BA	E8018NM	04P046	D217A27A
	E8018NM	07L669	K004A27A
	RACO1NMM	3PA966	1214
BB	E8018NM	04P046	D217A27A
	E8018NM	07L669	K004A27A
	E8018NM	C3L46C	J020A24A
	RACO1NMM	3P4966	1214
	E8018NM	08M365	G128A27A
BC	E8018NM	09L853	A111A27A
	E8018NM	C3L46C	J020A27A
	RACO1NMM	3P4966	1214
BD	E8018NM	C3L46C	J020A27A
	RACO1NMM	3P4966	1214
	E8018NM	C4P046	D217A27A
Ring 22			
BE	RACO1NMM	3P4966	1214

WELD ~~ID~~	TYPE	HEAT	LOT
BF	E898 NM	04P046	D217A27A
	E8018NM	05 P018	D211A27A
	RACOINM	3P4966	1214
BG	E8018 NM	624063	C223A27A
	E8018 NM	624039	D224A27A
	RACOINMM	3P4966	1214
BH	E8018NM	04P096	D217A27A
	E8018NM	624039	D205A27A
	RACOINMM	3P4966	1214

Q 121.2 Part a
(5.2.3)

For each of the ferritic materials of the pressure retaining components in the reactor coolant pressure boundary of the WNP-2 plant, supply the following information:

a. The unirradiated mechanical properties as required by the testing programs in Section III of the ASME Code and by Appendix G of 10 CFR Part 50. The test results should include the following material characteristics: (1) the Charpy V-notch energy; (2) the energy for the dropweight test; (3) the lateral expansion of the Charpy test specimens; (4) the tensile strength; (5) the upper shelf energy determined by the Charpy test; (6) the nil ductility temperature (T_{NDT}); and (7) the reference temperature (RT_{NDT}). If any of these properties have not been determined by a test method as required by Appendix G of 10 CFR Part 50; state the actual test procedure used and/or the method used to estimate the material properties, including a thorough technical justification of the procedure used.

Response:

The unirradiated mechanical properties of the pressure retaining components in the reactor coolant pressure boundary of the WNP-2 plant which require fracture toughness testing as defined by Section III of the ASME Code and Appendix G of 10 CFR 50 are provided as follows:

Reactor Pressure Vessel

A major condition necessary for full compliance to Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to Section III. Since this is not possible with components which were purchased to earlier Code requirements, an alternative approach was taken. For a discussion of this approach and the extent of compliance, see the response to Question 121.3 and the revised FSAR section 5.3.1.5. A specific response to this question is provided in the following tabulation of vessel beltline material characteristics.

Plate Material

	Charpy Impact ft-lb @+10°F	Charpy Expansion MLE	Drop Wt. NDT	RT_{NDT}
<u>Ring #21</u>				
PCMK 21-1-1 Ht# C1272-1	34,26,30/31,34,30	30,34,24/27,26,32	-10°F	28°F
PCMK 21-1-2 Ht# C1273-1	33,33,30/30,34,35	30,31,27/26,34,32	-20°F	20°F
PCMK 21-1-3 Ht# C1273-2	38,48,55/66,61,71	44,39,34/53,52,56	-30°F	4°F
PCMK 21-1-4 Ht# C1272-2	40,42,44/51,55,50	32,36,38/41,44,42	-30°F	0°F
<u>Ring #22</u>				
PCMK 22-1-1 Ht# B5301-1	64,62,66/52,52,55	56,56,56/45,44,44	-30°F	-20°F
PCMK 22-1-2 Ht# C1336-1	70,72,71/60,44,66	59,60,62/56,41,51	-30°F	-8°F
PCMK 22-1-3 Ht# C1337-1	71,76,74/70,72,55	61,60,60/63,61,52	-30°F	-20°F
PCMK 22-1-4 Ht# C1337-2	62,72,82/73,67,73	51,61,66/52,59,61	-50°F	-20°F

Weld Material

Type/Heat#	Charpy Impact Ft-lbs	Charpy Expansion MLE	Charpy Test Temp	Drop Wt NDT	RT_{NDT}
Girth Weld AB E8018NM/492L4871 Lot A422B27AF	78,82,105,93,81	55,60,72,64,60	-20°F	N/A	*
RAC01NMn/5P6756 Lot 0342	76,79,77,80,72	64,72,55,69,60	+10°F	N/A	*
RAC01NMM/3P4955 Lot 0342	49,63,47,49,64	39,48,36,43,57	+10°F	N/A	*
E8018NM/04T931 Lot A423B27AG	86,84,102,63,61	69,58,60,57,70	-20°F	N/A	*
Ring 21BA E8018NM/08P046 Lot D217A27A	34,36,37,39,40	23,28,24,20,24	-20°F	N/A	*
E8018NM/07L669 Lot K004A27A	50,50,54	44,44,46	+10°F	N/A	*
RAC01NMM/3P49U Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*
Ring 21BB E8018NM/04P046 Lot D217A27A	34,36,37,39,40	23,28,24,20,24	-20°F	N/A	*
E8018NM/07L669 Lot K004A27A	50,50,54	44,44,46	+10°F	N/A	*
E8018NM/C3L46C Lot J020827A	35,39,40	34,39,39	+10°F	N/A	-20°F Highest
RAC01NMM/3P4966 Lot 1214	29,38,38,82,84	68,64,63,81,72	+10°F	N/A	*
E8018NM/08M365 Lot G128A27A	49,50,51	38,40,43	+10°F	N/A	*
Ring 21BC E8018NM/19L853 Lot A111A27A	78,78,79	60,62,62	+10°F	N/A	*
E8018NM/C3L46C Lot J020A27A	35,39,40	34,39,39	+10°F	N/A	-20°F Highest
RAC01NMM/3P4966 Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*

Weld Material

Type/Heat#	Charpy Impact Ft-lbs	Charpy Expansion MLE	Charpy Test Temp	Drop Wt NDT	RT_{NDT}
Ring 21 BD E8018NM/C3L46C Lot J020A27A	35,39,40	34,39,39	+10°F	N/A	-20°F Highest
RAC01NMM/3P4966 Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*
E8018NM/C91046 Lot D217A27A	34,36,37,39,40	23,28,24,20,24	-20°F	N/A	*
Ring 22 BE RAC01NMM/3P4966 Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*
Ring 22 BF E8018NM/04P046 Lot D217A27A	34,36,37,39,40	23,28,24,20,24	-20°F	N/A	*
E8018NM/05P018 Lot D211A27A	29,30,31,36,38	26,26,31,33,35	-20°F	N/A	*
RAC01NMM/3P4966 Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*
Ring 22 BG E8018NM/624063 Lot C228A27A	27,40,51,57,70	33,34,41,47,55	-20°F	N/A	*
E8018NM/624039 Lot D224A27A	28,33,34,36,42	29,32,33,34,42	-20°F	N/A	*
RAC01NMM/3P4966 Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*
Ring 22 BH E8018NM/04P046 Lot D217A27A	34,36,37,39,40	23,28,24,20,40	-20°F	N/A	*
E8018NM/624039 Lot D205A27A	41,44,49,54,58	32,36,40,41,45	-20°F	N/A	*
RAC01NMM/3P4966 Lot 1214	39,38,38,82,84	68,64,63,81,72	+10°F	N/A	*

* only the hightest start RT_{NDT} was calculated.

Piping, Pumps, and Valves

There are no ferritic pumps in the reactor coolant pressure boundary. All the piping and valves that are part of the reactor coolant pressure boundary are ASME Section III Class 1 components. As such, the material for all these components is impact tested in accordance with the requirements of Subsection NB of the Code and meets the brittle fracture toughness requirements of the code.

As stated in FSAR Section 5.2.3.3.1, the Main Steam Safety/Relief Valves were exempted from fracture toughness requirements because Section III of the 1971 ASME Boiler and Pressure Vessel Code did not require impact testing on valves with inlet connections of 6 inches or less nominal pipe size.

Also, as stated in FSAR Section 5.2.3.3.1, the Main Steam Isolation Valves were also exempted because the Code existing at the time of the purchase order, April 1971, did not require brittle fracture testing on ferritic pressure boundary components when the system temperature was in excess of 250°F at 20% of the design pressure.

Where required, the test results for piping and valves in the reactor coolant pressure boundary are available for inspection at the WNP-2 site.

Q 121.2 part b
(5.2-3)

For each of the frritic materials of the pressure retaining components in the reactor coolant pressure boundary of the WNP-2 plant, supply the following information:

b. Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves at the beginning-of-life.

Response:

The reactor vessel operating limits are defined by Figure 5.3-4 in the amended WNP-2 FSAR (a copy of Figure 5.3-4 is attached). Curves A, B, and C on Figure 5.3-4 give the limits for beginning-of-life conditions for hydrotest, non-nuclear heating, and nuclear heating. The minimum boltup temperature of $80^{o}F$ is based on an RT_{NDT} of $20^{o}F$ for a shell plate which connects to the lower flange (Heat and Slab No. C-1307-2). Above $80^{o}F$ the core beltline plate (Heat and Slab No. C-1272-1), which has an initial RT_{NDT} of $28^{o}F$, is most limiting for hydrotest (Curve A). The feedwater nozzles, which have an RT_{NDT} of $-14^{o}F$, are more restrictive than the core beltline at lower pressures during non-nuclear and nuclear heating (Curves B & C).

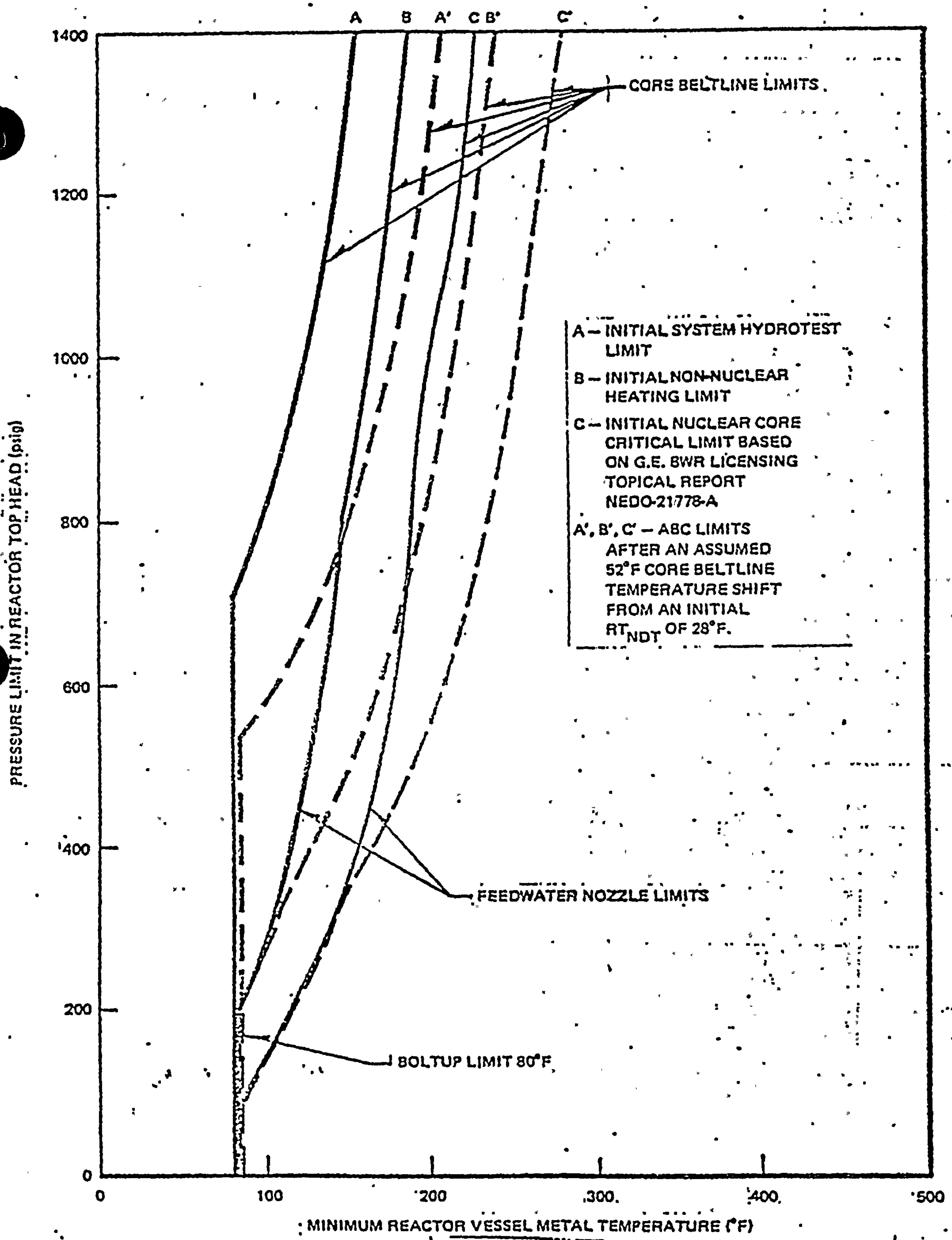


Figure 5.3-4. WPPSS Hanford Unit 2
Minimum Temperature Required versus Reactor Pressure

Q 121.2 b-7

Q 121.2 Part C
(5.2-3)

For each reactor vessel beltline weld, plate or forging provide the following additional information:

c. The chemcial composition (particularly the copper, phosphorus and sulphur content) and the maximum end-of-life neutron fluence.

Response:

The response to this question is provided in the attached tabulations.

WNP-2

VESSEL BELTLINE PLATE

PLATE	P	Cu	C	Mn	Si	S	Ni	Mo	V
C1272-1	.013	.15	.23	1.31	.26	.02	.55 .60	.53	—
C1272-2	.013	.15	.23	1.31	.26	.02	.60	.55	—
C1273-1	.014	.14	.23	1.28	.23	.018	.60	.57	—
C1273-2	.014	.14	.23	1.28	.23	.018	.60	.57	—
B5301-1	.017	.14	.20	1.34	.23	.014	.50	.52	—
C1336-1	.017	.13	.21	1.36	.22	.013	.50	.49	—
C1337-1	.018	.15	.22	1.32	.21	.013	.51	.50	—
C1337-2	.018	.15	.22	1.32	.21	.013	.51	.50	—

Peak EOL Fluence = 1.4×10^{18} n/cm^2 ($\frac{1}{4}$T wall)

Q121.2 C-9

VESSEL BELTLINE WELD

WELD SEAM	Cu	C	Mn	Si	S	Ni	Mo	V	P
492L4871	.03	.07	1.17	.32	.02	.98	.51	.02	.02
5P6756	.09	.063	1.27	.57	.011	.93	.45	.006	.01
	.09	.078	1.24	.53	.012	.92	.46	.006	.01
3P4955	.025	.035	1.33	.56	.011	.90	.52	.006	.016
	.025	.054	1.28	.55	.010	.95	.54	.007	.016
04T931	.03	.05	1.03	.28	.024	1.00	.53	.01	.02
04P046	.06	.044	1.04	.40	.021	.90	.58	.02	.009
07L669	.03	.05	1.24	.48	.016	1.02	.54	–	.014
3P4966	.03	.059	1.35	.38	.013	.80	.50	.005	.011
	.03	.067	1.39	.38	.014	.90	.53	.008	.011
03L46C	.02	.063	.96	.32	.017	.87	.53	–	.019
08M365	.02	.057	1.23	.47	.023	1.10	.57	–	.02
09L853	.03	.052	1.23	.46	.023	.86	.51	–	.018
05P018	.09	.057	1.21	.44	.021	.90	.53	.01	.008
624063	.03	.041	1.12	.41	.018	1.00	.54	.01	.009
624039	.07	.060	1.11	.45	.025	1.01	.57	.02	.015

Q121.2 C-10

Q 121.2 Part d
(5.2-3)

For each reactor vessel beltline weld, plate or forging provide the following additional information:

d. The relationship used to predict the shift in RT_{NDT} and the percent decrease in the upper shelf energy as a function to neutron fluence.

Response:

This information is contained in Subsection 5.3.1.4.1.7 of the WNP-2 FSAR.

Q 121.2 Part e
(5.2-3)

For each reactor vessel beltline weld, plate or forging provide the following additional information:

e. Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves at the end-of-life.

Response:

The reactor coolant pressure boundary materials that were limiting at the beginning-of-life are also limiting at the end-of-life. These materials are identified in the response to question 121.2b. The end-of-life operating limits assumed on Figure 5.3-4, Curves A', B', and C', correspond to a 52^{o}F core beltline shift due to a fluence of 1.4 x 10^{18} n/sq.cm and a Cu content of 0.15%.

QUESTION 121.3 (5.3.1)

In Section 5.3.1.5 of the FSAR, you state that compliance with Appendix G of 10 CFR Part 50 and Appendix G of Section III of the ASME Code was not possible for components purchased prior to the issuance of the Summer 1972 Addenda of the ASME Code without replacement of large amounts of material, reworking of fabricated components and the revision of most of the design analyses for the affected components. However, the details provided in the FSAR are insufficient to identify the areas of noncompliance with Appendix G of 10 CFR Part 50. State specifically which components do not comply with the regulations and also which sections of Appendix G are not satisfied.

We require that you provide the technical bases for the proposed alternate methods used to satisfy the requirements of those sections of Appendix G of 10 CFR Part 50 where strict compliance was not achieved. These bases should include technical justification which demonstrate that the proposed alternatives provide acceptable safety margins relative to the Appendix G requirements.

RESPONSE

The concerns identified in Question 121.3 have been addressed in the FSAR text update to Section 5.3.1.5.*

*See the attached draft FSAR Sections

5.3.1.4.1.3 Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Reactor pressure vessel specifications require that all low alloy steel be produced to fine grain practice. The requirements of the regulatory guide are not applicable to BWR vessels.

5.3.1.4.1.4 Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel

Controls to avoid severe sensitization are discussed in 5.2.3.4.1.1.

5.3.1.4.1.5 Regulatory Guide 1.50, Control of Preheat Temperature for Welding Low-Alloy Steel

Preheat controls are discussed in 5.2.3.3.2.1.

5.3.1.4.1.6 Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in 5.2.3.3.2.4.

5.3.1.4.1.7 Regulatory Guide 1.99, Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper shelf energy were made in accordance with the requirements of Regulatory Guide 1.99.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance With 10 CFR 50 Appendix G

A major condition necessary for full compliance to Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to Section III. This is not possible with components which were purchased to earlier Code requirements without the replacement of large amounts of material, reworking of fabricated components, and the revision of most all of the design analysis for these components. On the basis of the last paragraph on Page 19013 of the July 17, 1973 Federal Register, the following is considered an appropriate method of compliance.

Replace ~~Delete~~ entire Section 5.3.1.5 ~~and replace~~
with new write up attached

5.3.1.5.1.1 Method of Compliance

Basically, the method of compliance with Appendix G for this vessel is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits should assure that a margin of safety against a non-ductile failure of this vessel will be very nearly the same as a vessel built to the Summer 1972 Addenda. Also, it should be noted that this vessel must also be operated within the requirements of the original applicable Code.

5.3.1.5.1.2 Acceptable Fracture Energy Levels

Operating limits on reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using as a guide Appendix "G" Summer 1972 Addenda of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition.

These operating limits will assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. In addition the specific additional margins required by 10 CFR 50 Appendix G paragraph G-II.A.2.E are included in the operating limits for core operations.

For the purpose of setting these operating limits the reference temperature, RT_{NDT}, was determined from the impact test data taken in accordance with requirements of the Code to which this vessel is designed and manufactured. The dropweight NDT temperature was used as the reference temperature.

The highest reference temperature of any part of the reactor pressure vessel pressure boundary material was used as the reference temperature for calculating one set of operating temperature and pressure limits for the shell remote from the core beltline region. A second set of temperature and pressure limits for the core beltline region was calculated based on the core beltline region material reference temperature.

The requirements of the Code to which the vessel was designed and manufactured results in a third set of vessel shell temperature and pressure limits; namely, NDTT +60F or CVN +60F at pressure greater than 20% of preoperational system hydrostatic test pressure. The more conservative of the above three limits will be used to set pressure and temperature limits for the vessel shell.

5.3.1.5.1.3 Operating Limits During Heatup, Cooldown and Core Operation

Since 100°F/hour is the maximum average normal heatup or cooldown rate for which the reactor vessel is designed, a conservative fracture toughness analysis was done for this assumed rate.

The maximum temperature gradient through the wall corresponding to this rate was considered. The results of this analysis are a set of operating limits for non-nuclear heatup or cooldown following nuclear shutdown, and another set for operating limits for operation whenever the core is critical (except for low level physics tests).

5.3.1.5.1.4 Temperature Limits for ISI Hydrostatic or Leak Pressure Tests

The fracture toughness analysis for pressure tests resulted in the curves shown in Figure 5.3-4a, b and c of minimum vessel shell and head temperature versus vessel pressure (measured in vessel top head). The dashed line curve, beltline region, is based on an assumed initial RT_{NDT} of +10°F, the predicted shift in the RT_{NDT} from Figure 5.3-5 based on neutron fluence at 1/4 of vessel wall thickness must be added to the beltline curve to account for the effect of fast neutrons on the beltline material properties. The curve for areas remote from the beltline (upper curve) is based on an assumed RT_{NDT} of +40°F. The controlling minimum temperature for a desired pressure is then selected as the greater of the solid curve or the dashed curve plus the shift.

5.3.1.5.1.5 Temperature Limits for Boltup

Minimum closure flange and closure stud temperature of 70°F (NDTT + 60°F) are required whenever the closure studs are under preload or are being tensioned.

5.3.1.5.1.6 Reactor Vessel Annealing

In-place annealing of the reactor vessel because of radiation embrittlement is not anticipated to be necessary because the predicted value in transition of adjusted reference temperature will not exceed 200°F - see 10 CRF 50, Appendix G, Paragraph IV.C.

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5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with Code Requirements

The ferritic pressure boundary material of the reactor pressure vessels was qualified by impact testing in accordance with the 1971 edition of Section III ASME Code and Summer 1971 Addenda. From an operational standpoint, the minimum temperature limits for pressurization defined by the Summer 1972 Addenda, Appendix G, Protection Against Nonductile Failure, are used as the basis for compliance with ASME Code Section III.

5.3.1.5.2 Compliance with 10 CFR 50 Appendix G

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A major condition necessary for full compliance to Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to Section III. This is not possible with components which were purchased to earlier Code requirements. For the extent of the compliance, see Table 5.3-1a.

Ferritic material complying with 10 CFR 50 Appendix G must have both drop weight tests and Charpy V-notch (CVN) tests with the CVN specimens oriented transverse to the maximum material working direction to establish the RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and a lateral expansion criteria. The maximum acceptable RT_{NDT} must be determined in accordance with the analytical procedures of ASME Code Section III, Appendix G. Appendix G of 10 CFR 50 requires a minimum of 75 ft-lb upper shelf CVN energy for beltline material. It also requires at least 45 ft-lb CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature.

By comparison, material for the WNP-2 reactor vessels was qualified by either drop weight tests or longitudinally oriented CVN tests (both not required), confirming that the material nil-ductility transition temperature (NDTT) is at least 60°F below the lowest service temperature. When the CVN test was applied, a 30 ft-lb energy level was used in defining the NDTT. There was no upper shelf CVN energy requirement on the beltline material. The bolting material was qualified to a 30 ft-lb CVN energy requirement at 60°F below the minimum preload temperature.

From the previous comparison it can be seen that the fracture toughness testing performed on the WNP-2 reactor vessel material cannot be shown to comply with 10 CFR 50 Appendix G. However, to determine operating limits in accordance with 10 CFR 50 Appendix G, estimates of the beltline material RT_{NDT} and the highest RT_{NDT} of all other material were made, ~~as explained~~ and are dis in Subsection 5.3.1.5.2.2. The method for developing these operating limits is also described therein.

On the basis of the last paragraph on page 19013 of the July 17, 1973, Federal Register, the following is considered an appropriate method of compliance.

5.3.1.5.2.1 Intent of Proposed Approach

The intent of the proposed special method of compliance with Appendix G for this vessel is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits assure that a margin of safety against a nonductile failure of this vessel is very nearly the same as that for a vessel built to the Summer 1972 Addenda.

The specific temperature limits for operation when the core is critical are based on a proposed modification to 10 CFR 50, Appendix G, Paragraph IV, A.2.C. The proposed modification and the justification for it is given in GE Licensing Topical Report NEDO-21778-A.

5.3.1.5.2.2 Operating Limits Based on Fracture Toughness

Operating limits which define minimum reactor vessel metal temperatures vs reactor pressure during normal heatup and cooldown and, during in-service hydrostatic testing, were established using the methods of Appendix G of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition (Appendix G first appeared in the Summer 1972 Addenda). The results are shown in Figure 5.3-4.

All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles were evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} + 60°F. The maximum through-wall temperature gradient from continuous heating or cooling at 100°F per hour was considered. The safety factors applied were as specified in ASME Code Appendix G and GE Licensing Topical Report NEDO-21778-A.

For the purpose of setting these operating limits the reference temperature, RT_{NDT}, is determined from the toughness test data taken in accordance with requirements of the Code to which this vessel is designed and manufactured. This toughness test data, Charpy V-notch (CVN) and/or drop-weight nil ductility transition temperature (NDT) is analyzed to permit compliance with the intent of 10 CFR 50 Appendix G. Because all toughness testing needed for strict compliance with Appendix G was not required at the time of vessel procurement some toughness results are not available. For example, longitudinal CVN's, instead of transverse, were tested, usually at a single test temperature of +10°F or +40°F, for absorbed energy. Also, at the time either CVN or NDT testing was permitted; therefore, in many cases both tests were not performed as is currently required. To substitute for this absence of certain data, toughness property correlations were derived for the vessel materials in order to operate upon the available data to give a conservative estimate of RT_{NDT}, compliant with the intent of Appendix G criteria.

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These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of WRC Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels", and from toughness data from the WNP-2 vessel and other reactors. In the case of vessel plate material (SA-533 Grade B, Class 1), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lb temperature minus 60°F. NDT values are available for WNP-2 vessel shell plates. The transverse CVN 50 ft-lb transition temperature is estimated from longitudinal CVN data in the following manner. The lowest longitudinal CVN ft-lb value is adjusted to derive a longitudinal CVN 50 ft-lb transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equals or exceeds 50 ft-lb, the test temperature is used. Once the longitudinal 50 ft-lb temperature is derived, an additional 30°F is added to account for orientation effects and to estimate the transverse CVN 50 ft-lb temperature minus 60°F, estimated in the preceding manner.

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For forgings (SA-508 Class 2), the predicted limiting property is the same as for vessel plates. CVN and NDT values are available for the vessel flange, closure head flange, and feedwater nozzle materials for WNP-2. RT_{NDT} is estimated in the same way as for vessel plate.

For the vessel weld metal the predicted limiting property is the CVN 50 ft-lb transition temperature minus 60°F, as the NDT values are -50°F or lower for these materials. This temperature is derived in the same way as for the vessel plate material, except the 30°F addition for orientation effects is omitted since there is no principal working direction. When NDT values are available, they are also considered and the RT_{NDT} is taken as the higher of NDT or the 50 ft-lb temperature minus 60°F. When NDT is not available, the RT_{NDT} shall not be less than -50°F, since lower values are not supported by the correlation data.

For vessel weld heat affected zone (HAZ) material the RT_{NDT} is assumed the same as for the base material as ASME Code weld procedure qualification test requirements and post weld heat treatment indicates this assumption is valid.

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Closure bolting material (SA-540 Grade B24) toughness test requirements for WNP-2 were for 30 ft-lb at 60°F below the bolt-up temperature. Current Code requirements are for 45 ft-lb and 25 mils lateral expansion (MLE) at the preload or lowest service temperature, including boltup. All WNP-2 closure stud materials meet current requirements at +10°F.

Using the above general approach, an initial RT_{NDT} of 28°F was established for the core beltline region.

The effect of the main closure flange discontinuity was considered by adding 60°F to the RT_{NDT} to establish the minimum temperature for boltup and pressurization. The minimum bolt-up temperature of 80°F for Hanford Unit 2, which is shown on Figure 5.3-4, is based on an initial RT_{NDT} of +20°F for the closure flange forgings.

The effect of the feedwater nozzle discontinuities were considered by adjusting the results of a BWR/6 reactor discontinuity analysis to the reactor. The adjustment was made by increasing the minimum temperatures required by the difference between the Hanford-2 and BWR/6 feedwater nozzle forging RT_{NDT}'s. The feedwater nozzle adjustment was based on an RT_{NDT} of -14°F.

The reactor vessel closure studs have a minimum Charpy impact energy of 48 ft-lbs and a 26-mil lateral expansion at 10°F. *The lowest service temperature for the closure studs is 10°F.*

5.3.1.5.2.3 Temperature Limits for Boltup

A minimum temperature of 10°F is required for the closure studs. A sufficient number of studs may be tensioned at 70°F to seal the closure flange O-rings for the purpose of raising reactor water level above the closure flanges in order to assist in warming them. The flanges and adjacent shell are required to be warmed to a minimum temperature of 80°F before they are stressed by the full intended bolt preload. The fully preloaded bolt-up limits are shown on Figure 5.3-4.

5.3.1.5.2.4 Temperature Limits for Preoperational System Hydrostatic Tests and ISI Hydrostatic or Leak Pressure Tests

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~~Based on 10 CFR 50 Appendix G IV.A.2.d, which allows a reduced safety factor for tests prior to fuel loading, the preoperational system hydrostatic~~ test at 1563 psig may be performed at a minimum temperature of 128°F *which is established by core beltline plate material.* The fracture toughness analysis for *inservice inspection or leak* tests resulted in the curves labeled A shown in Figure 5.3-4. The curves labeled "core beltline" are based on an initial RT_{NDT} of 28°F.

The predicted shift in the RT_{NDT} from Figure 5.3-5 (based on the neutron fluence at 1/4 of the vessel wall thickness) must be added to the beltline curve to account for the effect of fast neutrons*, See Figure 5.3-4.*

5.3.1.5.2.5 Operating Limits During Heatup, Cooldown and Core Operation

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/hour. The temperature gradients and thermal stress effects corresponding to this rate was included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown as curves labeled B on Figure 5.3-4. Curves labeled C on these figures apply whenever the core is critical. The basis for Curves C is described in GE BWR Licensing Topical Report NEDO-21778.

Add sentence here

5.3.1.5.2.6 Reactor Vessel Annealing

Inplace annealing of the reactor vessel because of radiation embrittlement is unnecessary because the predicted value in transition of adjusted reference temperature will not exceed 200°F (see 10 CFR 50, Appendix G, Paragraph IV.C).

The maximum hydrostatic test pressure is 1563 + 6% psig (1613 psig) with a minimum temperature of 131°F.

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5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

Materials for the program are selected to represent materials used in the reactor beltline region. The specimens are manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld metal, and the transition zone between base metal and weld. The plate and weld are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Each in-reactor surveillance capsule contains 36 Charpy-V-Notch specimens. The capsule loading consists of 12 specimens each of base material, weld material, and weld heat affected zone material. A set of out-of-reactor baseline Charpy-V-Notch specimens is provided with the surveillance test specimens.

Charpy impact specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issue of the Summer 1972 Addenda and ~~ASTM E~~ 185-73. Based on GE experience the amount of shift measured by these irradiated longitudinal test specimens will be essentially the same as shift in an equivalent transverse specimen.

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~~The program includes three capsules in the reactor. Since the predicted increase in transition temperature of the reactor vessel steel is less than 100°F and the calculated peak neutron fluence is less than 5×10^{18} n/cm^2, the use of three capsules meets the requirements of ASTM-E-185-73. The withdrawal schedule is:~~

add new paragraphs (attached next page)

The program includes three sets of specimens in the reactor.

The withdrawal schedule of the three sets of specimens in the reactor is planned as follows:

a. The first set will be withdrawn when its exposure corresponds to the calculated exposure of the reactor vessel wall at 25% of the reactor design life.

b. The second set will be withdrawn when its exposure corresponds to the calculated exposure of the reactor vessel wall at 75% of the reactor design life.

c. The third set will be a spare to be withdrawn based on previously developed data.

For the extent of compliance to 10 CFR 50 Appendix H, see Table 5.3-1b.

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a. First capsule - 10 years (1/4 service life)

b. Second capsule - 30 years (3/4 service life)

c. Third capsule - Standby

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in 4.1.4.5 and 4.3.2.8.

Fluxes and Fluences at Surveillance Sample Locations (Nominal Values)

Fluxes (n/cm^2-sec)

>3 Mev = 4×10^8

1-3 Mev = 2.7×10^8

0.1-1 Mev = 2.4×10^8

Fluence (n/cm^2) for 40 years

>1 Mev = 6.7×10^{17}

Fluxes and Fluences at 1/4 Depth of Vessel Wall (Nominal Values)

Fluxes (n/cm^2-sec)

>3 Mev = 2.2×10^8

1-3 Mev = 2.8×10^8

0.1-1 Mev = 5.7×10^8

Fluence (n/cm^2) for 40 years

>1 Mev = 4.9×10^{17}

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The listed nominal flux and fluence data require the following qualification:

a. The data was calculated adjacent to the maximum axial power plane in the core. No azimuthal angle variation was included. Neutron fluxes and fluences could be from 0.5 to 1.4 times the nominal flux depending on the angular location.

b. The peak location in the axial plane is below the core midplane. Samples at other locations will be in lower flux fields.

c. All calculations were done using nominal dimensions. There can be variations in the vessel diameter.

d. The overall accuracy of the analysis method is estimated to be no better than a factor of 2.

e. As a result of the above qualifications, the calculated flux could range from 0.25 to 2.8 times the nominal values.

f. Fluences are based on 80% capacity factor for 40 years.

All predictions of radiation damage to the reactor vessel beltline material and surveillance samples were made using peak flux and fluence values which were 2.8 times the nominal values.

5.3.1.6.3 Expected Effects of Radiation on Vessel Wall Materials

The steel plates used in the beltline region of the vessel contain a maximum of 0.15 percent copper and phosphorus is below 0.018 percent. The evaluation of radiation effects was made in accordance with the requirements of Regulatory Guide 1.99 — using peak fluence values.

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Effects of Radiation at Surveillance Sample Location

Time (Years)	Fluence (n/cm²) Nominal	Peak	Trans. Temp. Change (°F)	Change In Upper Shelf Energy (%)
40	6.7×10^{17}	1.9×10^{18}	70	-16
30	5×10^{17}	1.4×10^{18}	61	-15
10	1.7×10^{17}	4.8×10^{17}	20	-12

Effects of Radiation at 1/4T Depth of Vessel Beltline Region

Time (Years)	Fluence (n/cm²) Nominal	Peak	Trans. Temp. Change (°F)	Change in Upper Shelf Energy (%)
40	4.9×10^{17}	1.37×10^{18}	60	-15
30	3.7×10^{17}	1.04×10^{18}	51	-14
10	1.2×10^{17}	3.4×10^{18}	16	-11

Regulatory Guide 1.99 is not valid below a 50°F transition temperature change. Values below 50°F were obtained from General Electric Co., data per the curve shown in Figure 4-1, "BWR Radiation Effects Design Curve", in NEDO-20651.

5.3.1.6.2 Positioning of Surveillance Capsules and Method of Attachment

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding as shown in Figure 5.3-3. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. These brackets are designed, fabricated and analyzed to the requirements of Section III ASME Code. A positive spring loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

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5.3.1.6.~~8~~ 3 Time and Number of Dosimetry Measurements

GE provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence-to-thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output.

5.3.1.7 Reactor Vessel Fasteners

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The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a thread hole in its vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by a sequential tensioning using hydraulic tensioners. The design and analysis of this area of the vessel is in full compliance with all Section III Class I Code requirements. The material for studs, nuts and washers is SA-540 Grade B23 or B24. The maximum reported ultimate tensile stress for the bolting material was 167,000 psi which is less than the 170,000 psi limitation in Regulatory Guide 1.65. Also the Charpy impact test recommendations of Paragraph IV.A.4 of Appendix G to 10 CFR 50 were not specified in the vessel order since the order was placed prior to issurance of Appendix G to 10 CFR 50. However, impact data from the certified materials report shows that all bolting material has met the Appendix G impact properties. For example, the lowest reported C_V energy was ~~48~~ ft-lbs at 10°F versus the 48 required 45 ft-lbs at 70°F and the lowest reported C_V expansion was 26 mils at 10°F versus the required 25 mils at 70°F.

Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed. Studs, nuts, and washers are ultrasonically examined in accordance with Section III, NB2585 and the following additional requirements:

5.3.1.6. 4 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in 4.1.4.5 and 4.3.2.8.

a. Examination is performed after heat treatment and prior to machining threads.

b. Straight beam examination is performed on 100 percent of each stud. Reference standard for the radial scan is a 1/2 inch diameter flat bottom hole having a depth equal to 10 percent of the material thickness. For the end scan the reference standard is a 1/2 inch flat bottom hole having a depth of 1/2 inch.

c. Nuts and washers are examined by angle beam from the outside circumference in both the axial and circumferential directions.

There are no metal platings applied to closure studs, nuts, or washers. A phosphate coating is applied to threaded areas of studs and nuts and bearing areas of nuts and washers to act as a rust inhibitor and to assist in retaining lubricant on these surfaces.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

Limits on pressure and temperature for inservice leak and hydrostatic tests, normal operation (including heatings and cooldown) and reactor core operation are shown in Figures 5.3-4 ~~4a, b and c.~~ The basis used to determine these limits is described in 5.3.1.5. ~~The following are the actual results of the dropweight tests for the vessel cylindrical shell plates:~~

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N.D.T.

	MARK NO.	TOP	BOTTOM
#1 SHELL RING	21-1-1	$-20^{\circ}F$	$-20^{\circ}F$
	21-1-2	$-30^{\circ}F$	$-40^{\circ}F$
	21-1-3	$-30^{\circ}F$	$-30^{\circ}F$
	21-1-4	$-30^{\circ}F$	$-40^{\circ}F$
#2 SHELL RING	22-1-1	$-30^{\circ}F$	$-30^{\circ}F$
	22-1-2	$-30^{\circ}F$	$-30^{\circ}F$
	22-1-3	$-50^{\circ}F$	$-50^{\circ}F$
	22-1-4	$+10^{\circ}F$*	$+10^{\circ}F$*
#3 SHELL RING	23-1-1	$+10^{\circ}F$*	$+10^{\circ}F$*
	23-1-2	$+10^{\circ}F$*	$+10^{\circ}F$*
	23-1-3	$+10^{\circ}F$*	$+10^{\circ}F$*
#4 SHELL RING	24-1-1	$+10^{\circ}F$*	$+10^{\circ}F$*
	24-1-2	$+10^{\circ}F$*	$+10^{\circ}F$*
	24-1-3	$-20^{\circ}F$	$-20^{\circ}F$

*Only sufficient testing to establish $+10^{\circ}F$ N.D.T was performed.

The Charpy V-notch tests for all plates were performed at $+10^{\circ}F$ and all C_V energy test results were at or above 30 ft-lbs, and the lowest reported C_V expansion was 26 mils.

The limit curves shown in Figures 5.3-4a, b and c were established using a nil-ductility transition temperature of $+10^{\circ}F$ which is consistent with the above impact data results.

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5.3.2.2 Operating Procedures

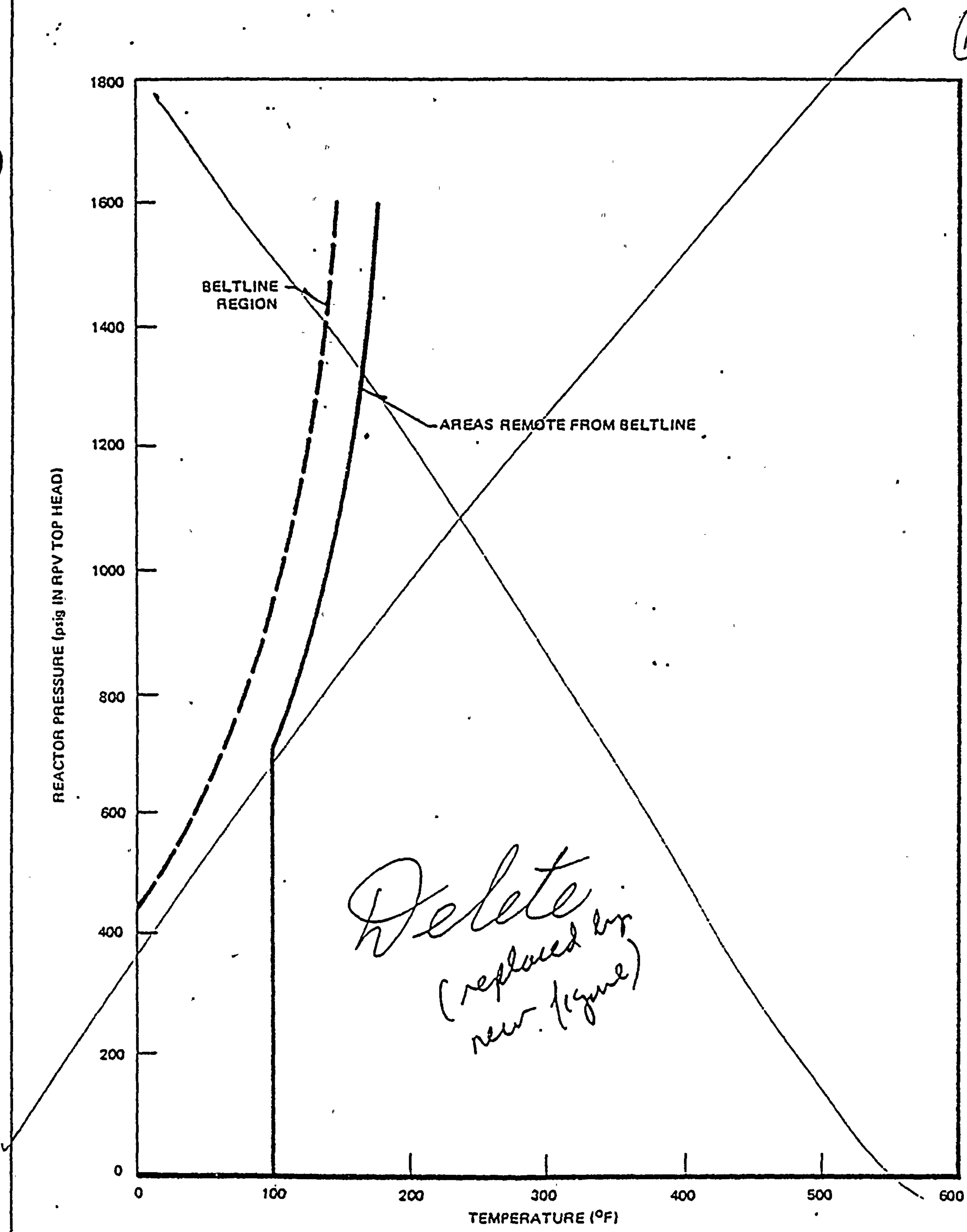
By comparison of the pressure vs. temperature limits in ~~5.3.2.1~~ Figure 5.3-4 with intended normal operating procedures for the most severe upset transient, it is shown that the limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas has a minimum fluid temperature of 250°F and a maximum pressure peak of 1180 psig. Scram automatically occurs with initiation of this event, prior to the reduction in fluid temperature, so the applicable operating limits are given by Figure 5.3-4, curve A. ~~For a temperature of 250oF, the maximum allowable pressure exceeds 1600 psig for the intended margin against nonductile failure.~~ Using Figure 5.3-4 it can be seen that the maximum transient pressure of 1180 psig is therefore within the specified allowable limits. Y

5.3.3 REACTOR VESSEL INTEGRITY

The reactor vessel was fabricated for General Electric's Nuclear Energy Division by CBI Nuclear Co., and was subject to the requirements of General Electric's Quality Assurance program.

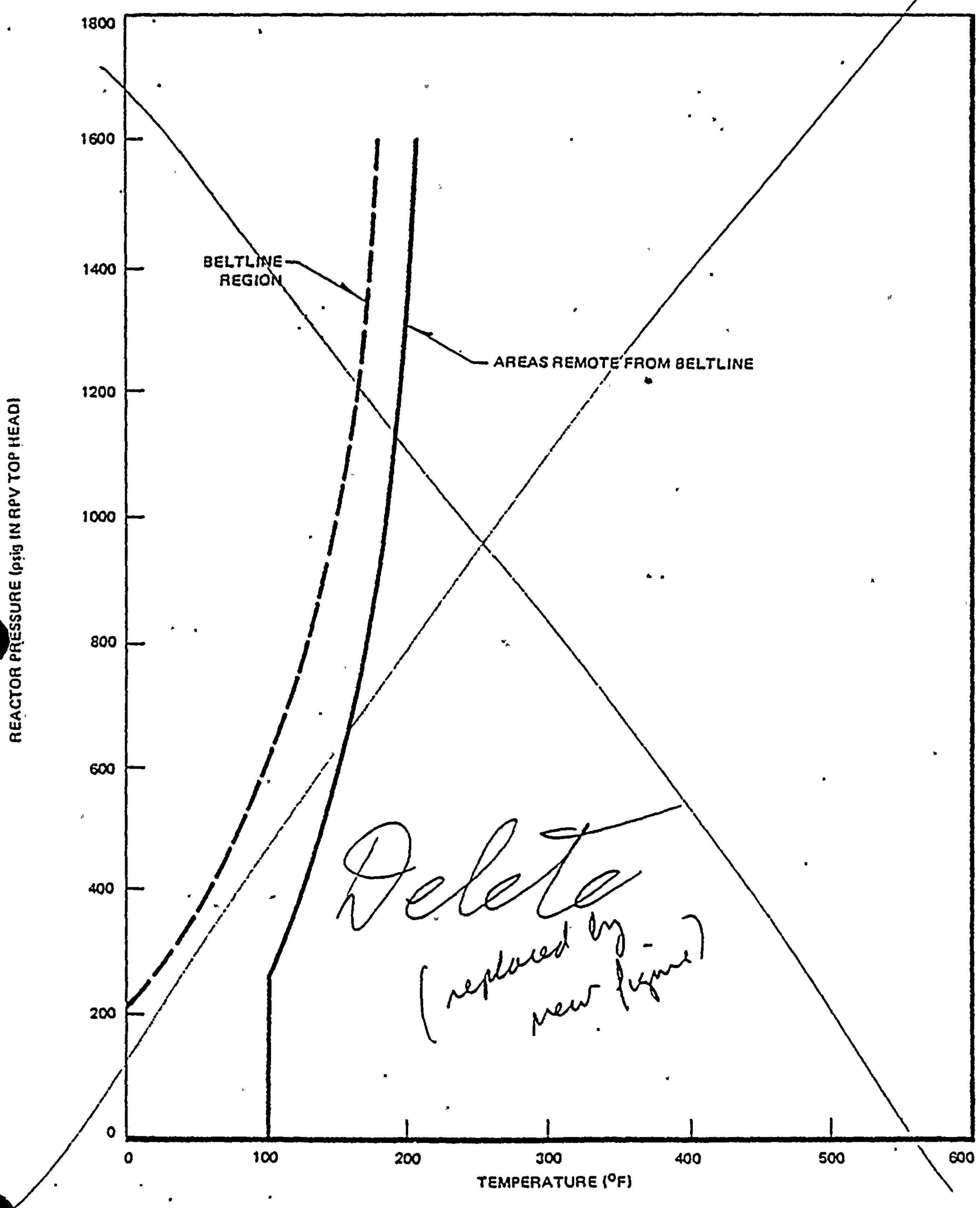
The CBI Nuclear Co., has had extensive experience with G.E. reactor vessels and has been the primary supplier of G.E. domestic reactor vessels and some foreign vessels since the company was formed in 1972 from a merger agreement between Chicago Bridge and Iron Co. and G.E.. Prior experience by the Chicago Bridge and Iron Co. with G.E. reactor vessels dates back to 1966.

Assurance was made that measures were established requiring that purchased material, equipment, and services associated with the reactor vessel and appurtenances conform to the requirements of the subject purchase documents. These measures included provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished, inspection at the vendor source, and examination of the completed reactor vessel.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MINIMUM TEMPERATURE FOR PRESSURE TESTS AS REQUIRED BY ASME CODE SECTION XI	FIGURE 5.3-4

(18)



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MINIMUM TEMPERATURE FOR NON-NUCLEAR HEATUP OR COOLDOWN FOLLOWING NUCLEAR SHUTDOWN	FIGURE 5.3-4b

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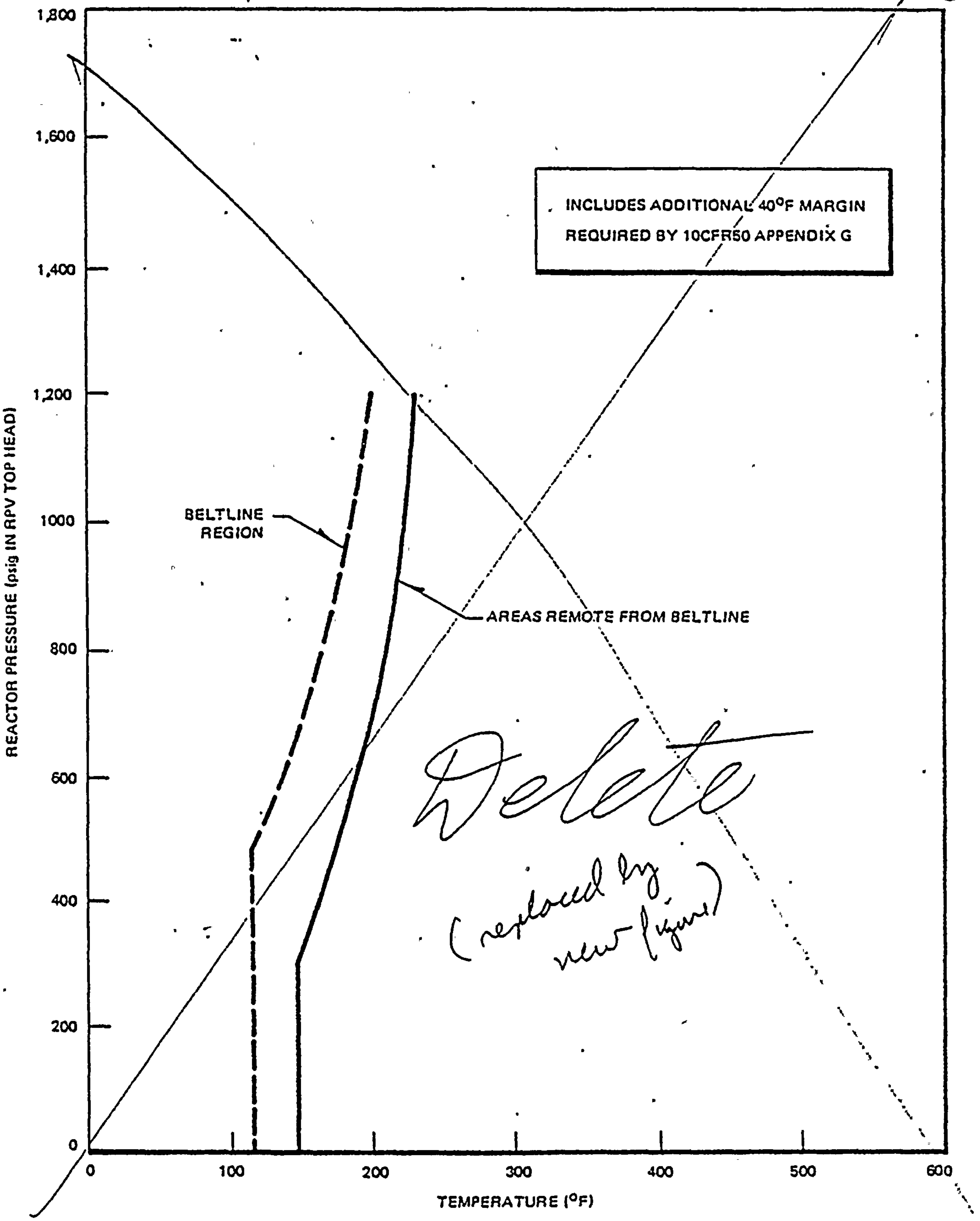


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MINIMUM TEMPERATURE FOR CORE OPERATION (CRITICALITY) INCLUDING ADDITIONAL 40°F MARGIN REQUIRED BY 10 CFR 50, APPENDIX G	FIGURE 5.3-4c

Add: (This figure replaces 5.3-4a, b, &c)

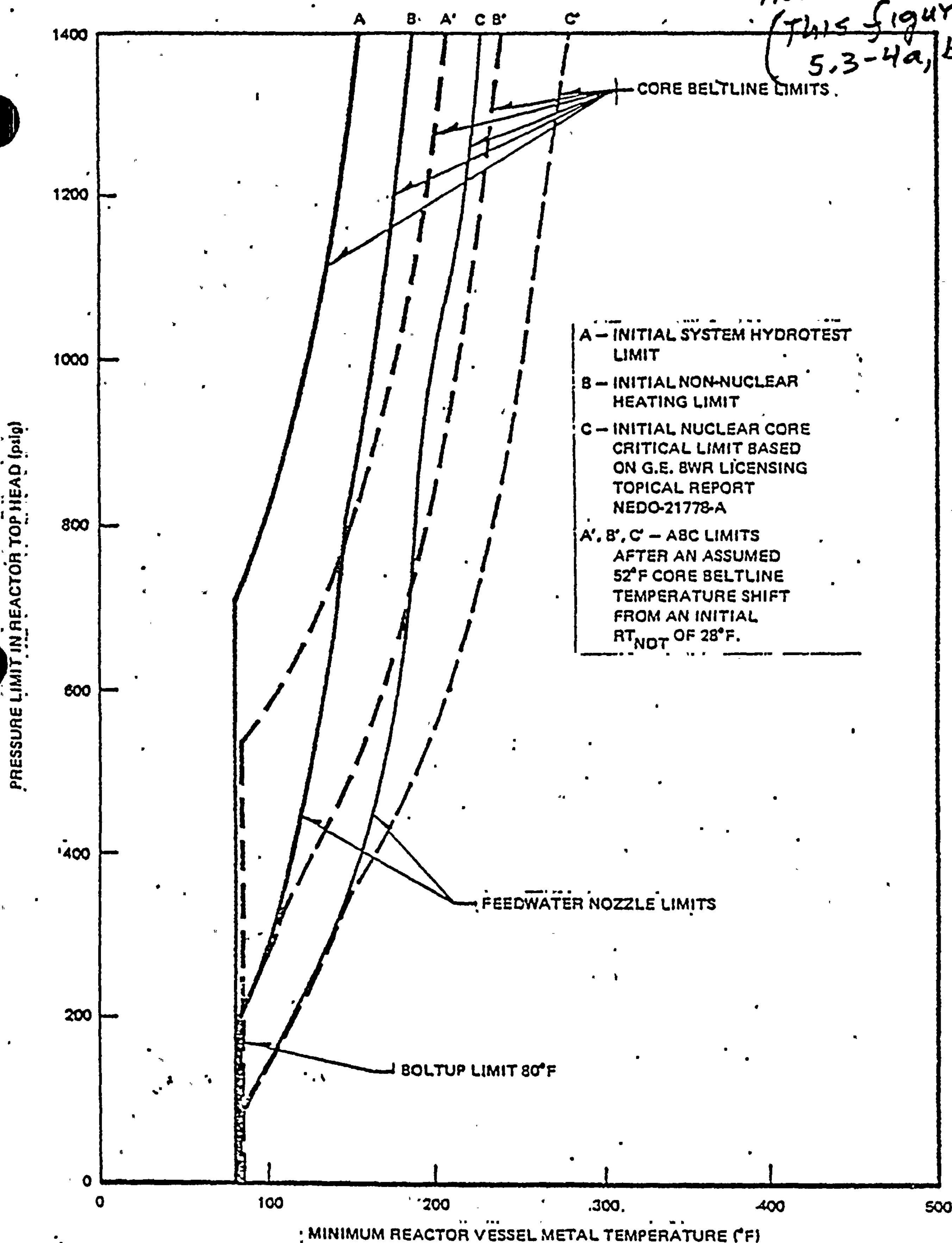
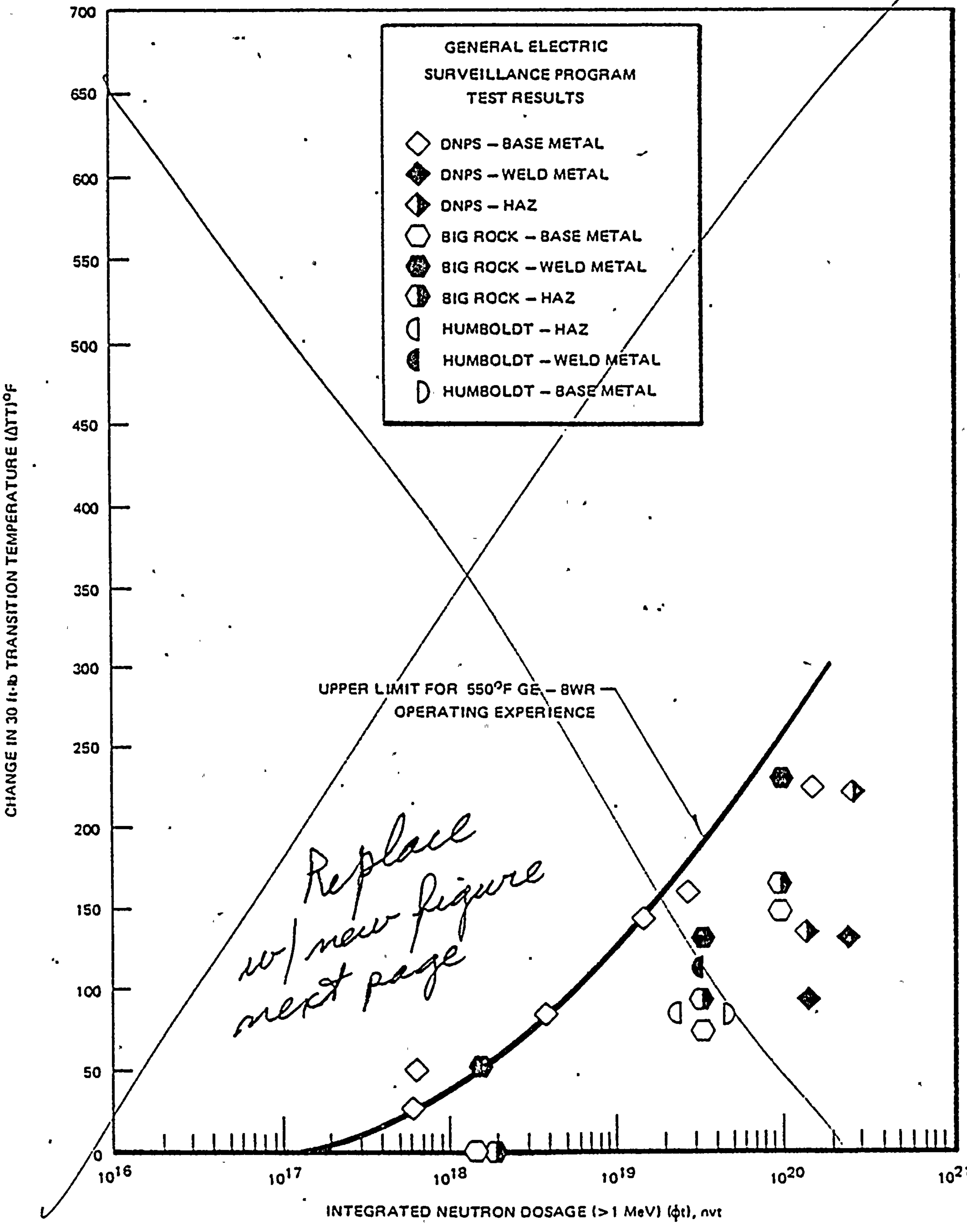


Figure 5.3-4. WPPSS Hanford Unit 2
Minimum Temperature Required versus Reactor Pressure



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CHANGE IN CHARPY V TRANSITION TEMPERATURE VERSUS NEUTRON EXPOSURE	FIGURE 5.3-5

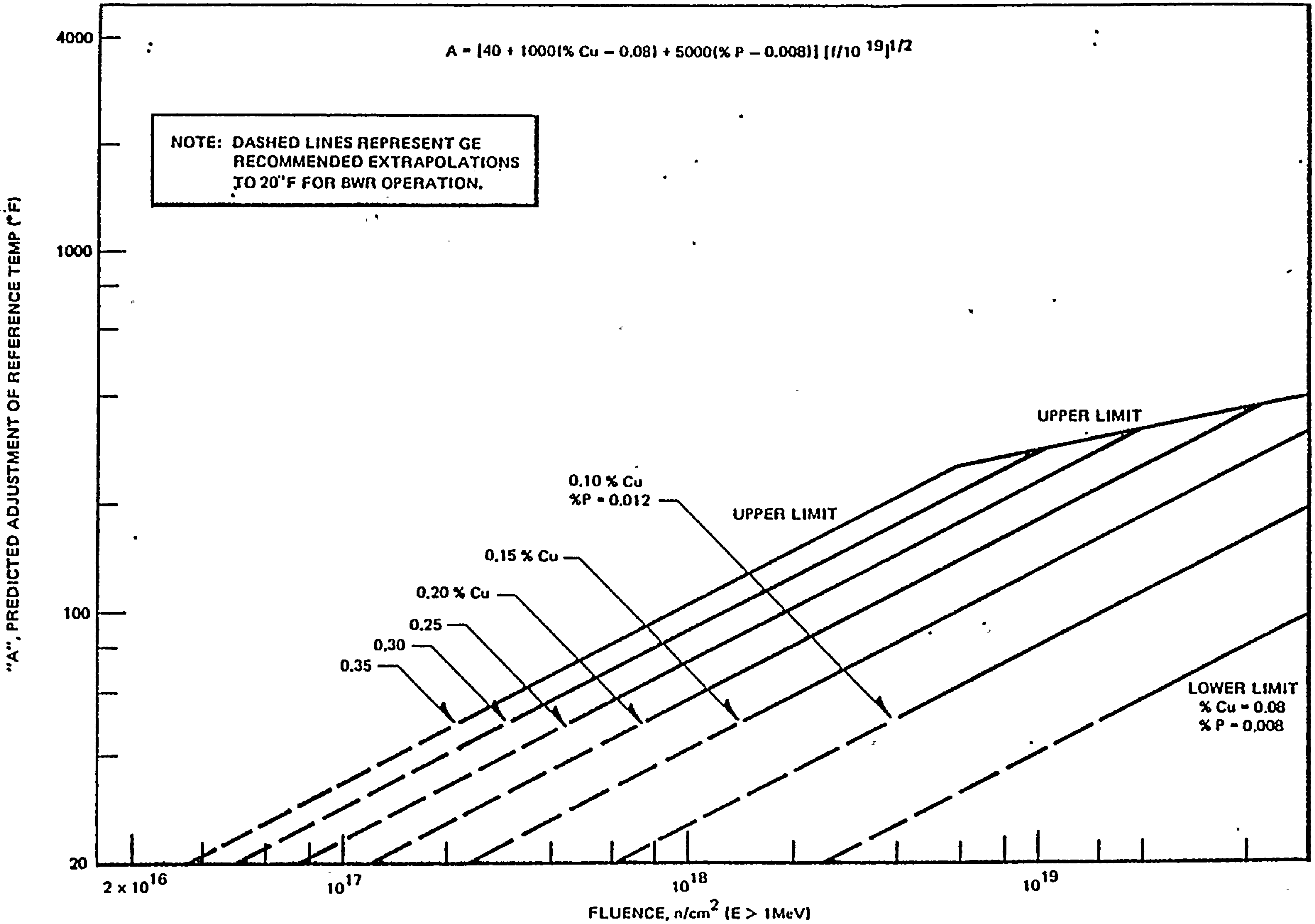


Figure 5.3-5. Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content.
For Copper and Phosphorus Contents Other Than Those Plotted, Use the Expression for "A" Given on the Figure.

(27)

Table 5.3-1a
APPENDIX [illegible]TRIX FOR WNP-2

Appendix G Par. No.	Topic	Comply Yes/No Or N.A.	Alternate Actions Or Comments
I, II	Introduction; Definitions	- -	- -
III.A	Compliance With ASME Code, Section NB-2300	Yes	See Section 5.3.1.5.2 for discussion.
III.B.1	Location & Orientation of Impact Test Spec	Yes	See III.A, above.
III.B.2	Materials Used to Prepare Test Specimens	No	Compliance except for CVN orientation and CVN upper shelf.
III.B.3	Calibration of Temp. Inst. and Charpy Test Machines	No	Paragraph NB-2360 of the ASME B&PV code Section IV was not in existence at the time of purchase of the WNP-2 reactor pressure vessel. However, the requirements of the 1971 edition of the ASME B&PV Section III Code, Summer 1971 addenda, were met. For the discussions of the GE interpretations of compliance and NRC acceptance see References 1 and 2. The temperature instruments and Charpy Test Machines calibration data are retained until the next recalibration. This is in accordance with Reg. Guide 1.88 Rev. 2, GE Alternative Position 1.88 and ANSI N45.2.9, 1974. Therefore, the instrument calibration data for WNP-2 would not be currently available.
III.B.4	Qualification of Testing Personnel	No	No written procedures were in existence as required by the Regulation; however, the individuals were qualified by on-the-job training and past experience. For the discussion of the GE interpretation of compliance and NRC acceptance see References 1 and 2.
III.B.5	Test Results Recording & Certification	Yes	See Reference 1 and 2.
III.C.1	Test Conditions	No	See III.A, III.B.2, above.
III.C.2	Materials Used to Prepare Test Specimens for Reactor Vessel Beltline	Yes	Compliance on base metal and weld metal tests. Test weld not made on same heat of base plate, necessarily.

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Table 5.3-1a (Continued)
APPENDIX G MATRIX WNP-2

Appendix G Par. No.	Topic	Comply Yes/No Or N.A.	Alternate Actions Or Comments
IV.A.1	Acceptance Standard of Materials	- -	- -
IV.A.2.a	Calculates Stress Intensity Factor	Yes	
IV.A.2.b	Requirements for Nozzles, Flanges & Shell Region Near Geometric Discontinuities	No	Plus 60°F was added to the RT_{NDT} for the reactor vessel flanges. For feedwater nozzles the results of the BWR/6 analysis was adjusted to WNP-2 RT_{NDT} conditions.
IV.A.2.c	RPV Metal Temperature Requirement When Core is Critical.	No	Regulation change in process (see LTR NEDO-21778-A).
IV.A.2.d	Minimum Permissible Temp. During Hydro Test	Yes	
IV.A.3	Materials for Piping, Pumps & Valves	No	Main steamline piping is in compliance. See Subsection 5.2.3.3.1 for discussions on pumps and valves.
IV.A.4	Materials for Bolting & Other Fasteners	Yes	Current toughness requirements for closure head studs are met at +10°F even though testing was done per the 1971 ASME code.
IV.B	Minimum Upper Shelf Energy for RPV Beltline	No	Weld and longitudinal CNV data were taken at -20°F and +10°F only. An estimate of compliance to requirements should be made from the first surveillance capsule results per MTEB5-2. Beltline plates were tested with longitudinal CVN's at +10°F only. The minimum values are for Heat C1272-1 (0.15% Cu; 34, 26, 30, 31, 34, 30 ft-lb; 10 and 40% shear at +10°F) and Heat C1273-1 (0.14% Cu; 33, 33, 30, 30, 34, 35 ft-lb; 10% shear at +10°F). Beltline welds were tested with CVN's at +10°F or -20°F only. Lowest weld values are found for Heat 04P046/Lot D217A27A (0.06% Cu; 34, 36, 37, 39, 40 ft-lb; 20 and 30% shear at -20°F), Heat C3L46C/Lot J020A27A (0.02% Cu; 35, 39, 40 ft-lb; 60% shear at +10°F) and Heat 05P018/Lot D211A27A (0.09% Cu; 29, 30, 31, 36, 38 ft-lb; 30 and 40% shear at -20°F). Because of the preceding relatively low test temperatures and Cu contents, it is anticipated that end-of-life upper shelf CVN values would be in excess of 50 ft-lb.

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Table 5.3-1a (Continued)
APPENDIX G MATRIX FOR WNP-1

Appendix G Par. No.	Topic	Comply Yes/No Or N.A.	Alternate Actions Or Comments
IV.C	Requirement for Annealing When RT_{NDT} >200°F	N/A	
V.A	Requirements for Material Surveillance Program	See App. H	
V.B	Conditions for Continued Operation	By Applicant	
V.C	Alternative if V.B Cannot be Satisfied	By Applicant	
V.D	Requirement for RPV Thermal Annealing if V.C Cannot be Met	N/A	
V.E	Reporting Requirement for V.C & V.D	By Applicant	

References

1. Letter MFN-414-77, G. G. Sherwood (GE) to Edson G. Case (NRC) dated October 17, 1977.

2. Letter, Robert B. Minogue (NRC) to G. G. Sherwood (GE) dated February 14, 1978.

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Table 5.

APPENDIX H MATRIX FOR WNP-2

Appendix H Par. No.	Topic	Comply Yes/No Or N.A.	Alternate Actions Or Comments
I	Introduction	N/A	
II.A	Fluence $<10^{17}$ n/cm^2 - Surveillance Program Not Required	N/A	
II.B	Standards Requirements (ASTM) for Surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from actual beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are to be supplied.
II.C.1	Surveillance Specimen Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from alongside specimens required by Section III of Appendix G and transverse CVN's may not be employed. However, representative materials have been used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of Surveillance Capsules in RPV	Yes	Code basis is used for attachment of brackets to vessel cladding. See Section 5.3.1.6.2.
II.C.3.a	Withdrawal Schedule of Capsules, RT_{NDT} <100°F	Yes	Three capsules planned. Starting RT_{NDT} of limiting material is based on alternative action (see paragraph III.A of Appendix G).
II.C.3.b	Withdrawal Schedule of Capsules, RT_{NDT} <200°F	N/A	
II.C.3.c	Withdrawal Schedule of Capsules, RT_{NDT} >200°F	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	By Applicant	
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	By Applicant	
IV.A	Reporting Requirements of Test Results	By Applicant	

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Table 5.3-1b (Continued)
APPENDIX H MATRIX FOR WNP-2

Appendix H Par. No.	Topic	Comply Yes/No Or N.A.	Alternate Actions Or Comments
IV.B	Requirement for Dosimetry Measurement	By Applicant	
IV.C	Reporting Requirements of Press/Temp. Limits	By Applicant	

Q 121.4 (5.3.1)

Paragraph C.2.b of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," October 1973, suggests that the reactor pressure vessel studs and nuts be ultrasonically examined after final heat treatment according to ASME Specification SA-388. However, in Section 5.3 of the FSAR, you state that paragraph C.2.b of the regulatory guide cited above was not followed in the ultrasonic examination of the reactor vessel closure studs. Accordingly, provide the details of the ultrasonic procedure which was used so that we may compare the test procedure used with the requirements of SA-388.

Response:

We can not find any statement in Section 5.3 which says we are not following paragraph C.2.b of Regulatory Guide 1.65. However, the write-up for Regulatory Guide 1.65, in Appendix C, pages C.2-65 and -66, was in error and will be corrected.*

*See attached draft pages.

Regulatory Guide 1.65, Rev. 0, October 1973

Materials and Inspection for Reactor Vessel Closure Studs.

Regulatory Guide Intent:

Regulatory Guide 1.65 defines acceptable materials and testing procedures with regard to reactor vessel closure stud bolting for light-water-cooled reactors.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design and/or equipment utilized in this facility is in compliance with the intent of the subject regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The reactor pressure vessel closure studs are SA540 Grade B23 or 24 (AISI4340) and have a maximum ultimate tensile strength of 170 ksi. Additionally, specified bolting material must have Charpy V notch impact properties of 45 ft. lbs. minimum with 25 mils lateral expansion. Nondestructive examination before and after threading is specified to be in accordance with subarticle NB-2580 ASME Section III, which complies with regulatory position C. 2. Subsequent to fabrication, the studs are manganese phosphate coated and are lubricated with a graphite/alcohol or a nickel powder base lubricant.

In relationship to regulatory position C.2.b, the bolting materials were ultrasonically examined after final heat treatment and prior to threading, as specified. The specified requirement for examination according to SA-388 was ~~not~~ complied with. ~~However,~~ The specific procedures approved for use in practice are judged to insure comparable material quality and, moreover, are considered adequate on the basis of compliance with the applicable requirements of ASME Code Paragraph NB2585.

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General Compliance or Alternate Approach Assessment: (Cont'd)

Additionally, straight beam examination was performed on 100% of cylindrical surfaces, and from both ends of each stud using a 3/4 maximum diameter transducer. In addition to the code required notch, the reference standard for the radial scan contained a 1/2 inch diameter flat bottom hole with a depth of 10% of the thickness, and the end scan standard contained a 1/4 inch diameter flat bottom hole 1/2 inch deep. Also, angle beam examination was performed on the outer cylindrical surface in both a flat and circumferential direction. Surface examinations were performed on the studs and nuts after final heat treatment and threading, as specified in the guide, in accordance with NB2583 of the applicable ASME Code.

Radial scan calibration is based on a 1/2 inch (12.7 mm) diameter flat bottom hole of a depth equal to 10% of the material thickness. ~~End scan calibration is in accordance with NB 2585.~~ Angle beam examination is performed on the outer cylindrical surface of nuts and washer per ASME SA-388 in both axial and circumferential directions. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve per NB-2585 is used for the longitudinal wave examination.

In relationship to regulatory position C.3, General Electric practice allows exposure of stud bolting surfaces to higher purity fill water; nuts and washers are dry stored during refueling.

Specific Evaluation Reference:

Refer to 5.3.1.7.

Similar Application Reference:

Similar application was utilized on Zimmer and LaSalle.

QUESTION 121.5 (5.3)

Paragraph II.C.2 of Appendix H to 10CFR Part 50 states in part: "Surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region,..." In Section 5.3 of the FSAR, you state that the capsule holder brackets were welded to the reactor pressure vessel innerwall. Present sufficient design and fabrication detail to demonstrate that the capsule attachments were designed and constructed in accordance with accepted standards, such as the Section III rules of the ASME code for attachments to vessels.

RESPONSE

The surveillance brackets are welded to the clad material which surfaces the pressure vessel walls and is not attached to the pressure boundary directly. As attached, the brackets do not have to comply with specifications of the ASME Pressure Vessel Code.

EC:cas/49C3

QUESTION 121.6 (1.6 and 5.3)

In Sections 1.6 and 5.3 of the FSAR, you reference the General Electric report NEDO-20631, "Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR-6 Plants," dated March 1975. At present this report has not been submitted to the NRC staff. Accordingly, submit the information contained in this particular GE report so that we may determine whether the WNP-2 facility complies with the requirements of Appendix H to 10CFR Part 50.

RESPONSE

The report NEDO-20631 has been withdrawn as a reference for WNP-2, and was replaced by report NEDO-21708, "Radiation Effects in Boiling Water Reactor Pressure Vessels Steels," dated October 1977. The NRC staff has been provided NEDO-21708 for review.

NEDO-21708 addresses the requirements of Appendix H to 10 CFR Part 50 and supports the current application of Regulatory Guide 1.99.

See revised Section 5.3* and 1.6**.

*draft submitted with the response to question 121.2.
**draft attached.

TABLE 1.6-1 (Continued) Page 7 of 10

REPORT NUMBER	TITLE	FSAR PORTIONS WHERE REFERENCED
NEDO-20626-2	Studies of BWR Designs for Mitigation of Anticipated Transients without Scrams (July 1975)	15.8
~~NEDO-20631~~	~~Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR/6 Plants (March 1975)~~	~~5.3~~ delete
NEDO-20913	Lattice Physics Methods (June 1975)	4.3(13)*
NEDO-20922	Experience with BWR Fuel Through September 1974 (June 1975)	4.2(21)*, 11.1
NEDO-20939	Lattice Physics Methods Verification (August 1975)	4.3(5)*
NEDO-20944 NEDE-20944P	BWR/4 and BWR/5 Fuel Design (October 1976)	Table 1.3-1 4.2*, 4.3*, 4.4*
NEDO-20946	BWR Simulator Methods Verification (May 1976)	4.3(6)*
NEDO-20948-P	BWR/6 Fuel Design (June 1976)	4.2(3)*
NEDO-20953	Three-Dimensional Boiling Water Reactor Core Core Simulator (May 1976)	15.4
NEDO-20964	Generation of Void and Doppler Reactvity Feedback for Application to BWR Plant Transient Analysis (August 1975)	4.3(7)*
NEDO-21061 and NEDE-21061P	Mark II Containment Dynamics Forcing Functions Information Report (September 1976)	1.5, 6.2

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Q. 121.7
(10.2.3)

To provide assurance that there is a low probability of producing high energy missiles at the operating speed or design overspeed of the turbine-generator, provide documentation including the results of material property testing, to show the degree of conformance of the turbine-generator with the guidelines in paragraph II of section 10.2.3 of the Standard Review Plan "Turbine Disk Integrity."

RESPONSE:

Amendment 2 to the FSAR submitted in December 1978, changes section 10.2.3 to incorporate two references which provide the requested information.*

*copies of changed pages attached.

The main steam stop and control valves are located in the steam chest assembly which is parallel to the axis of the high pressure turbine. The closure time is .2 seconds for the stop valves and .25 seconds for the control valves.

A failure of one valve causes the other valves to increase or decrease their opening to compensate for that valve. If one valve fails open at low load condition, the other valves close. If the closing of the other valves is not enough to compensate for that valve, the turbine overspeeds thereby increasing the generator output frequency.

The reheat stop and intercept valves are inline valves located in the crossover piping between the moisture separator/ reheater and low pressure turbine. The closure time for these valves is .15 seconds.

Each of the extraction steam lines has a reverse current valve and a gate valve, with the exception of the extraction lines to low pressure heater number 1. These valves are located near the condenser. Upon turbine trip the reverse current valves close. The closure time for these valves is .5 seconds. Because of the fast closure time and the short distance between these valves and the extraction points at the turbine, the amount of steam in these lines does not effect the turbine coastdown following a turbine trip.

10.2.3 TURBINE DISK INTEGRITY

Analysis of potential turbine missile hazards and drawings showing the orientation of the turbine with respect to important structures are presented in 3.5. Discussions concerning disk materials and their properties, design, and inspection are available in References 10.2-1 and 10.2-2.

10.2.4 SAFETY EVALUATION

The steam entering the high pressure turbine may contain fission, coolant activation and activated corrosion products. The anticipated concentration of nitrogen-16, which is the dominant radionuclide entering the high pressure turbine, is discussed in 12.2. Moisture separation and transit time between the high pressure and low pressure turbines reduces the concentration of radionuclides in the steam prior to entering the low pressure turbine. Most of the gaseous radioactivity is removed by the steam-jet air ejector and routed to the off-gas system (Refer to 11.3). The condensate in the condenser hotwell contains significantly less radioactive material than the inlet steam.

10.2.6 REFERENCES

10.2-1 Westinghouse Electric Corporation, "Report Covering the Effects of a High Pressure Turbine Rotor Fracture and Low Pressure Turbine Disk Fractures at Design Overspeed" 296/281 A, (April 1975).

10.2-2 Westinghouse Electric Corporation, "Report Covering the Effects of a Turbine Accelerating to Destructive Overspeed" 296/281 B, (April 1975).

Q 121.8

(In responding to this item, refer to the responses to Items 121.15 and 121.18 on the Hatch-2 docket.) Additional information is required to demonstrate that: (1) the thermal sleeve/sparger design of the feedwater inlet nozzle has been evaluated with respect to potential nozzle cracking resulting from thermal cycling; and (2) a program of scheduled augmented inservice inspection has been developed.

These inservice inspections should be conducted with a method sufficiently sensitive to provide assurance that small cracks can be detected. Accordingly, we require you to supply the following information:

a. That technical basis to assure the structural integrity of both the feedwater inlet nozzle and the sparger.

b. An evaluation of the feasibility of installing automated ultrasonic testing (UT) fixtures on all feedwater inlet nozzles with particular attention focused on the examination of the nozzle bore region.

c. An evaluation of the feasibility of performing the internal surface examination by magnetic particle methods.

Your response should contain: (1) a description of the nozzle and sparger design including the significant dimensions, the materials of construction and the weld locations; (2) a description of the analyses and test data, referencing appropriate data previously submitted to the NRC staff if it is applicable for the WNP-2 facility; (3) the detailed projected crack growth rates, stress levels and usage factors for both the nozzle and the sparger; (4) any plant modifications that are planned to reduce the temperature differential between the feedwater and the water in the reactor pressure vessel during low power operation; and (5) a description of any instrumentation that will be installed in the reactor pressure vessel to verify the conclusions of your design analysis.

Several ultrasonic testing concepts and procedures have been used to examine the feedwater inlet nozzle regions in operating plants. Identify which of these ultrasonic testing procedures will be used in the WNP-2 facility. Discuss the influence on crack detection, using your ultrasonic testing methods in the WNP-2 faility, of local grindouts.

In addition, provide a description of the augmented inservice inspection (ISI) program to be implemented including scheduled surface examination, ultrasonic testing and verification of the leak tight integrity of the joint between the thermal sleeve and the safe end on all nozzles. The essential elements of an acceptable program are given in the Appendix attached to this set of questions.

Response:

A description of the WNP-2 feedwater nozzle and sparger is presented on Figures 121.8-1 and 121.8-2.

The mechanisms which have caused a cracking in operating BWRs are understood. A summary discussion of problems and the solutions incorporated in the WNP-2 design is presented in the following.

A detailed evaluation fo the problems of the feedwater nozzle and sparger is presented in NEDE-21821 "BWR Feedwater Nozzle/Sparter Final Report" March 1978. The solution of the feedwater nozzle and sparger cracking problems involves several elements, including material selection and processing, nozzle clad elimination, and thermal sleeve and sparger redesign. The following summarizes the problems and solutions that have been implemented in the WNP-2 design.

PROBLEM	CAUSE	FIX
Sparger Arm Cracks	Vibration	Eliminate clearance between thermal sleeve and safe end.
RPV Feedwater Nozzle	Thermal Fatigue	Eliminate clad, eliminate leakage with a welded joint between the sparger and safe-end.

The sparger vibration has been attributed to a self-excitation caused by instability of leakage flow through the annular clearance between the thermal sleeve and safe end. Tests have shown that the vibration is eliminated if the clearance is reduced sufficiently or sealed. The solution which has been selected for WNP-2 uses a welded joint to assure no leakage. This feature is also an essential part of the solution of the nozzle cracking problem. Freedom from vibration over a range of conditions has been demonstrated by the tests reported in NEDE-23604.

The cracking of the feedwater nozzles is a two-part process. The crack initiation mechanism as discussed above is the result of self-initiated thermal cycling. If this were the only mechanism present, the cracks would initiate, grow to a depth of approximately 0.25 inch, and arrest. This degree of cracking could be tolerated, but unfortunately there is another mechanism which supports crack growth. This mechanism is the system induced transients, primarily the startup/shutdown transients. The welded thermal sleeve arrangement also assists in this area because without leakage, the heat transfer coefficient between the feedwater and the nozzle are reduced to the point where the thermal stresses in the nozzle are not high enough to cause a significant crack growth. Analyses presented in NEDE-21821, Section 4.7, demonstrates the benefits of the welded thermal sleeve and of using unclad nozzles. With these demonstrated benefits, WNP-2 does not believe it necessary to install instrumentation for design verification.

WNP-2 has installed an automatic feedwater low flow control valve, RFW-FCV-10. This valve has the capability to control flow down to 362 gpm, or about 1.25% of total flow. This valve will substantially reduce the temperature differential between the feedwater and the water in the RPV during low power operation.

The following paragraphs address RPV feedwater nozzle examination questions other than Appendix A to Section 121.

Feasibility of Installing Mechanized Ultrasonic Scanners

All feedwater nozzle inner radii, safe-end, and bore regions are capable of automated ultrasonic examination. Both perservice and inservice inspections of the inner radii and safe end welds will be performed using such equipment. Tooling has been contracted from the baseline examination agency that will allow nozzle inner radius scanning by contacting an angle beam transducer to the vessel plate surface adjacent to the nozzle to vessel weld. The scanner mechanism is removable and would be compatible with any of the six (6) feedwater nozzles. The Supply System is currently evaluating the benefit of performing an automated examination of the bore region versus a manual examination, in terms of radiation exposure (examination/setup time) and examination coverage. The technique providing the best balance of those two factors will be chosen. Adequate access exists for either technique.

Scanning of the nozzle bore region can be accomplished from the cylindrical section of the nozzle forging. A manual examination of this region is possible to accomplish in less than ten minutes of scanning time per nozzle by one operator supported inside the biological shield cut-out. Data recording, should it be necessary, can be accomplished by a second examiner positioned outside the shield using redundant electronic instrumentation and analog recorders. Assuming ten minutes examination time per nozzle and a radiation field of 150 mr/hr, the examiner would receive 25 mr per nozzle.

The automated examination devices would be mounted on temporary tracks, would be installed just prior to the examination and removed following the examination. It is not considered feasible to leave the equipment installed during plant operation as installation and removal time is minimal and would be quickly offset by equipment recalibration and maintenance costs considering the adverse environment such equipment would be subjected to during plant operation.

Feasibility of Magnetic Particle Examination

Handheld magnetic yokes will not readily fit in the envelope between the sparger body and the nozzle radius, and yet make good contact with the low alloy steel surface. Poor contact could result in arc-strikes below the electrodes, these surface defects are localized heat affected zones of higher hardness than the surrounding metal. If the arc-strike was accompanied by localized cracking, then surface grinding would be necessary to restore the nozzle to its original surface condition. Considering the above, magnetic particle examination methods are not considered feasible inside the reactor vessel with the present sparger configuration.

Ultrasonic Examination Methods

The nozzle inner radius examination will be made by pulse-echo ultrasonic techniques from the exterior of the reactor pressure vessel by contacting the vessel plate surface. This technique is similar to that used by the General Electric Company and the firm of Lambert MacGill and Thomas. Procedures for the examination will be in a format consistent with others used by the Supply System, but the technical content will be comparable to procedures previously qualified by the above referenced testing organizations.

Examination of the nozzle bore region will be performed by pulse-echo ultrasonic techniques from the cyclindrical section of the nozzle forging using sound beam geometry similar to that used by the General Electric Company. The Supply System plans to extend the coverage of this technique toward the inner radius by added sound beam refraction. Prior to use on the WNP-2 feedwater nozzles, a qualification check is intended to be made on a mock-up to demonstrate the techniques validity.

Should local grind outs be made in the examination surface creating a depression with definable sides, depth, and length, the ultrasonic techniques being used would obtain reflections from these cavities. Such reflections can be minimized by blending the grind cavity into the surrounding base metal. This would result in improved detection sensitivity to postulated thermal fatigue cracks propagating from the grind cavity.

The Supply System will implement the reactor feedwater (RFW) RPV nozzle inspection program described below, which addresses Appendix A to this question on an item-by-item basis. Justification for any deviations from the Appendix A requirements is presented following the response.

I. AUGMENTED INSERVICE INSPECTION PROGRAM

A. Preservice Examination

The Supply System will perform a PSI ultrasonic examination of RFW nozzle inner radii, bore and safe end regions as described in the WNP-2 PSI Program Plan. The personnel and UT procedures used will be qualified as described in II.C below.

In addition, a preservice liquid penetrant examination will be performed on the accessible areas of all RFW nozzle inner radius surfaces.

B. Inservice Examination

B.1 The Supply System will perform an ultrasonic examination of 1 of 6 reactor feedwater nozzle inner radii, bore and safe end regions each refueling outage using procedures and personnel subject to the same qualifications used during the PSI examinations. A different nozzle will be examined each outage. No surface examinations will be performed on the nozzle inner radii unless such a test is required to verify the nature of an indication discovered using the ultrasonic technique when the indication is suspected to result from service induced cracks on the nozzle inner surfaces. In the event an indication is discovered and found to result from service induced cracks propagating from the nozzle inner surfaces, the following action will be taken:

1. All remaining feedwater will be examined using both ultrasonic (from the OD) and penetrant techniques during the refueling outage in which the cracking is verified.

2. All surface indications determined to be service induced cracks will be removed by local grinding.

3. An inspection method, such as a leak test, will be used to determine the integrity of each of the RFW thermal sleeve to safe end joints.

4. Appropriate corrective action will be taken as required and as practical to prevent recurrence of crack initiation. A program and schedule for implementing such corrective action will be prepared and submitted to the Commission prior to its implementation.

5. A RFW nozzle examination program for subsequent refueling outages will be modified to include an external ultrasonic examination of all feedwater nozzle inner radii, bore and safe end regions for each scheduled refueling outage for 3 consecutive outages. If no new indications are discovered, or if new indications are determined to not result from service induced cracks at the nozzle inner surfaces, the original Supply System program will be resumed. If after 3 additional outages no new indications resulting from surface induced cracks are detected, subsequent examinations will be performed in accordance with normal ASME Section XI requirements.

6. The conduct of surface examinations of accessible nozzle inner radius surfaces will continue to be used throughout plant life only to confirm or characterize new ultrasonic indications which are suspected to result from service induced cracks at the nozzle inner surfaces.

B.2 As stated in B.1 above, the Supply System will perform a surface (penetrant) examination of accessible inner surfaces on all RFW nozzles during the preservice examination program. Subsequent surface examinations of those surfaces will be performed only to verify the nature of an indication discovered using the ultrasonic technique when the ultrasonic indication provides evidence of previously unidentified service induced cracks.

B.3 See response to B.2 above.

If after the sixth planned refueling outage following commercial operation no indications resulting from service induced cracks are found, the subsequent inservice examinations will be performed in accordance with the normal ASME Section XI requirements. Any indications resulting from service induced cracks which are subsequently found will result in the corrective action described above.

C. <u>Thermal Sleeve to Safe End Joint</u>

As stated in B.1 above, the Supply System will perform an inspection of the thermal-sleeve-to-safe-end weld joint, such as a leak test, only if service induced cracks or some other anomoly is discovered which would bring the integrity of the joint into question. In that case, the feedwater piping will be filled with water and the area of the thermal-sleeve-to-safe-end joint will be inspected for indications of leakage.

II. ACCEPTANCE CRITERIA

A. The Supply System will comply with this criteria as stated in B.1 above.

B. The Supply System will comply with this criteria as stated in B.1 above.

C. The Supply System will comply with option (b), in that both the examination personnel and the procedures to be used on the nozzles will be qualified on a full size nozzle mock-up. Supply System examiners will be trained by individual NDE specialists having previous experience with the General Electric Company procedures and their nozzle test program. These examiners will undergo further training, practice, and qualifications on a full size nozzle mock-up. The mock-up will be unclad if negotiations can be reached with a utility owning such a mock-up. As an alternative, the examiners will qualify on a clad mock-up owned by the General Electric Company. Following the qualification process, the examinations will be conducted under the direct supervision of the experienced NDE specialists responsible for ultrasonic technique and procedure development for the Supply System.

III. RECORDING AND REPORTING STANDARDS

The Supply System will record crack indications and report inspection results in compliance with the requirements stated in NUREG-0312.

121.8 JUSTIFICATION OF DEVIATION FROM APPENDIX A

I.B.1 Ultrasonic Examination Frequency

The Supply System will examine only one RFW nozzle per refueling outage rather than all nozzles using an ultrasonic technique from the outside of the vessel. This is justified for the following reasons, which reflect a significant advance in the WNP-2 design and operating procedures towards the long term solution of the BWR nozzle cracking problems per NUREG-0312, Section 8.0, Part. 1.

a. Improved Design: The WNP-2 RFW welded thermal-sleeve-to-safe-end joint provides a "zero leakage" design. This design essentially eliminates the primary historical initiating source of nozzle cracking in BWR's.

b. No Nozzle Cladding: The WNP-2 RFW nozzle surfaces are not clad. The likelihood of crack initiation in unclad nozzles is more than a factor of a 5 less than for clad nozzles. All cracks in BWR feedwater nozzles have initiated in the clad metal.

c. Proven Examination Technique: The ultrasonic examination equipment and personnel to be used in performing both baseline and inservice ultrasonic examinations will be qualified on a full scale mockup of the nozzle, simulating the nozzle geometry and anticipated fatigue crack defects. Since the WNP-2 reactor feedwater nozzles are unclad as stated in b) above, a more sensitive examination is possible due to lack of clad/basemetal interface.

d. Augmented Examination Frequency: The above stated program provides RFW nozzle examination coverage at nearly twice the frequency of the ASME Section XI requirements, i.e., all RFW nozzles will be examined within 6 years (approximately) rather than within 10 years.

e. Feedwater Temperature Controls: As previously stated, WNP-2 has incorporated a feedwater, low flow control valve. The advantages gained from low flow control are identified in Section 4.7 of NEDE-21821.

f. Projected Crack Growth Rates: As presented in Section 4.7 of NEDE-21821, the WNP-2 design should have greater than 35 years of operation, considering our low flow control, prior to an initiated crack reaching 1 inch in depth. This provides for a minimum of 4 examinations per nozzle before reaching a point requiring repair. Even if the extremely conservative (factor of 5) upper bound crack growth curve is applied, each RFW nozzle would be examined by the Supply System program prior to a crack becoming 1 inch in depth. It is clear that ample conservatism exists in the Supply System examination frequency of the RFW nozzles.

The above factors, when combined, provide a great deal of assurance that the factors which have led historically to BWR RFW nozzle cracking have been virtually eliminated. Furthermore, any cracking have been virtually eliminated. Furthermore, any cracking which might occur from unanticipated sources will be discovered before propogating to a significant depth due to low flow controls and an augmented examination schedule with state-of-the-art qualified ultrasonic examination techniques.

I.B.2&3 <u>Surface Examinations</u>

The Supply System will perform surface (penetrant) examinations of the accessible internal surfaces of RFW nozzles during the preservice inspection program. Inservice surface examinations will be performed only when indications of service induced cracking are detected using the ultrasonic examination technique. This is justified as follows:

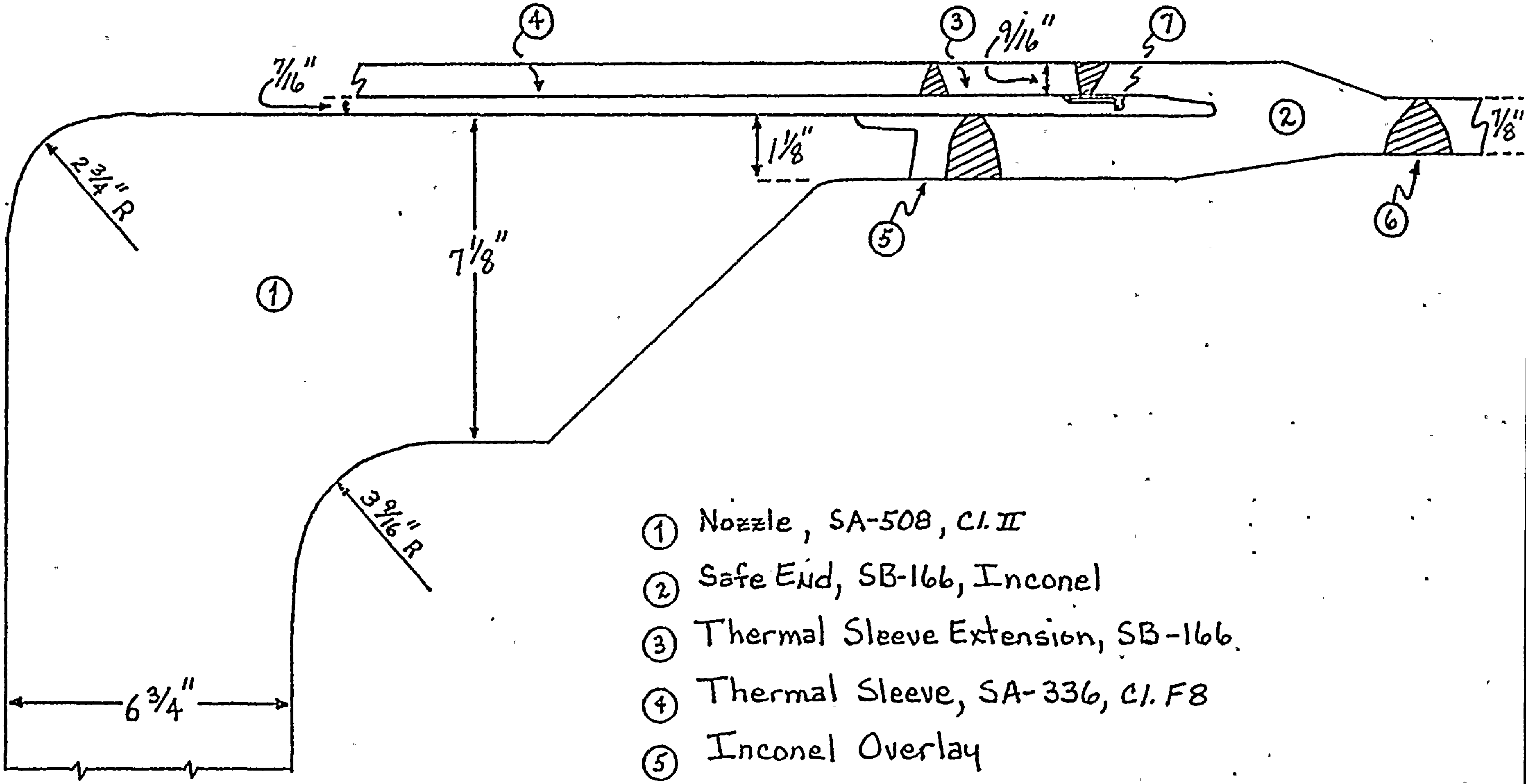
a. Reduced probability of crack initiation and growth as stated in the justification under I.B.1a) through f) above.

b. Access: In order to obtain access to perform a penetrant surface examination of the RFW nozzle surfaces during a refueling outage, the vessel water level would have to be lowered below the level of the spargers and hydrolaser decontamination performed. A special shielded work platform would have to be devised to minimize radiation exposure. This technique was performed at Vermont Yankee resulting in about 15 man rem.

I.C <u>Leak Test</u>

Thermal-Sleeve-to-Safe-End Joint: The Supply System inspection to determine the integrity of the thermal-sleeve-to-safe-end joint only when indications of service induced cracking are detected using the ultrasonic examination technique. The justification for this exception is similar to the justification for not performing inservice surface examinations cited above, with the following additional justification:

1) Test Effectiveness: The maximum pressure which could practically be placed on the subject weld joint would be that available from the static head of a filled sparger, or approximately 6" of water. The effectiveness of this test to reveal throughwall cracks in the weld joint is questionable, since the weld experiences significantly higher differential pressure and temperature during operation. Furthermore, this test would not provide evidence of other than gross throughwall cracks which, if and when detected via such a test, will in all likelihood have been resulting in some degree of leaking for a significant period of time. As was previously demonstrated, any cracks developing as a result of such leakage will be detected prior to the crack propagating to a depth which would jeopardize the nozzle integrity. There is, therefore, no appreciable benefit from performing the leakage test of the spargers other than to determine the status of their integrity in the event service induced cracks are confirmed. Since there is no appreciable benefit, and the cost in dollars and man rem exposure for such a test is quite high, the performance of this test on a routine basis is not justified.



① Nozzle, SA-508, Cl. II

② Safe End, SB-166, Inconel

③ Thermal Sleeve Extension, SB-166

④ Thermal Sleeve, SA-336, Cl. F8

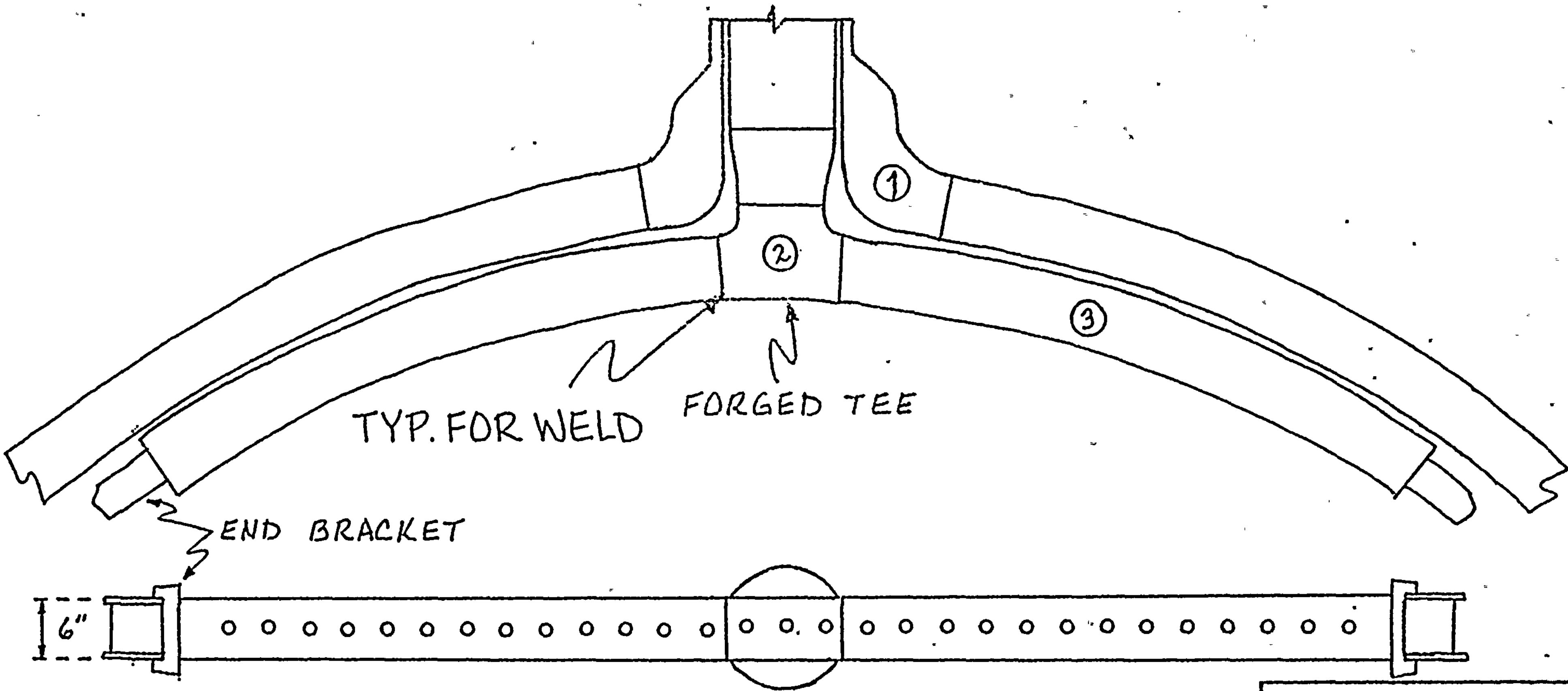
⑤ Inconel Overlay

⑥ Weld Illustration

⑦ Back-up Ring, SB-168

WNP-2
FEEDWATER
NOZZLE
FIG. 121.8-1

① NOZZLE, SA-508, Cl. II
② FORGED TEE, 304 S.S.
③ SPARGER HEADER, 304 S.S.



WNP-2
FEEDWATER
SPARGER
FIG. 121.8-2

Q 121.9

Considering the recent service experience of boiling water reactors with respect to cracking of the vessel nozzle and the vessel wall near the control rod drive (CRD) return line, we require that you provide a description of any proposed plant modifications (e.g., changes in material, location of the CRD return line and deletion of the CRD return line) that will preclude such cracking, including a complete technical justification for the proposed modifications.

Response:

In order to preclude control rod drive return line cracking on WNP-2, the return line will be deleted, and the system modified. The modification primarily consists of adding pressure equalizing valves between the exhaust and cooling water headers, and the use of reverse flow through multiple HCV solenoid valves V-121 as the CRD system exhaust flow path. Upon completion of this design change, the appropriate sections of the FSAR will be updated. The technical justification for this modification is demonstrated in the fact that system analysis and performance tests on operating BWR's has shown satisfactory system operation. The system tests showed that system pressure transients, control rod drive setting times, and control rod drive speeds were all unchanged. The tests also showed that all systems functions performed normally.

Responses to:

Core Performance Branch Questions
Reactor Fuels Section (231.3)
Reactor Physics Section (232.2 - 232.5)

QUESTION 231.3 (4.2)

The staff has established a new requirement for routine fuel surveillance; this is discussed in Revision 1 of Section 4.2 of the Standard Review Plan. Accordingly, submit a description of: (1) the method you propose to use to detect fuel rod failures while at power; and (2) the post-irradiation fuel surveillance program for the WNP-2 facility.

RESPONSE

"Issuance of the standard review plans (SRP) post date the WNP-2 construction permit. Therefore, no attempt was made to design the plant to the requirements of the SRPs. The FSAR was prepared using Revision 2 of Regulatory Guide 1.70 as much as practical for a plant of its vintage, with assurance from the NRC management that compliance with this regulatory guide assured submittal of all necessary licensing information.

As documented in a letter of August 5, 1977 from G. G. Sherwood to E. G. Case of the NRC the SRPs constitute a substantial increase in the information required just to describe the degree of compliance of various systems. This increase in turn represents a substantial resource expenditure which is unjustified and which could cause project delays if required of this and other projects. As stated in the reference letter, General Electric believes that SRPs should be applied to FSARs only to the extent that they were required in the PSARs.

General Electric believes the above position, which is the essence of a directive from Ben C. Rusche, Director of Nuclear Reactor Regulation to the NRC staff, dated January 31, 1977 is the appropriate procedure for review of the WNP-2 FSAR."

EC:cas/49C7

ESTION: 2.2

Provide values of the azimuthal peaking factor and the factor assumed to cover analytical uncertainties referred to in footnote 2 of Table 4.3.5 of the FSAR. Indicate how this peaking factor was calculated.

RESPONSE:

The azimuthal peaking factor is derived from the results of a two dimensional transport calculation. The two dimensional analysis models the reactor bundle pattern in an r,θ geometry. Fluxes were calculated at the cylindrical core shroud surrounding the core. It is expected that the peaking factors at the shroud are greater than the peaking factors at the vessel wall. The peaking factor used for the Hanford plant was for an 848 bundle core. The numerical value of the factor is 1.5 and the factor represents the peak angular flux divided by the mean angular flux.

In addition to the angular peaking flux, a safety factor of 1.5 was used. This factor was applied to ensure that the predicted values are conservative.

Q 232.3

The Rod Block Monitor (RBM) setting of 107 percent shown on page 15.4-8 of the FSAR is inconsistent with the value given in Table 15.4-2. Clarify this apparent discrepancy.

Response:

The RBM setting of 106 percent is the correct value. The referenced text on page 15.4-8 is being changed to read as follows:

> Settings are 106%, 98% and 90% of initial, steady state, operating power at 100% flow.

*The modified FSAR page 15.4-8 is attached here.

e. The operator is assumed to ignore all warnings during the transient.

f. Of the four LPRM strings nearest to the control rod being withdrawn, the two highest reading LPRM during the transient are assumed to have failed.

g. One of the two instrument channels is assumed to be bypassed and out-of-service. The A and C LPRM chambers input to one channel while the B and D chambers input to the other. The channel with the greatest response is assumed to be bypassed.

The conservative assumptions indicated above provides a high degree of assurance that the transient as analyzed bounds all RWE which could possibly occur. Table 15.4-2 presents the other parameters used in the analysis of this event.

15.4.2.3.2.1 RBM System Operation

The RBM system minimizes the consequences of a RWE by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. The highest trip level is set so that the safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. 106%, 98%, and 90% Settings are ~~105%, 99% and 91%~~ of initial, steady-state, operating power at 100% flow. The trip levels are automatically varied with reactor coolant flow to protect against fuel damage at lower flows. The variation is set to assure that no fuel damage will occur at any indicated coolant flow. The operator may encounter any number (up to three) of trip points depending on the starting power of a given control rod withdrawal. The lower two points may be passed up (reset) by manual operation of a push button. The reset permissive is actuated (and indicated by a light) when the RBM reaches 2% power less than the trip point. The operator should then assess his local power and either reset or select a new rod. The highest (power) trip point may not be reset.

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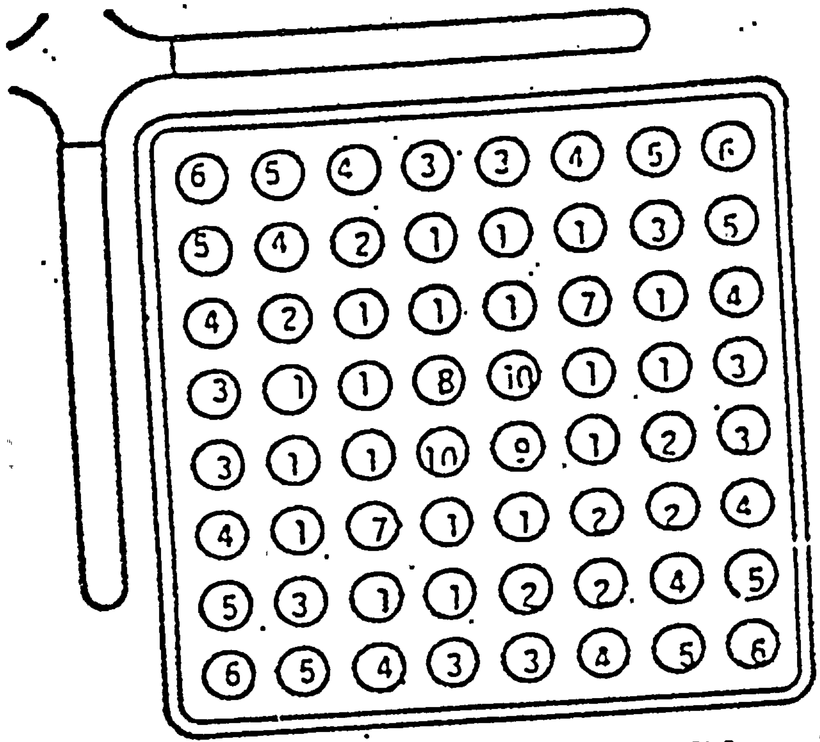
QUESTION 232.4

Indicate whether the average enrichment of 2.33 percent shown in Table 15.4-12 is correct or whether it should be 2.23 percent as indicated in the GE Topical Report NEDO-20944, "BWR/4 and BWR/5 Fuel Design", October 1976.

RESPONSE

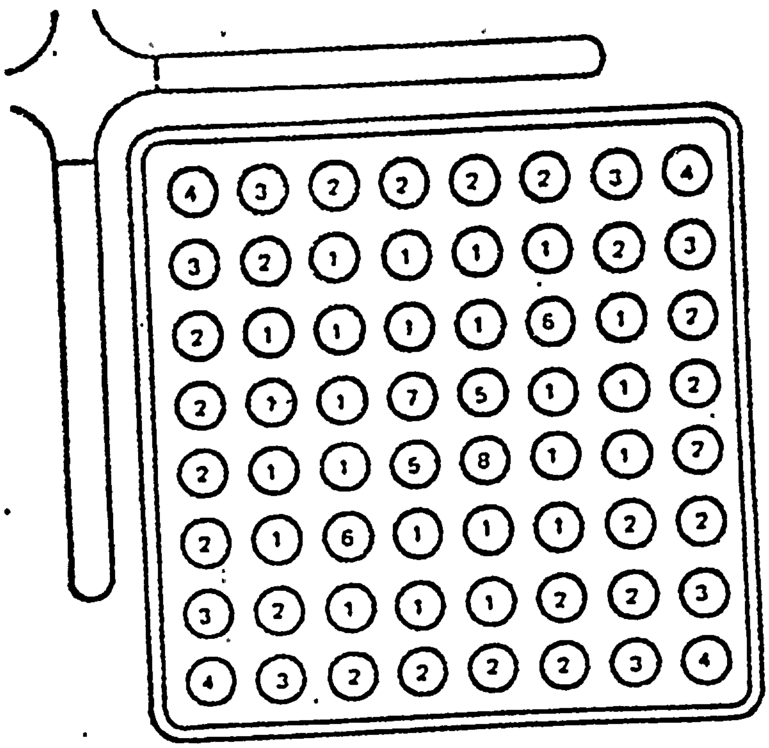
The core average enrichment of Hanford 2 is 1.88 percent, and the core is composed of bundles with average enrichments of 2.19 w/o, 1.76 w/o, and .711 w/o (natural uranium). The corresponding lattice enrichments, i.e., neglecting the natural uranium at the top and bottom of the bundles are 2.33, 1.83, and .711. Figures 3-3 and 3-5 of the GE Topical Report NEDO-20944-P, "BWR/4 and BWR/5 Fuel Design" dated October 1976, are outdated and the updated Figures are attached here. In addition, Table 15.4-12 (Page 15.4-49) is not referenced in the FSAR, does not apply to WNP-2, and will be deleted from the FSAR.*

*Draft pages attached.



SEE FIGURE 3-5 FOR AXIAL DISTRIBUTIONS

Figure 3-3. Rod Type Designations for Enrichment and Gadolinia Distributions in the High Enrichment, 2.19 wt% U^{235} Bundle (GE Company Proprietary)



SEE FIGURE 3-6 FOR AXIAL DISTRIBUTIONS

Figure 3-4. Rod Type Designations for Enrichment and Gadolinia Distributions in the Medium Enrichment, 1.76 wt% U^{235} Bundle (GE Company Proprietary)

2 of 5

DESIGNATION	ENRICHMENT (WT %)	Gd (WT %)
A	3.0	–
B	2.6	–
C	2.2	–
D	2.0	–
E	1.7	–
F	1.3	–
G	1.7	5
H	1.7	4
I	1.7	2
J	0.711	–
WATER	–	–

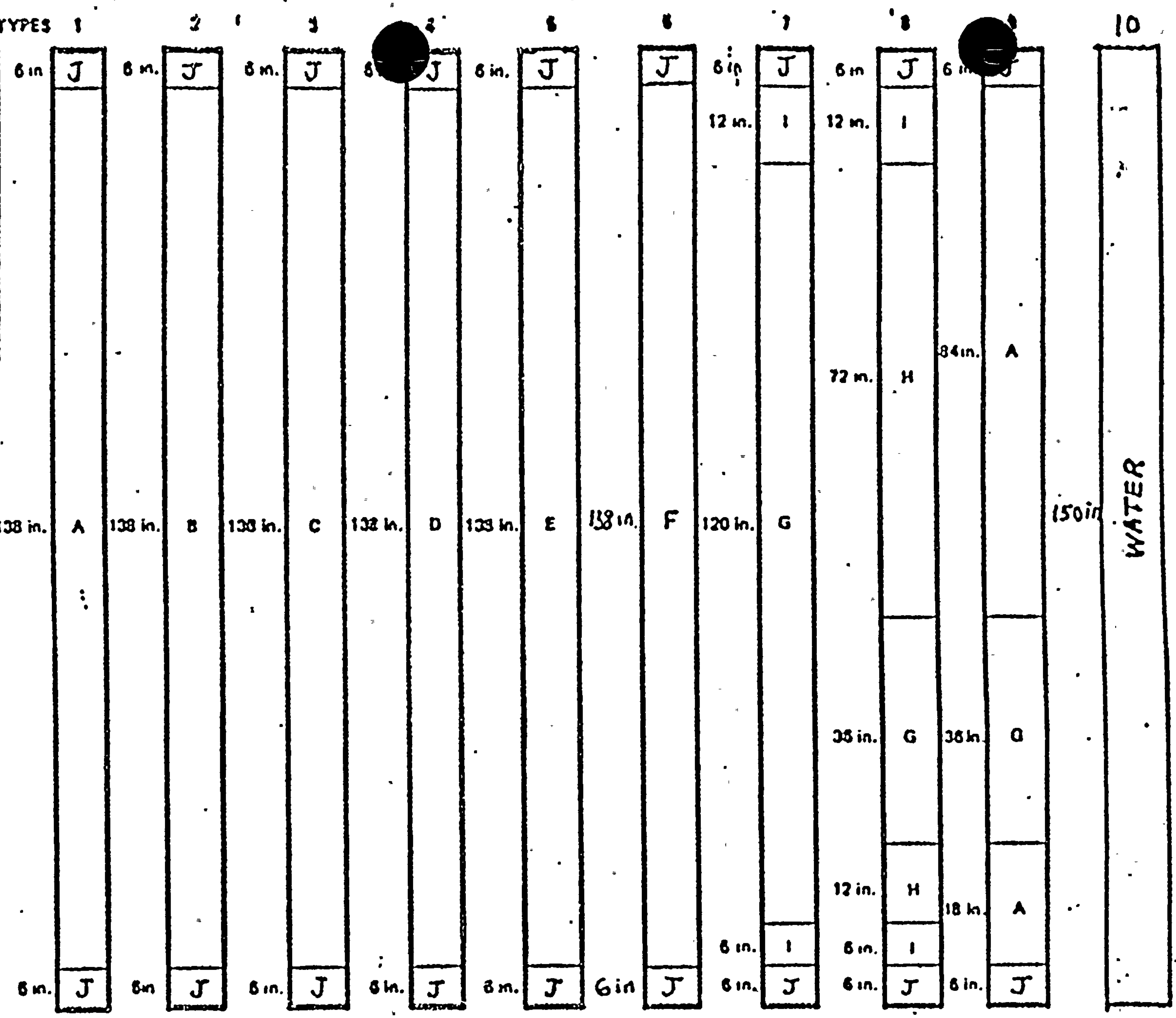


Figure 3-5. Axial Enrichment and Gadolinia Distribution, High Enrichment Bundle (GE Company Proprietary)

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TABLE OF CONTENTS (Continued)

TABLES

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TABLE 15.4-12

DOPPLER REACTIVITY COEFFICIENT DURING COLD STARTUP
FOR PREDOMINANT FUEL AT BEGINNING OF CYCLE

(2.33 Average Enrichment, 2Gd (5 wt% Gd_2O_3), 1Gd (4 wt% Gd_2O_3)

Average Fuel Temperature °C	Doppler Coefficient (10^{-5}/°C) Calculated	Tech. Basis
20	-2.21	-
500	-1.37	-1.28
1000	-1.07	-1.00
1500	-0.91	-0.85
2000	-0.81	-0.75
2500	-0.74	-0.67
2800	-0.70	-0.64

DELETED
(~~Blank Page~~)

5 of 5

QUESTION: 232.5

The Process Computer program developed by General Electric was developed for active core heights of 144". We note that you propose to use this program for the WNP-2 core whose fuel rods are 150".

(1) Indicate how this process computer program will be modified to accomodate the 150 inch fuel.

(2) Additionally indicate whether the travelling in-core probe (TIP) travel has been increased to cover your longer fuel and

(3) whether there are still 24 nodes or whether an additional node has been added.

(4) Describe how the core averaged and peak linear heat generation rates are calculated.

RESPONSE: 232.5

(1) and (3)

The present process computer treatment of fuel >144" is to totally ignore the amount of fuel >144" of each bundle. The power distribution and exposure distribution are maintained on a 24 node basis where each node is exactly 6" long. The amount of power generated above 144" is very small, and the resulting thermal limit evaluations are conservative since the actual nodal power is about 1/24 of the power generated above 144" lower than the process computer calculates.

In the near future a correction is planned to be incorporated to the process computer to correctly calculate thermal limits for fuel >144".

First we will calculate the power in each fuel bundle above 144"

$PTOP_{Bundle} = P(L,J,24)*ENDPF(ITYP)$

P(L,J,24) is the 24th nodes power

ENDPF(ITYP) is a constant, (fuel type dependent), which will be defined as the fraction of 24th nodes power generated above 144".

Next we sum over all bundles $\sum PTOP_{Bundle}$

Next we calculate a correction factor to be applied to thermal limit evaluations only.

$$PTOPF = (CTP - \sum PTOP_{Bundle})/CTP$$

The power distribution and exposure distribution will still be stored on a 24 node basis, 6 inch nodes with no corrections (PTOPF) supplied.

RESPONSE: 232.5

(2)

The TIP travel will not be increased to 150" from 144".

(4)

a) No explicit calculations for core average linear heat generation is made by the Process Computer

b) The peak linear heat generation rate is determined by searching all nodes for the maximum value of

$$\frac{P(L,J,K)*FLOP(L,J,K)*1000}{NRB(ITYP)*\Delta Z}$$

where;

P(L,J,K) = Fuel segment power, MW

FLOP(L,J,K) = Maximum rod power/average rod power in cross-section of fuel segment L,J,K (Local Peaking Factor)

NRB(ITYP) = the number of fuel rods/bundle type

ΔZ = 0.5 ft.

[illegible]

Responses to:

Accident Analysis Branch Questions
(312.15 - 312.19)

Q. 312.15
(6.1.2)

You indicated in your response to Item 312.5 of the first acceptance review that you will qualify protective coatings to the requirements of ANSI N101.4-1972. However, Regulatory Gide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," June 1973, endorses ANSI N101.4-1972 on the condition that ANSI N45.2-1971 be used in conjunction with it. Accordingly, indicate your intended degree of compliance with the recommendations of Regulatory Guide 1.54 in this regard.

RESPONSE

The protective coating contractor is required to have both a Quality Assurance program and procedures which meet the requirements of ANSI N45.2, "Quality Assurance Requirements for Nuclear Power Plants." This meets the requirements of Regulatory Guide 1.54.

Q 312.16

Provide an estimate, including your basis, of the total amount of hydrogen and methane gases that can be generated by the radiolytic and chemical decomposition of organic materials and protective coatings under the conditions which would exist following a design base accident (i.e., a postulated loss-of-coolant accident). Your estimate should be limited to those materials and coatings that would be directly exposed to the containment atmosphere.

RESPONSE:

The estimate of the total amount of hydrogen and methane gas that can be generated by the radiolytic and chemical decomposition of organic materials and protective coatings under the conditions of a postulated loss-of-coolant accident ~~are explained~~ is discussed in the answers to Question ~~00248~~ 022.048.

Q022.048

You state in Section 6.2.5.3.1.3 of the FSAR that the corrosion of aluminum, zinc, and zinc base paints located either in the drywell or in the suppression chamber were determined to be insignificant. However, we have determined that a potential hydrogen release from the corrosion of zinc following a postulated loss-of-coolant accident should be considered in the analysis of the total hydrogen production and accumulation within the containment. Accordingly, provide the following information:

a. Provide the corrosion rate as a function of temperature for all materials in the containment that could become a source of hydrogen due to corrosion.

b. Describe how the corrosion rates assumed for the materials identified in Item (a) were established: Identify the experimental data base, including the appropriate references, and discuss the conservatism in the applicability of the data in view of the calculated environmental conditions following a postulated loss of coolant accident.

c. Provide the mass and surface area of zinc paint and galvanized steel and other corrodible materials in both the drywell and the wetwell.

d. Provide a graphic representation of the total hydrogen concentration inside the containment as a function of time with (1) no hydrogen recombiners operating; (2) one recombiner operating; and (3) both recombiners operating.

e. Provide a graphic representation of the contribution of each source of hydrogen as a function of time.

f. Describe the periodic surveillance that will be done to demonstrate the operability of the hydrogen recombiners and the backup purge system.

g. Identify the location of (1) the hydrogen sample points in the drywell and the suppression chamber; and (2) the suction and discharge points of the combustible gas control system with respect to nearby structures and equipment.

RESPONSE:

A review of tests conducted to date on ~~A~~luminum, ~~Z~~inc or ~~Z~~inc coatings, indicates that several factors which would tend to mitigate the evolution of hydrogen following a postulated loss-of-coolant accident have not been reported or have not been investigated. A brief explanation, therefore, is required to substantiate the rationale for the conclusions drawn in this response.

Question 022.048 asks a question with respect to the corrosion of aluminum and the subsequent evolution of hydrogen. The water chemistry of WNP-2 is such that the water is free from additives and is neutral, i.e., a pH of 6.5-7.5.

With reference to Aluminum, Uhlig[1] states: - "Aluminum base alloys are not appreciably affected by distilled water even at elevated temperatures (up to 180°C (350°F) at least). Furthermore, distilled water is not contaminated by contact with most aluminum base alloys."

Uhlig[2] states: - "Condensate from steam boilers, if free from carryovers of water from the boiler, is similarly inert to aluminum base alloys. Thus, either wrought or cast aluminum alloys are used successfully for steam radiators as unit heaters. Where aluminum alloys are used it is desirable to install suitable traps in the steam lines, since entrapped boiler water, especially if alkaline water treating compounds are employed, may be corrosive."

Uhlig[3] states: - "Steam causes a definite protective white film to form ~~as~~ on aluminum alloys. This film is highly protective at temperatures up to 180° to 350°C (350° to 500°F). At temperatures above this range, under some conditions at least, the steam reacts with aluminum with the formation of aluminum oxide and hydrogen."

Experimental data from the aforementioned references indicate that aluminum and aluminum alloys are nonreactive with pure water and/or steam at temperatures up to and including 500°F. Aluminum rapidly forms a protective oxide film, in oxygen containing atmospheres, which is insoluble in neutral water or steam. Since the containment ~~area in the BWR under discussion~~ is noninerted, there is ~~a~~ free access to oxygen and has been throughout ~~the~~ construction. ~~phases.~~ The oxygen has reacted with the aluminum to form the protective tight adherent water insoluble and nonreacting film, which eliminates the case of hydrogen evolution at the temperature and/or environment present during or following a postulated loss-of-coolant accident. ~~in a BWR.~~

ring
eration,

Question 022.048 also addresses the corrosion of zinc and zinc base paints and the evolution of hydrogen following a loss-of-coolant accident.

Hubbell and Finkeldy[4] stated: "Like several other metals which exhibit marked resistance to corrosion in the atmosphere bright zinc rapidly tarnishes when first exposed, forming a smooth tightly adherent protective film. The film is apparently a combination of zinc oxide, zinc carbonate and zinc hydroxide. It is not readily soluble in ordinary atmospheric waters nor easily destroyed by other atmospheric agencies.

The film varies in thickness depending upon the exposure conditions, probably reaching a maximum thickness of .0003 in. If removed or worn thin by abrasion, it is renewed in a few days to its original thickness."

McKay and Worthington[5] state in their chapter on Defining Non-Corrosive Neutral Range of Aqueous Solutions: "Zinc has useful resistance only in a relatively narrow; neutral range of solution; this resistance being due in the simplest case to a protective film of hydrate, abetted in the case of impure solutions by other precipitated corrosion products and compounds deposited from solution. The hydrate is soluble on both the acid and alkaline sides of this neutral zone.

Work by Roetheli, Cox and Litteral has very effectively drawn the limits of the neutral, hydrate-forming zone in tests in distilled water with hydrochloric used to throw the solution acid and/ sodium hydroxide alkaline. The solutions were kept rather strongly agitated." Figure ~~1b~~ depicts the results.
022.048-1

McKay and Worthington[6] explain that: "The hydrate is seen to have been most protective between the neutral point of pH 7 and pH 12.5. The fact is brought out that the exact shape and location of this curve is typical probably only of the particular set of conditions under which the tests were made. Factors such as agitation, aeration, salts in the solution, and temperature, in conjunction with hydrogen-ion concentration, affect the characteristics of film formation. The investigators conclude in a universal sense the condition of low or negligible corrosion probably lies between pH values of 6 and 8 as a minimum and 11 as a maximum."

Cox [7] in a paper titled "Effect of Temperature on the Corrosion of Zinc", established the effect of temperature on the behavior of zinc in distilled water. The specimens were constantly in motion in the solution, and the solution was aerated with a stream of unwashed air bubbles. The duration of the test was 15 days.

The results of their investigation are shown in Figure 022.048-2.

Examination of the zinc hydrate film showed that the strong increase in corrosion coincided with a change in the nature of the film from an adherent gelateneous state to a non-adherent granular state.

Table 022.048-1 gives the results of the change in the film structure of zinc with respect to temperature.

TABLE 022.048-1

EFFECT OF TEMPERATURE ON THE CORROSION OF ZINC IN DISTILLED WATER *

Temp. °F	Temp. °C	Corrosion Rate $mg/dm^2/day$	mil/yr	Appearance of Corrosion Film
68	20	3.9	.78	Gelatenous, very adherent
122	50	13.7	2.74	Less gelatenous, adherent
131	55	76.2	15.2	Mostly granular, nonadherent
149	65	577.0	115.4	Granular to flaky, nonadherent
167	75	460	92.0	Granular flaky, nonadherent
203	95	58.7	11.7	Compact dense, nonadherent
212	100	23.5	4.7	Very dense and adherent

* Rolled high-grade zinc, immersed for 15 days in water aerated by air bubbles. Specimens rotated at 56 RPM.

McKay and Worthington,[8] site work by Bengaugh and Hudson which says: "Of the neutral solutions, distilled water is relatively high in its action on zinc. An idea of rates may be gained by the following data, from 24 hour tests at room temperature on cast 99.97% zinc."

Distilled water	mg. per sq. dm. per day
Quite suspension	53; 76
Air-agitated	167; 199
Agitated with Carbon Dioxide	143
Agitated with air	33

The authors state that a fair average range of rates in distilled water would be 30 to 200 mg./sq. dm. per day.

~~Section 1.2 Question 312.16 requests that we provide~~ An estimate of the total amount of hydrogen and methane gases that can be generated by the radiolytic and chemical decomposition of organic materials and protective coatings under conditions which would exist following a postulated loss-of-coolant accident is considered below.

~~The answer to this question is extremly difficult to even evaluate.~~ There is, to our knowledge, no published experimental data which states the amount of hydrogen attributable to the effects of a LOCA on specific coatings or organic materials. There is also no published data which gives the amount of methane produced from the decomposition of organic materials in a non-inerted environment at the temperature and radiation levels resulting from a loss-of-coolant accident. All of the organic coating materials used within the containment have been subjected to the test requirements stipulated in ANSI N-101.2 and ANSI N-512 and to an accumulative dose of 1×10^9 rads, at the Oak Ridge National Laboratory. Test results show the coatings were intact with no defects.

Mattson[9] states:

"Decomposition of Organics: A substantial amount of organic materials is used in protective coating systems, including those over zinc-based primer paints inside PWR and BWR containments. When exposed to the LOCA environment (high temperature, chemical, and radiation fields), these organic materials undergo a process of decomposition to form hydrogen and hydrocarbons. The Accident Analysis Branch (DSE) has estimated the resultant hydrogen and hydrocarbon concentrations resulting from the radiolytic decomposition of organics and the thermal and chemical reaction of organic coatings on concrete surfaces. Assuming a conservatively integrated radiation exposure of 10^8 rads, the Accident Analysis Branch (AAB) estimates the hydrogen concentration due to radiolytic decomposition of organic coatings to be less than 0.4% for PWR's and less than 0.2% for BWR's. For hydrogen generation due to thermal and chemical reaction of organic coatings on painted concrete surfaces the AAB estimates the resultant hydrogen concentration to be less than 0.3% for PWR's and less than 0.2% for BWR's. If we sum these hydrogen contributions from organic materials which were heretofore not included in our analysis, the additional hydrogen represents roughly a 10% increase in the hydrogen generated from all sources previously considered, i.e., zirconium water reaction, radiolysis of water, and oxidation of zinc with its organic top coat during the post-LOCA period.

Since there will be a large amount of water, relative to the amount of organic materials, it can be concluded that the hydrogen gas generated from radiolysis of water should dominate that from decomposition of the organic materials."

The following paragraphs address Question 022.048 item by item.

a. The corrosion rates as a function of temperature for all materials in the containment that could become a source of hydrogen due to corrosion are:

1. Zinc - applies to galvanize and zinc base paints

Temp. ^{o}F	Corrosion mg/dm.^{2}/day	Corrosion oz/sq.ft./day
68	3.9	.0013
122	13.7	.0045
131	76.2	.0250
149	577	.1892
167	460	.1508
203	58.7	.0192
212	23.5	.0077

2. Aluminum - applies to reflective insulation around the RPV and piping

Temp: 70^{o}F to 340^{o}F - Nonreactive, will not produce hydrogen at temperature indicated.

3. Organic Materials - Consists of all organic coating materials on steel and concrete in drywell and wetwell. No specific reaction rates have been published wherein the function of temperature on the corrosion rates of organic materials has been addressed. Most data addressed the loss in or deterioration of physical properties such as strength, durometer and the like. All of the materials used for coating of concrete or steel have been subjected to the required tests stipulated in ANSI N-101.2 and ANSI N-512 in accordance with Bechtel Corporation's specifications CP-951 and CP-956, by the Analytical Chemistry Division of Oak Ridge National Laboratory. The test reports indicate that there were no defects in the coating materials, i.e., there were no signs of chaulking, flaking, cracking, delamination or blistering beyond the requirements of the acceptance criteria of the ANSI Standards. Reference 9 states that "The Accident Analysis Branch (DSE) has estimated the resultant hydrogen and hydrocarbon concentration resulting from radiolytic decomposition of organics and the thermal and chemical reaction of organic coatings on concrete surfaces. Assuming a conservatively integrated radiation exposure of 10^{8} rads, the Accident Analysis Branch (AAB) estimates the hydrogen concentration due to radiolytic decomposition of organic coatings to be less than 0.4% for PWR's and less than 0.2% for BWR's. For hydrogen generation due to thermal and chemical reaction of organic coatings on painted concrete surfaces the AAB estimates the resultant hydrogen concentration to be less than 0.3% for PWR's and less than 0.2% for BWR's."

For the specific BWR in question, we are speaking of a possible 0.4% total hydrogen and hydrocarbon concentration.

4. Metal water reactions and radiolytic decomposition of water are addressed in ~~the FSAR Chapter~~ 6.2.5.

b. The corrosion rates for the materials identified in item (a) of the question were established as follows:

1. Zinc - Since no definitive data with respect to the corrosion rate of zinc in a BWR environment, containing neutral water without additives, has been reported in the test conducted by Oak Ridge National Laboratory, Franklin Institute Research Laboratories or Brookhaven National Laboratory, it was necessary to refer to the investigations conducted by other recognized corrosion experts and institutions. 022.048 Figure-2[11] and the accompanying 022.048 Table-1[12] explains the investigations conducted to determine the corrosion rate of zinc (rolled high grade) immersed for 15 days in distilled water aerated by air bubbles while rotating the specimens at 56 RPM. The table which appears in the referenced publications shows the actual corrosion in milligrams per square decimeter per day and mils per year at the various temperatures. The data which was used as a basis for calculating the evolution of hydrogen is actual measured data.

   The premise that all of the metal which corrodes will react to produce a stoichiometric quantity of hydrogen is ultra conservative, since there are other competing reactions which will produce zinc carbonate and zinc oxide. In our evaluation we determined the highest corrosion rate of zinc occurred at 149°F and resulted in a rate of 577 mg/dm²/day or .189 oz/sq.ft./day[13]. In our analysis we used this maximum amount as the corrosion rate, at temperature, in the containment. It can be determined from the published data that this is conservative by a factor of more than 27, if we consider that at 212°F the corrosion rate is 23.5 mg/dm²/day or .0077 oz/sq.ft./day and that at 68°F it is 3.9 mg/dm²/day or .0013 oz/sq.ft./day. The data of Figure-2 and Table-1 show that at the temperature of 149°F the maximum rate occurs and that the rate falls sharply with temperature increase so that at a temperature of 340°F it would be below that shown at 212°F.

022.048

2. Aluminum - Our search of the literature (1), (2), (3) clearly indicates that aluminum will not corrode or produce hydrogen since the pH of the water is in the neutral range and the temperature is below that required to produce hydrogen.

3. Organic Coatings and Materials - ~~We searched the~~ The literature was searched in an effort to determine if there was any firm data which showed that the organic coatings and materials used would produce hydrogen as a result of a postulated loss-of-coolant accident. ~~We found that~~ It has been postulated that organics do form hydrocarbons as a result of radiation, but this theory is based on the fact that there is a loss of physical properties as radiation exposure increases. We have stated ~~in the FSAR~~ that the organic top coats and coatings used within containment were subjected to the test requirements of ANSI N-101.2 and ANSI N-512 to an accumulative dose of 1 x 10^9 rads, with a resulting no defects.

Mattson[9] states that "The Accident Analysis Branch (DSE) has estimated the resultant hydrogen and hydrocarbon concentration resulting from the radiolytic decomposition of organics and the thermal and chemical reactions of organic coatings on concrete surfaces. Assuming a conservatively integrated radiation exposure of 10^8 rads", the resultant concentration is considered to be less than 0.2% for BWR's. He further states that ~~it is estimated that~~ thermal and chemical reactions are estimated to produce less than 0.2% of hydrogen for BWR's.

c. The mass and surface area of zinc paint, galvanized steel, aluminum and organic coatings is as follows:

1. Zinc Paint

280,853 sq. ft. - 18,367.8 lbs. of zinc in drywell and wetwell above the water level

2. Galvanize

67,483 sq. ft. - 3018.3 lbs. of galvanize in drywell and wetwell above the water level

3. Aluminum

460,700 sq. ft. - 32,374 lbs. around reactor pressure vessel and piping

4. Organic Topcoats

On steel 400,000 sq. ft. - approx. 187,500 lbs.

On concrete 33,000 sq. ft. - approx. 24,750 lbs.

~~Question .0222048, Part d~~

d. The graphic representation of the total hydrogen concentration inside containment as a function of time is shown in Figure 4.

~~Question ,0022048, Part e~~

e. The graphic representation of the contribution of each source of hydrogen ~~on~~ as a function of time is shown in Figures 3 and 4.

~~Question 00248, Part f~~

f. The periodic surveillance that will be done to demonstrate the operability of the hydrogen recombiner and the backup purge system is ~~explained~~ discussed in ~~Chapter 6 of the FSAR.~~ 6.2.1.1.8 and 6.2.5.4.

~~Question 00248, Part g~~

g. The location of the hydrogen sample points in the drywell and the suppression chamber and the suction and discharge points of the combustible gas control system with respect to nearby structures and equipment has been answered ~~in the FSAR~~ in response to Question ~~022.025.~~ 022.25.

See, in addition, revised Section 6.2.5 pages of the FSAR.*

*draft revised pages attached.

BIBLIOGRAPHY TO 022.048

1. The Corrosion Handbook - H.H. Uhlig p. 42, Pub. J. Weley & Sons

2. The Corrosion Handbook - H.H. Uhlig p. 43, Pub. J. Weley & Sons

3. The Corrosion Handbook - H.H. Uhlig p. 617, Pub. J. Weley & Sons

4. Corrosion Resistance of Metals and Alloys - R. McKay & R. Worthington p. 168, Pub. Reinhold Pub. Corp.

5. Corrosion Resistance of Metals and Alloys - R. McKay & R. Worthington p. 159, Pub. Reinhold Pub. Corp.

6. Corrosion Resistance of Metals and Alloys - R. McKay & R. Worthington p. 169, Pub. Reinhold Pub. Corp.

7. Corrosion Resistance of Metals and Alloys - R. McKay & R. Worthington p. 159, Pub. Reinhold Pub. Corp.

8. Corrosion Resistance of Metals and Alloys - R. McKay & R. Worthington p. 163, Pub. Reinhold Pub. Corp.

9. U.S.N.R.C., Memorandum 10/17/78, R.J. Mattson, Director, Division of Systems Safety to R.S. Boyd, Director, Division of Project Managment (Available NRC Public Document Room)

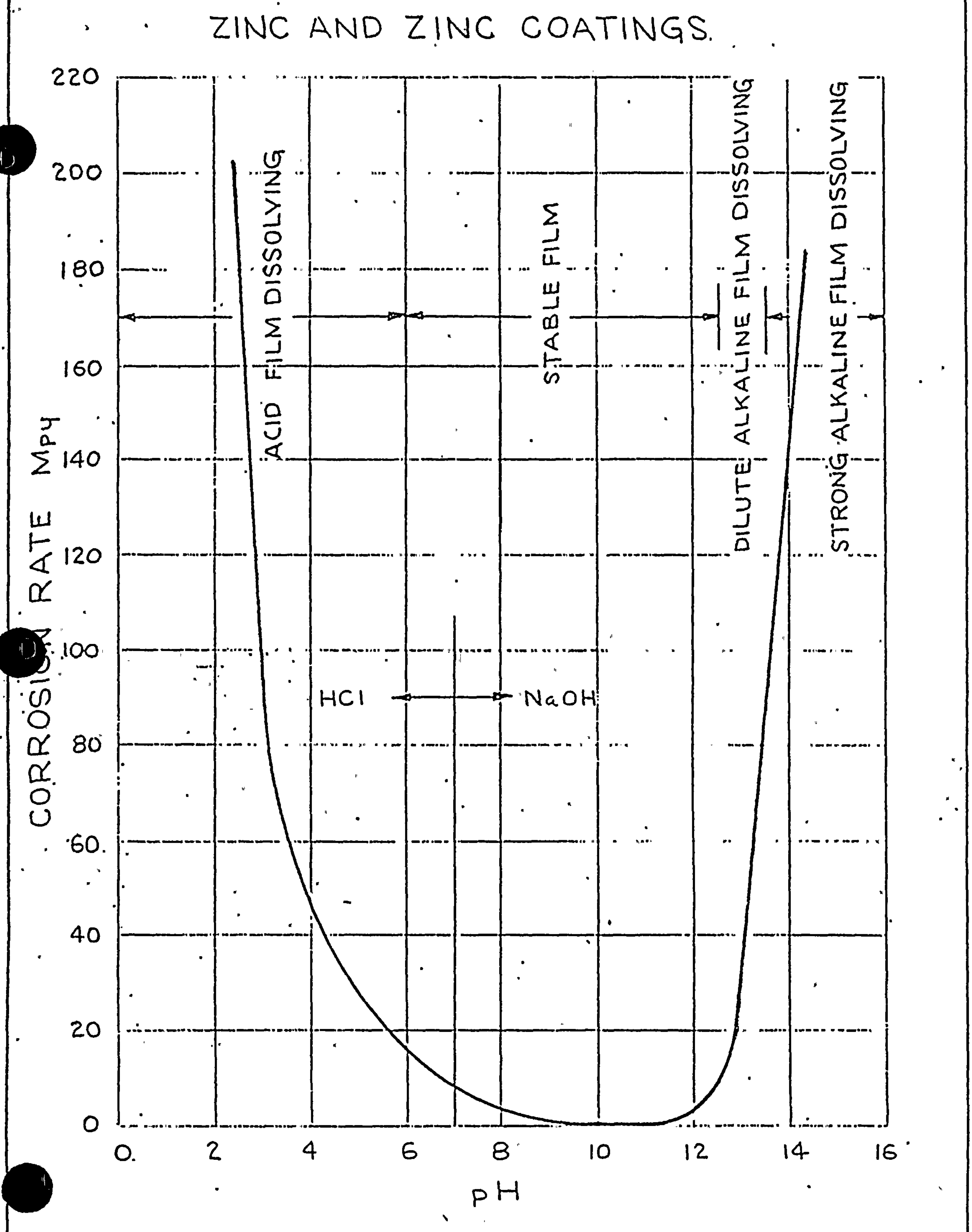
10. Fig. 022.048-1 p. 160 McKay & Worthington
    p. 13 Zinc: Its Corrosion Resistance Pub. Zinc Institute

11. Fig. 022.048-2 p. 161 McKay & Worthington
    p. 104 Zinc: Its Corrosion Resistance, Pub. Zinc Institute
    p. 232 LaQue & Copson - Corrosion Resistance of Metals and Alloys

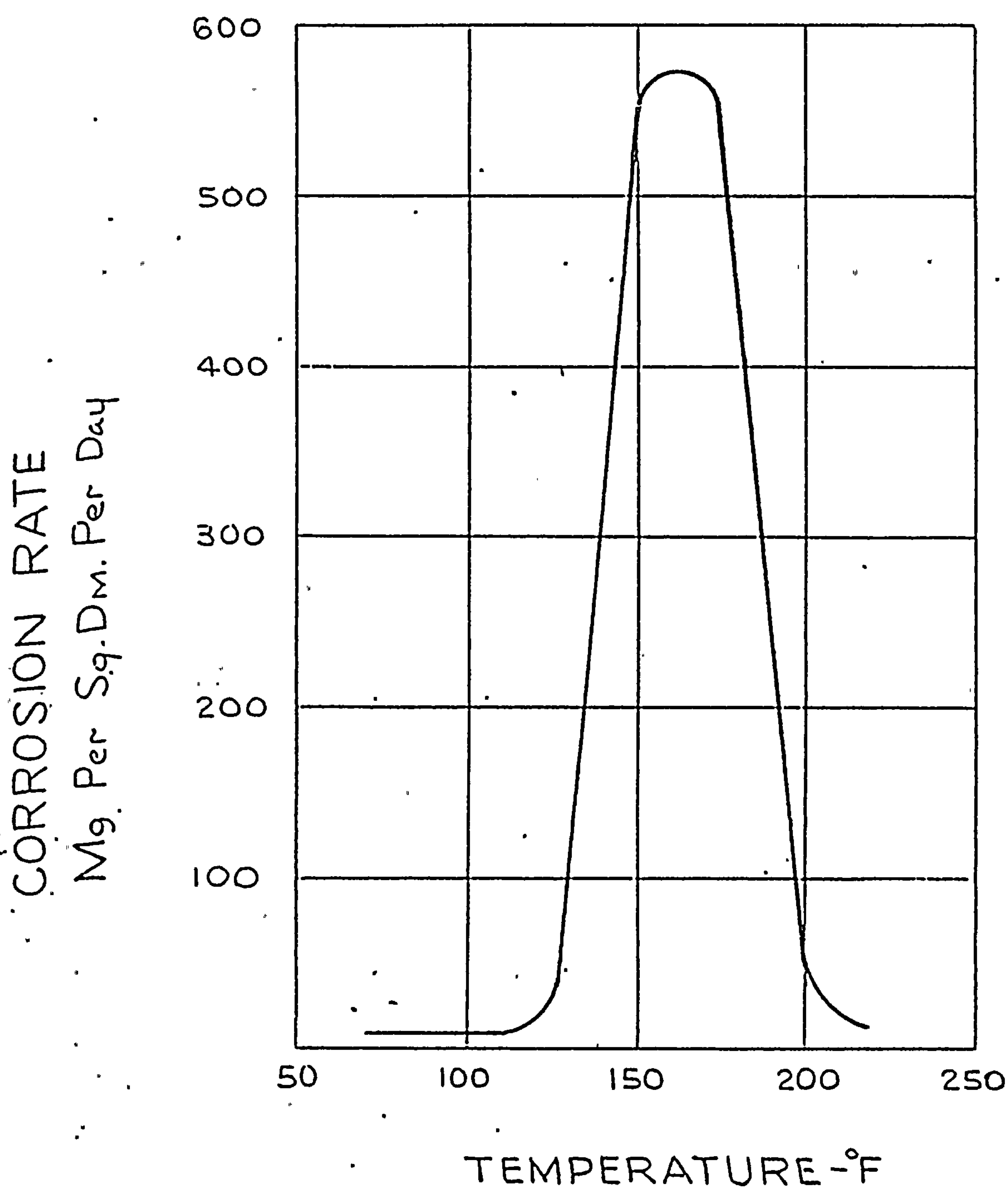
12. Table 022.048-1 p. 160 McKay & Worthington
    p. 233 LaQue & Copson - Corrosion Resistance of Metals and Alloys
    p. 103 Zinc: Its Corrosion Resistance, Pub. Zinc Institute

13. Conversion Factor H. H. Uhlig The Corrosion Handbook p. 1160 Table 19

14. Final Report F-C4290, Hydrogen Evaluation From Zinc Corrosion Under Simulated Loss-Of-Coolant Accident Conditions - Franklin Institute Research Laboratories 8/76



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	EFFECT OF pH ON CORROSION RATE OF PURE ZINC IN OXYGENATED SOLUTIONS	FIGURE 022.048-1



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ZINC vs DISTILLED WATER EFFECT OF TEMPERATURE	FIGURE 022.048- 2

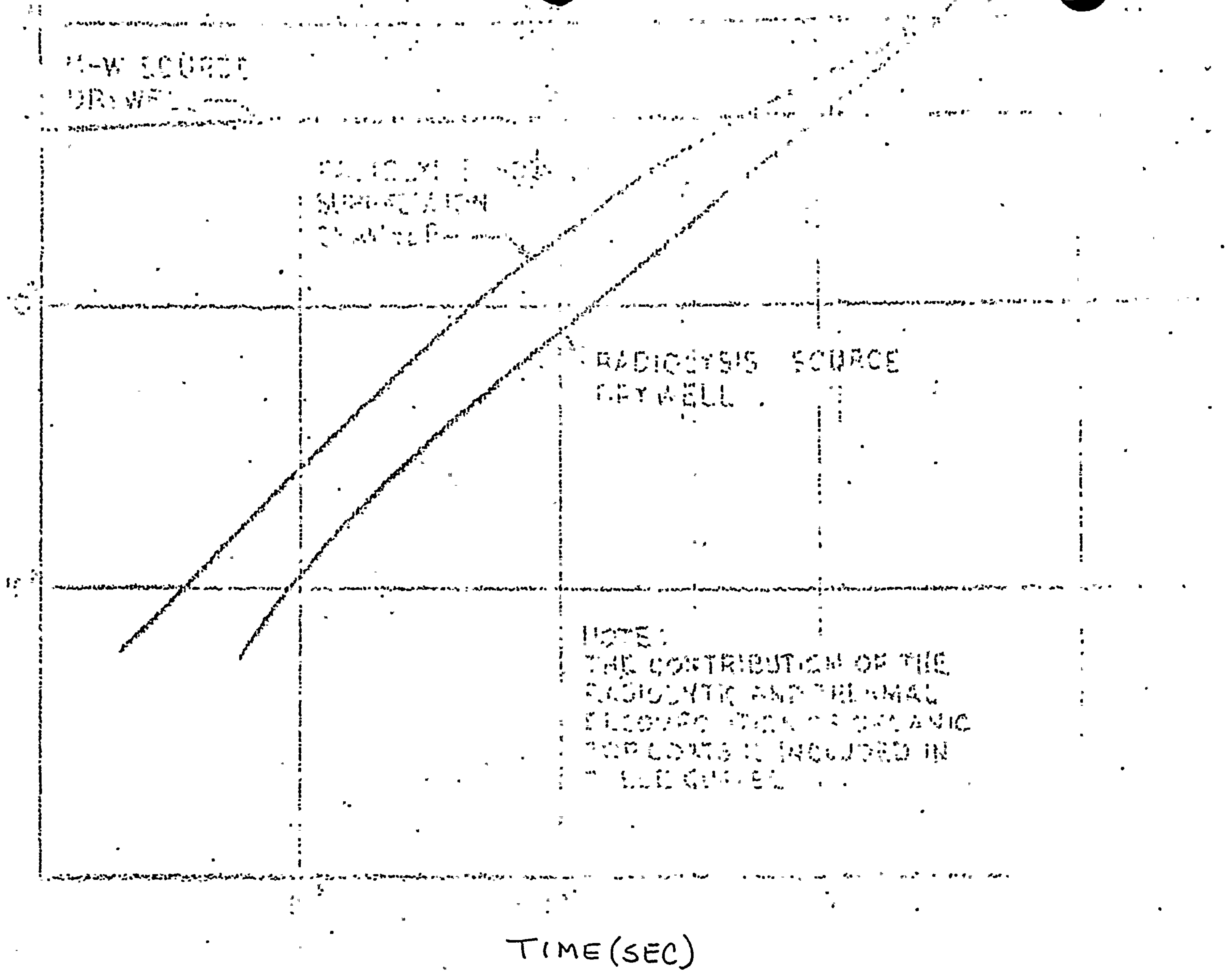


FIGURE 022.048-3

INTEGRATED HYDROGEN PRODUCTION AS A FUNCTION OF TIME (WITHOUT HYDROGEN RECOMBINING)

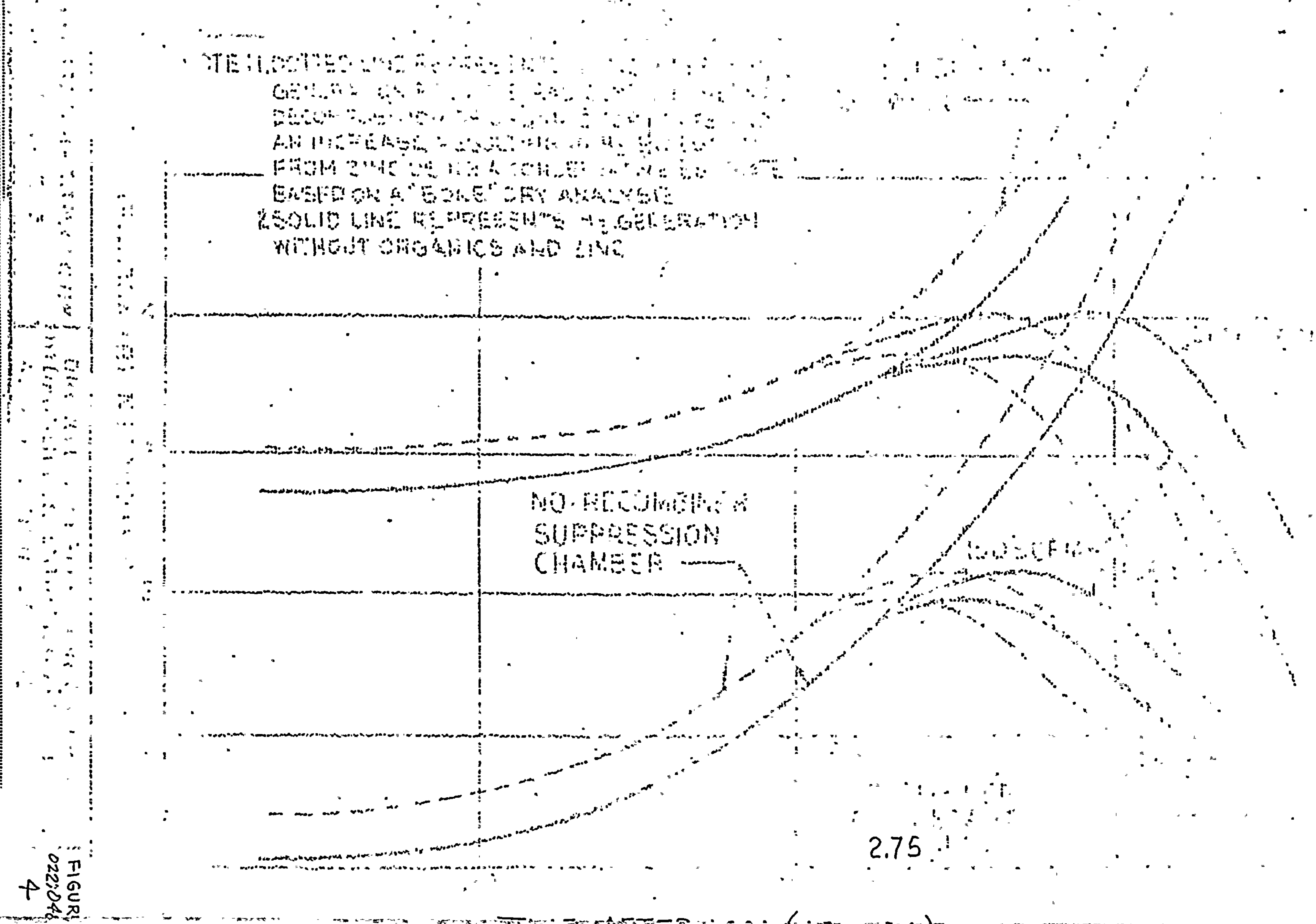


FIGURE 4

k. The system is designed to meet quality assurance, redundancy, power supply and instrumentation requirements for an engineered safety feature system.

l. Since the system is redundant and is not shared with other nuclear units, transportation of the hydrogen recombiners is not required.

m. Since all components of the system are redundant, a containment purge system as a backup is not required. A containment purge system used for other environmental controls is discussed in 6.2.1.1.8.

6.2.5.2 System Design

The containment atmosphere control system provides effective control of the hydrogen generated following a postulated LOCA. Piping and instrumentation for the system is shown in Figures 3.2-17, 3.2-15 and 3.2-6. Equipment details are given in Table 6.2-17.
The system consists of the following:

1. A hydrogen mixing system which operates to assure a well mixed atmosphere in both the drywell and suppression chamber. This system is the containment spray system and can be actuated approximately 10 minutes after the postulated LOCA.

2. A hydrogen concentration monitoring system measures the amount of hydrogen in the drywell and suppression chamber atmosphere.

3. Two 100 percent capacity hydrogen recombiners, one of which is manually initiated approximately 2.75 hours after the accident to preclude the hydrogen concentration from exceeding four percent by volume. The recombiners are catalytic type hydrogen oxygen recombiners.

6.2.5.2.1 Hydrogen Mixing System

The function of the hydrogen mixing system is to provide a well mixed atmosphere in the drywell and suppression chamber.

The cooling water supplied to the aftercooler is returned to the standby service water system. The cooling water supplied to the scrubber is discharged to the suppression pool.

All components of the containment atmosphere control system are redundant. Controls include the control panel located in the main control room and the local control panel for each recombiner located in environmentally suitable rooms in the reactor building. All of the functions necessary to control the system are located in the main control room.

6.2.5.2.4 Containment Purge

Since active and passive components of the containment atmosphere control system are redundant, containment purge as a backup system is not required.

6.2.5.3 Design Evaluation

Based on the assumptions of the model described below, it is calculated that the hydrogen concentration in the drywell eventually reaches 4% by volume approximately 10.0 hours after the postulated LOCA if the hydrogen recombiner is not in operation. The recombiner is started, however, 2.75 when the hydrogen concentration reaches approximately 3.5% by volume (hours after the postulated LOCA) to limit the hydrogen concentration below 4% by volume. Figure 6.2-26 shows the drywell and suppression chamber hydrogen concentration as a function of time, with and without operation of the hydrogen recombiner system at design capacity of 150 scfm and at 105 scfm, minimum flow required to maintain the hydrogen concentration below 4% by volume.

The determination of the time dependent hydrogen concentration in the drywell and suppression chamber atmospheres is based on a two-region model of the primary containment, a drywell and a suppression chamber atmosphere.

The drywell and suppression chamber free volumes contain air and water vapor at atmospheric pressure just prior to the postulated LOCA. Gases considered available for hydrogen dilution are the non-condensibles and water vapor present during normal operating conditions. Water vapor generated from blowdown is not considered. The radiolytic generation of free oxygen is added to the total inventory of gases. The pressure in containment is assumed to remain at atmospheric pressure and the temperature history of Figure 6.2-7 curve a, is used. The hydrogen contribution from zinc and organics took no credit for dilution.

and Decomposition

6.2.5.3.1.3 Corrosion of Containment Materials

and decomposition

The corrosion of containment materials was considered as a potential source of hydrogen. The corrosion of aluminum, zinc, ~~and~~ zinc base paints, located either in the drywell or suppression chamber was evaluated as a potential source of hydrogen. ~~It was determined that these potential sources were insignificant for the following reasons:~~

d the radiolytic and
emical decomposition
organic materials

~~a. The containment spray does not contain any chemical additives. The pH of the spray solution is 6.5 - 7.0.~~

~~b. Aluminum corrosion is negligible at a pH of 6.5 - 7.5.~~

~~c. The corrosion of both zinc and aluminum is highly temperature dependent. The duration of elevated post-LOCA temperature in the drywell and suppression chamber is short (See Figures 6.2-3, 6.2-12 and 6.2-15) and the magnitude is not sufficient to produce significant corrosion.~~

replace

6.2.5.4 Testing and Inspections

The hydrogen recombiners and the associated instrumentation are periodically inspected and tested to ensure reliable operation.

Each hydrogen recombiner system has been shop tested. Written test procedures and acceptance criteria were established for all tests. Test results were recorded in performance records. The full scale performance tests were accomplished by placing each unit in operation, starting the hydrogen recombiner and allowing atmospheric air, hydrogen and steam to flow through the unit. A flow of at least 155 SCFM was maintained throughout all tests. The simulated environmental conditions (temperature, pressure and hydrogen at 0.5 to 4% by volume) following a postulated LOCA (Figures 6.2-6 and 6.2-7, curve c) were used during these tests.

The manufacturer has also conducted a series of catalyst performance tests including the effect of iodine and methyl iodide.

The evaluation is included in the response to NRC question 022.048. The results are taken into account in Figures 6.2-26, 6.2-79 and 6.2-30.

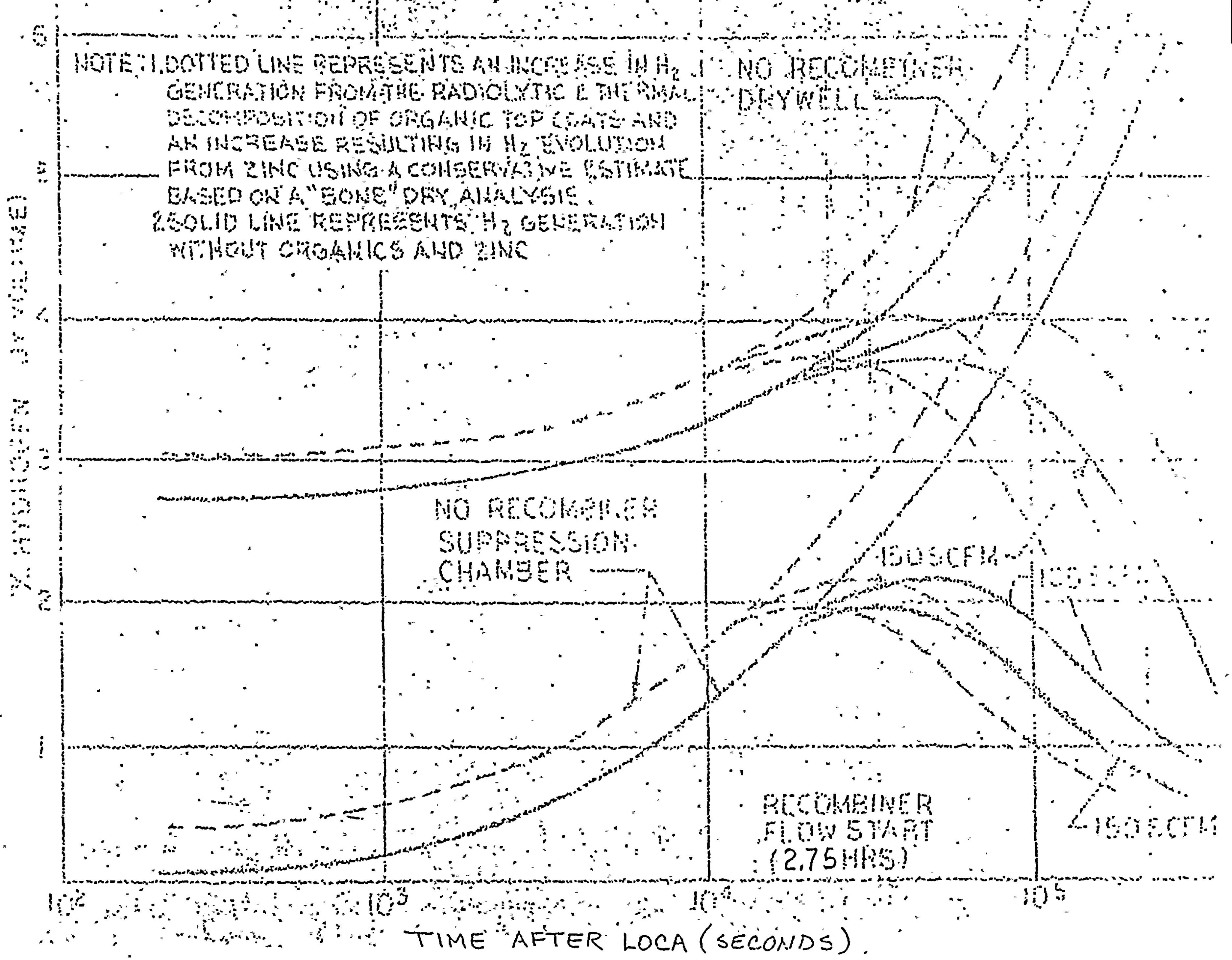
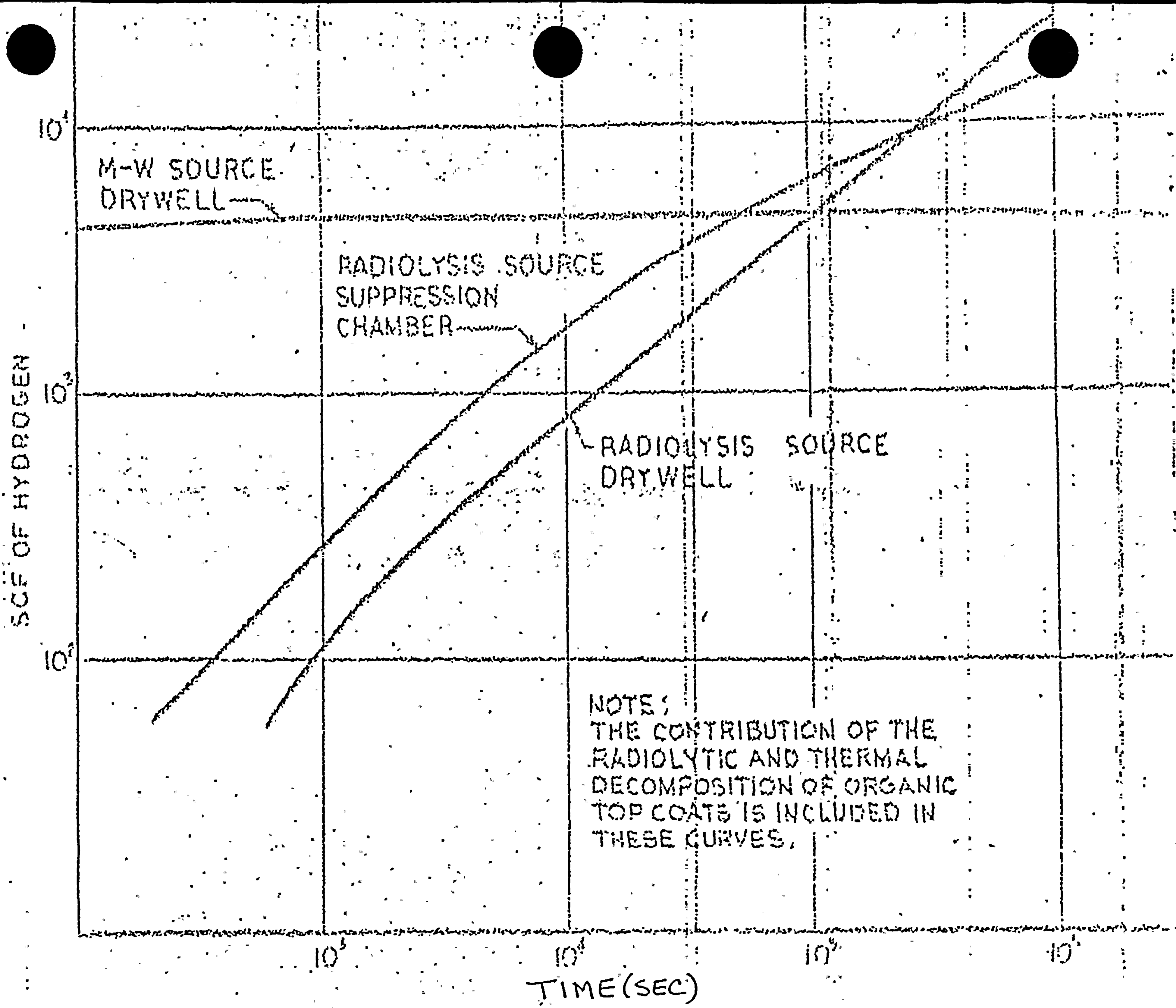

NOTE: 1. DOTTED LINE REPRESENTS AN INCREASE IN H_2 GENERATION FROM THE RADIOLYTIC & THERMAL DECOMPOSITION OF ORGANIC TOP COATS AND AN INCREASE RESULTING IN H_2 EVOLUTION FROM ZINC USING A CONSERVATIVE ESTIMATE BASED ON A "BONE" DRY ANALYSIS.
2. SOLID LINE REPRESENTS H_2 GENERATION WITHOUT ORGANICS AND ZINC
NO RECOMBINER DRYWELL
NO RECOMBINER SUPPRESSION CHAMBER
150 SCFM
150 SCFM
RECOMBINER FLOW START (2.75 HRS)
10^2
10^3
10^4
10^5
TIME AFTER LOCA (SECONDS)
1
2
3
4
5
6

FIGURE 6.2-26



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

INTEGRATED HYDROGEN PRODUCTION AS A FUNCTION OF TIME (WITHOUT HYDROGEN RECOMBINING)

FIGURE 6.2-30

Question 312.17

(15.6.5)

Your statement on Page 15.6-34 of the FSAR regarding the bypass leakage of 0.11 percent per day is ambiguous. Indicate whether this leakage is in addition to, or is part of, the primary containment leakage. The bypass leakage, which will be a major contributor to offsite doses, is not listed as an assumption in Table 15.6-12 of the FSAR. Correct this omission.

Answer

The statement given on Page 15.6-34 regarding bypass leakage provided a quantitative assessment of how much bypass leakage could hypothetically occur in addition to the primary to secondary containment leakage identified in 15.6.5.5.1.2 with the resulting radiological consequences not exceeding 10 CFR 100 criteria. This hypothetical leakage value is 0.11%/day. The primary to secondary containment leakage would be processed by SGTS. Bypass leakage would not be processed prior to release. Since the bypass leakage value presented is a bounding value provided for information only, it is not appropriate to include this hypothetical leakage in the assumption identified in Table 15.6-12. If it were included, the radiological consequences would, by definition, just match the 10 CFR 100 criteria. The text on page 15.6-34 wil be modified to clarify the statements.*

No bypass leakage paths which would be a major contributor to off-site doses have been identified. The bypass leakage identified in 6.2.3.3 (0.4 SCFH) is less than 1/50th of the hypothetical bounding leakage valve identified above. Since this contribution will not change the conclusion that the dose consequences would be a small fraction of 10CFR100 criterion, the leakage is not included in Table 15.6-12.

*See attached draft.

b. Leakage from engineered safety feature (ESF) components outside the primary containment - all ESF equipment which circulates primary coolant or suppression pool water during the course of the postulated accident is located within the secondary containment so that any leakage from the pressure barriers for these systems is into the secondary containment atmosphere and is therefore processed by the SGTS prior to release to the environment. Due to the higher SSW pressure any leakage through the RHR heat exchangers would be from the service water side to the ECCS side.

c. Hydrogen purge - Since the hydrogen recombining system consists of two 100% redundant recombiners, no hydrogen purge is required nor assumed throughout the post accident period.

d. Leakage from the main steam isolation valve leakage control system (MSIV-LCS). The MSIV-LCS routes any leakage through the MSIVs to an area serviced by the SGTS. Assuming the MSIVs leak at 11.5 SCFH per valve, leakage past the inboard MSIVs is conservatively estimated to begin 4 hours after the accident. The airborne fission products are assumed to be uniformally mixed in the drywell air volume neglecting the suppression pool air volume. For conservation this leakage is assumed to be in addition to the .5% containment leakage discussed in paragraph a above.

Fission product release to the environment based on the above assumptions is given in Table 15.6-14 and 15.

15.6.5.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-16 and are well within the guidelines of 10 CFR 100.

In addition to the primary to secondary containment leakage identified in 15.6.5.5.1.2, an additional .11%/day of bypass leakage could hypothetically occur before the resulting ~~can occur which bypasses the SGTS with the~~ radiological consequences would exceed ~~still within~~ 10 CFR 100 criteria using the design basis assumptions. This value is far in excess of the bypass leakage identified in Section 6.2.3.3.

The calculated exposures to control room personnel is presently being evaluated and will be incorporated into the FSAR by means of an amendment.

2 of 2

Q 312.18
(15.6.5)

Your latest response to Item 022.7 in your letter of November 21, 1978, indicates that you cannot provide the secondary containment pressure response following postulated loss-of-coolant accident until February 1979. Since we need this information before we can begin our evaluation of the potential off-site doses, we request that you expedite your response on this matter. Our original request for this information was forwarded to you in our first acceptance review dated June 24, 1977.

Response:

An analysis of the Secondary Containment Pressure Temperature Response per questions 22.7 and 312.12 has been completed. The conclusions of the analysis were that the system was capable of maintaining building pressure at -0.25" w.g. More evaluation is required, however, to resolve specific concerns on fuel pool cooling. When we have completed our evaluation and decide upon a course of action, we will inform you. An analysis will then also be submitted. At this time we expect the decision to be made by May, 1979.*

*modified draft response to Question 22.7 attached.

Q 22.7

Provide the secondary containment pressure time response for the design basis accident. List and discuss all assumptions made in this analysis.

Response:

~~The analysos is continuing along the outline stated in Amendment No. 1. However, the effort is more time consuming than originally contemplated. It is now estimated a reponse will be forthcoming in February 1979.~~

See the response to question 312.18

Q. 312.19
(15.6.5)

Indicate the length of piping between the inboard and outboard main steam line isolation valves. Indicate the measures provided to prevent the inboard leakage control system on the main steam line from operating in the event that an inboard main steam line isolation valve failed to close.

RESPONSE:

The length of piping between the inboard and outboard main steam line isolation valves is approximately 21' 6½", center line to center line.

The inboard MSLC system is interlocked with the inboard main steam isolation valve position switch. The inboard MSLC system will remain isolated if the inboard main steam isolation valve is not fully closed.

Responses to:

Effluent Treatment Systems Branch Questions
(321.3 - 321.5)

Q 321.3
(6.5.2)

Revise Table 6.5-2 of the FSAR and the discussion beginning on Page 6.5-6 to indicate whether the WNP-2 facility conforms with the guidance contained in Regulatory Guide 1.52, Revision 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978. Alternatively, provide justification for those design features which do not conform with the cited regulatory guide.

Response:

The pages in question will be revised to show degree of conformance with respect to Reg. Guide 1.52, Rev. 2.* Note that the pages were revised in December to show conformance with Rev. 1.

*See draft revised pages attached.

The chlorine detectors and remote air intake radiation monitors are located in the control room HVAC equipment room. All other controls associated with the control room habitability systems are located in the main control room. All components of the system, including all fresh air intake valves, can be controlled manually from the main control room. Suitable temperatures for equipment operation are maintained in the control room HVAC equipment room by the emergency switchgear area ESF ventilation system (see 9.4.1). All controls associated with the emergency switchgear area ventilation system are also located in the main control room.

The emergency filter units through which the control room is pressurized in the event of a LOCA, are each composed of a roughing filter, HEPA filter, charcoal filter and fan in series. An electric duct heater is provided upstream of the unit to limit the relative humidity of the air entering the unit to 70 percent. A detailed description of the unit is given in 9.4.1. A comparative evaluation of the units which respect to Regulatory Guide 1.52, ~~(June 1973)~~ Rev.2, is given in 6.5.1.

Ceiling mounted ionization and thermal fire detectors are provided for the following areas in the noted quantities:

Area	Number of Detectors: Ionization	Number of Detectors: Thermal
Control Room	10	-
Kitchenette and Dining Area	-	2
Office Area	1	-
Computer peripherals	1	-

Ceiling mounted detectors are uniformly distributed spatially to provide full detection coverage of each area. The fire protection system is described in 9.5.1. The Halon fire extinguishing system is provided within each panel assembly and is discussed in 9.5.1.

In the event of primary fan failure a flow switch on the primary fan discharge automatically closes the isolation valve on the primary fan intake, deenergizes the primary heater bank, energizes the secondary heater, opens the secondary fan isolation valves, and starts the standby fan.

The plant operator may stop one of the SGTS units from the control room after start up is complete. In the event that the radiation monitors in the discharge duct indicate an unaccepatable radiation level in the system discharge air the operator starts the second unit and diverts the discharge air of the operating unit back into the reactor building to minimize off-site release of halogens, and/or for cooling of the charcoal bed.

The following is a comparison of the ESF filteration systems with each position detailed in Regulatory Guide 1.52, ~~issued June 1973~~. Rev. 2.

Article A - "Introduction"

The engineered safety feature filtration systems provided for WNP-2 are designed to the General Design Criterion referenced in Article A. Those systems which are designed to meet the criterion are as follows:

a. Standby Gas Treatment System (6.5.1)

b. Control Room Emergency Filter System (9.4.1)

Article B - "Discussion"

The above two systems are both classed as secondary systems and are not subjected to the drywell environment during any design basis accident and are not subjected to containment cooling sprays. Equipment design includes the ability to operate under all environmental conditions to which they can be subjected during accident conditions. The components of each control room filter unit are as described in this article except that no demisters are required and HEPA filters are not provided downstream of the charcoal adsorber section. The effects of aging, weathering and relative humidity have been considered in the design of these atmosphere cleanup systems, and are tested periodically to verify the required performance capability.

The effects of moisture on the charcoal adsorber media is minimized by the use of strip heaters for humidity control in the plenum of the charcoal adsorbers section of the SGTS units and by periodically circulating heated air through the control room filter units. Adequate space and accessibility for personnel has been incorporated in the filter unit designs to ensure maintainability and testability. Testing of all filters is performed to meet the objectives of Regulatory Guide 1.52, ~~issued June 1973.~~ Rev. 2.

Article C-"Regulatory Position"

Table.6.5-2 provides an analysis of the engineered safety feature air filtration systems with respect to the regulatory positions of Regulatory Guide 1.52, ~~issued June 1973.~~ Rev. 2.

6.5.1.3 Design Evaluation

A design evaluation of the control room emergency filter units is given in 9.4.1.

The standby gas treatment system is designed to prevent the exfiltration of contaminated air from the secondary containment following an accident or abnormal occurrence which could result in high airborne radiation in the secondary containment. All necessary equipment and surrounding structures are Seismic Category I. The engineered safety features buses supply power to the SGTS in the event of loss of normal a-c power. Two fully redundant equipment trains separated by a missile wall are provided to ensure that a single failure does not impair or preclude system operation. A standby gas treatment system failure analysis is presented in Table 6.5-3.

6.5.1.4 Tests and Inspections

The test and inspection program applicable to the control room emergency filter units is discussed in 9.4.1.

The standby gas treatment system and its components are thoroughly tested in a program consisting of the following classifications:

a. Predelivery tests and component qualification tests

b. Post-delivery acceptance tests

c. Post-operation surveillance tests.

Written test procedures establish acceptance criteria for all tests. Test results are recorded in performance records, thus enabling early determination of end life performance.

All predelivery, post-delivery and post-operation tests are performed to meet the objectives of Regulatory Guide 1.52, ~~issued June 1973.~~ Rev. 2.

HEPA filters are factory tested to a minimum efficiency of 99.97% when measured with a 0.3 micron DOP aerosol. ~~After delivery and installation, each HEPA filter bank is tested initially and semiannually thereafter in accordance with ANSI N510-1975 "Testing of Nuclear Air Cleaning System". The acceptance level for these tests is a penetration less than 0.05% at rated flow.~~

Charcoal media qualification tests meet the objectives of Regulatory Guide 1.52, ~~issued June 1973.~~ Rev. 2.

Once charcoal media is installed in the filter frames, at the site, each charcoal bed is leak tested with refrigerant 112. R-112 is injected upstream of the charcoal beds at a concentration of 20 ppm at rated flow. A gas chromatograph is used to sample the R-112 concentration downstream of the filter assembly. The maximum allowable concentration of R-112 downstream of the filter is 0.01 ppm. Concentrations greater than 0.01 ppm constitute failure of the filter and the filter will be repaired and retested. This test is performed semiannually at the same time the HEPA filters are tested.

Twelve 4 inch deep test canisters are installed in parallel with each of the charcoal adsorber sections. Once a year one test canister from each adsorber section is removed, with opening blanked off, and sent to a laboratory for testing. Each sample is tested with methyl iodide per RDT M16-1T as defined in Regulatory Guide 1.52, ~~issued June 1973.~~ Rev. 2.
In the event a sample fails to meet this test, the charcoal media in that adsorber section will be replaced.

All SGTS fans are factory tested in accordance with AMCA Standard 210 "Air Moving and Conditioning Association Test Code for Air Moving Devices". Once installed, fans are started once per month to ensure operability.

All valves associated with the SGTS are factory leak tested, bubble tight, at a pressure differential of 2 psig. All valves are factory tested to ensure that valve stroke time, full close to full open, does not exceed 4 seconds. Once installed, the valves of the SGTS are stroked at least once every six months to ensure operability.

6.5.1.5 Instrumentation Requirements

The instrumentation and control system of the control room emergency filter units are discussed in 7.3.1.1.6 and 9.4.1. The instrumentation and control system of the SGTS is discussed in 7.3.1.1.9.

All instrumentation and controls are designed to meet the objectives of Regulatory Guide 1.52, ~~issued June 1973.~~ Rev. 2.

The following instrumentation is provided for each SGTS train in addition to that described in 6.5.1.2.

a. An indicating differential pressure gauge is provided across each element (excluding heaters) in the SGTS train. High differential pressure annunciates an alarm in the main control room and is permanently recorded by the computer.

b. Relative humidity detectors, with humidity indication in the main control room, are located before the electric blast coil heaters and before the charcoal adsorber banks. High humidity annunciates an alarm in the main control room and is permanently recorded by the computer.

c. Thermostats with sensors on either side of an adsorber section control strip heaters in both adsorber plenum sections. Two thermostats in parallel energize the heaters to maintain a minimum temperature of 90°F. Another thermostat deenergizes the heaters on a temperature rise to 110°F with a manual reset thermostat cutting out the heaters on a temperature rise to 125°F.

d. Temperature indication is provided in the main control room for air entering the electric blast coil heater section and the air leaving both banks of charcoal filters.

e. The deluge spray systems of the prefilter and both charcoal adsorber sections are each controlled separately by temperature switches.

TABLE 6.5-2

ENGINEERED SAFETY FEATURE AIR FILTRATION SYSTEMS - COMPARISON WITH ARTICLE C, REGULATORY POSITION, REGULATORY GUIDE 1.52, REV. ~~1~~ 2, ISSUED ~~JULY 1976~~ MARCH 1978

Paragraph No.	SGTS	Control Room System
C-1. "Environmental Design Criteria"		
1.a	In compliance	In compliance
1.b	In compliance	In compliance
1.c	In compliance	In compliance
1.d	In compliance	In compliance
1.e	In compliance	In compliance
C-2. "System Design Criteria"		
2.a	In compliance	See Note 1
2.b	In compliance	In compliance
2.c	In compliance	In compliance
2.d	See Note 2	See Note 2
2.e	In compliance	In compliance
2.f	In compliance	In compliance
2.g	See Note 3	See Note 3
2.h	In compliance	In compliance
2.i --- In compliance	See Note 4	See Note 4 In compliance
2.j	In compliance	In compliance
2.k	In compliance	In compliance
2.l		
C-3 "Component Design Criteria and Qualification Testing"		
3.a	See Note 5	See Note 5
3.b	In compliance	In compliance
3.c	In compliance	In compliance
3.d	See Note 6	See Note 6
3.e	In compliance	In compliance
3.f	In compliance	In compliance
3.g	See Note 7	See Note 7
3.h	In compliance	In compliance
3.i	See Note 8	See Note 8
3.j	In compliance	In compliance
3.k	In compliance	In compliance
3.l	In compliance	In compliance
3.m	In compliance	In compliance
3.n	In compliance	In compliance
3.o	In compliance	In compliance
3.p	In compliance	In compliance

Paragraph No.	SGTS	Control Room System
C-4. "Maintenance"		
4.~~a~~	~~In compliance~~	~~In compliance~~
4.~~b~~a	See Note 9	See Note 9
4.~~c~~b	See Note 10	See Note 10
4.~~d~~c	In compliance	In compliance
4.~~e~~d	See Note 11	~~See Note 11~~ In compliance
4.~~f~~e	In compliance	In compliance
C-5. "In-Place Testing Criteria"		
5.a	In compliance	In compliance
5.b	In compliance	In compliance
5.c	In compliance	In compliance
5.d	In compliance	In compliance
C-6. "Laboratory Testing Criteria For Activated Carbon"		
6.a	See Note 12	In compliance
6.b	See Note 12	In compliance

TABLE 6.5-2 (Continued) Page 5 of 5

Note 11
C-4.£e

Strip heaters are provided in the charcoal filter plenum of the SGTS units to maintain charcoal beds moisture free; therefore operation of the fans is not required.

Note 12
C-6.a
C-6.b

The laboratory testing criteria for the carbon adsorber section of the SGTS meets the objectives of this section of the guide. Twelve representative test samples of 4 inch length are provided across each of the two 4 inch deep beds in each SGTS filter unit. Once per year, one sample from across each adsorber bed is removed and sent to a laboratory for testing. Each sample is tested separately. Batch tests are performed with methyl iodide at 70% relative humidity with a penetration of less than 1.0 percent as an acceptance level. In the event that a sample fails this test, the carbon adsorber in its bed will be replaced.

TABLE 6.5-2 (Continued) Page 4 of 5

"visible" carry-over. Farr contends that by adding 3 two inch fiberglass pads their demister (which was acceptable in the Fort St. Vrain clean up filter housing) should also be approved. There are no demisters on the control room filter units. See Note 1 (C-2.a) above.

Note 6
C-3.d HEPA filters are not subjected to iodine removal sprays; therefore, aluminum separators are used.

Note 7
C-3.g Access doors into SGTS units are 50 x 20 inches. Vacuum breakers are not provided on doors of SGTS and control room units. Unit fans are normally off. During tests, bypass is via temporary blanking off of doors.

Note 8
C-3.i <u>Test 4, Activity</u>

Base carbon (unimpregnated) activity test was not previously required and because all available carbon was of the impregnated type this test was not run.

<u>Test 5a, Radioiodine Removal Efficiency</u>

The activated carbon (Barnebey Cheney 727) radioiodine removal efficiency for methyl iodide at 25°C and 95% relative humidity is 98% instead of 99%.

The methyl iodide test at 25°C and 95% R.H. was not required by Regulatory Guide 1.52, Rev. 0, or contract specification. Considering also that the adsorber bed will not see any relative humidity above 70% the test results of 98% removal efficiency at 95% R.H. should be satisfactory.

Note 9
C-4.~~b~~a Doors provided on SGTS units are 50 x 20 inches. Access panels are provided on control room units. Vacuum breakers are not provided on any of the units since they are normally not operational.

Note 10
C-4.~~c~~b Control room filter units have approximately 18 inches between prefilter and HEPA filter frames, and approximately four feet are provided between HEPA and charcoal filter frames. SGTS filter units have a minimum of three feet provided between demister, heater, prefilter, HEPA, and charcoal filter frames.

2 MARCH 1978
Regulatory Guide 1.52, Rev. ~~1~~, ~~July 1976~~

Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance given in Revision ~~1~~ 2 of this regulatory guide.

General Compliance or Alternate Approach Assessment:

Standby gas treatment filter units and the control room emergency filter units are required to perform safety related functions. An analysis of the engineered safety feature air filtration systems with respect to the regulatory position of Regulatory Guide 1.52, ~~Rev. 1~~ Rev. 2 is given in Table 6.5-2.

Specific Evaluation Reference:

Refer to 6.5.1.

Q 321.4

Provide an evaluation showing conformance of your seismic design criteria for the tank support elements of the gaseous radwaste system charcoal delay tanks with the guidance contained in Section C.5 of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," July 1978. Alternatively, provide justification for not conforming with the staff guidance on this matter.

Response:

The tank support elements for the charcoal delay tanks in the off-gas system were designed to Seismic Category I Requirements. The design is therefore more than adequate with respect to the guidance in Regulatory Guide 1.143. Refer to paragraph 11.3.1.3.*

*See attached draft revision.

General Design Criteria 60

The system has sufficient capacity to reduce the off-gas activity to permissible levels for release during normal operation, including anticipated operational occurrences, and to alleviate any termination of releases or limitation of plant operation due to unfavorable site environmental conditions.

General Design Criteria 64

Continuous monitoring of activity levels in the system upstream of the delay line provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation at activity levels corresponding to administrative release limits and annunciation at lower levels, along with continuous monitoring of the reactor building elevated release duct, radwaste building ventilation exhausts and turbine-generator building ventilation exhausts, provide assurance that activity releases to the environment will in all events be maintained within established limits.

11.3.1.3 Equipment Design Criteria

A list of the off-gas system major equipment items which includes materials, process conditions, and number of units supplied, is provided in Table 11.3-2. Equipment and piping is designed and constructed in accordance with the requirements of the applicable codes as given in Tables 3.2-1 and 3.2-2.

The quality group classifications of the various systems are shown in Table 3.2-1. Seismic category, safety class, quality assurance requirements, and principal construction codes information is contained in 3.2. The system is designed to Quality Group Classification C.

The reactor building, turbine-generator building, and radwaste building contain radioactive gas sources. The design bases and characteristics for the ventilation systems for these three buildings are described in 9.4.

Equipment and components used to collect, process, or store gaseous radioactive waste are not designed as Seismic Category I, Conservative analyses similar to those presented in Reference 11.3-5 demonstrate that equipment failure will not result in off-site doses exceeding 0.5 Rem. The failure of the off-gas system, the related failure of the steam-jet air ejector lines and the gland sealing system are analyzed in 15.7.1.

with exception of the charcoal adsorber vessel supports.

Q 321.5

Provide a table showing your provisions for preventing uncontrolled releases of radioactive materials due to spillage in the safety-related buildings of the WNP-2 facility or from outdoor tanks, including the condensate storage tanks. Demonstrate conformance with the guidance contained in Section C.1.2 of Regulatory Guide 1.143, July 1978. Alternatively, provide justification for not conforming with the staff guidance on this matter.

Response:

The liquid waste management system meets the objectives of Regulatory Guide 1.143. Tanks which hold radioactive liquid, including the condensate storage tanks, are monitored for level and alarm primarily in the radwaste control room. The radwaste systems are operated from the radwaste control room; hence, additional local alarms are not required for radwaste tank levels. Condensate storage tank level alarms are located in the main control room.

All liquid waste management system tank overflows and drains are routed to the radwaste building equipment and floor drain sumps, refer to FSAR Figure 9.3-7. The condensate storage tanks overflow to the area floor drains within the dike. If radioactivity is detected in these drains, the liquid is routed to a turbine building radioactive floor drain sump. Refer to FSAR Figure 9.2-9. Radioactive liquid samples are primarily routed to sampling sinks. Those samples, which are local, drain into floor drain trenches, equipment or floor drain funnels, and pump beds. The condensate storage tank local samples drain into the area floor drains within the dike. The above sample receivers are routed to various radioactive sumps, all of which are processed by the liquid waste management system.

Indoor tanks which hold radioactive liquid are not enclosed by individual curbs or elevated thresholds. The portion of the radwaste building which houses radioactive liquid tanks is Seismic Category I. The radwaste building can retain the normal operating capacity of all liquid radwaste tanks in the event an accident occurs, refer to FSAR paragraph 15.7.2.* Therefore, the intent of Regulatory Guide 1.143 is met on this item.

The radioactive and non-radioactive equipment and floor drains within the plant are segregated. Equipment and floor drains within the reactor building and radwaste building are routed to the liquied waste management system. The turbine building equipment and floor drains are segregated for radioactivity by component source and area. In addition the non-radioactive floor drain sumps in the turbine building contain radiation monitors which route the sump discharges to the liquid waste management system upon detecting radioactivity.

*See attached draft FSAR changes.

The condensate storage tanks are the only outdoor tanks which may contain radioactivity. These tanks are enclosed by a dike which can retain the contents of the tanks.

The following table demonstrates individual tank design features used to prevent uncontrolled releases of radioactive material due to spillage from indoor outdoor tanks.

TANK DESIGN FEATURES

Component	High Level Alarm	Overflows & Drains To	Enclosed By
Floor Drain Collector Tank	yes (RCR)[a]	Floor Drain Sump W-2	Radwaste Bldg.
Waste Sludge Phase Separator	yes (RCR)	Floor Drain Sump W-2	Radwaste Bldg.
Floor Drain Sample Tank	yes (RCR)	Floor Drain Sump W-1	Radwaste Bldg.
Waste Collector Tank	yes (RCR)	Floor Drain Sump W-2	Radwaste Bldg.
Spent Resin Tank	yes (RCR)	Floor Drain Sump W-2	Radwaste Bldg.
Waste Surge Tank	yes (RCR)	Equipment Drain Sump, W-3	Radwaste Bldg.
Waste Sample Tanks	yes (RCR)	Equipment Drain Sump, W-3	Radwaste Bldg.
Detergent Drain Tanks	yes (RCR)	Chemical Waste Sump, W-4	Radwaste Bldg.
Chemical Waste Tanks	yes (RCR)	Chemical Waste Sump, W-4	Radwaste Bldg.
Distillate Tanks	yes (RCR)	Chemical Waste Sump, W-4	Radwaste Bldg.
Decon. Sol. Conc. Waste Tanks	yes (RCR)	Chemical Waste Sump, W-4	Radwaste Bldg.
Condensate Back-wash Receiving Tank	yes (RCR)	Equipment Drain Sump, W-3	Radwaste Bldg.
Condensate Phase Separators	yes (RCR)	Equipment Drain Sump, W-3	Radwaste Bldg.
Decon. Sol. Conc. Waste Measuring Tank	yes (local)	Decon. Sol. Conc. Waste Tanks	Radwaste Bldg.
RWCU Phase Separators	yes (RCR)	Equipment Drain Sump, W-3	Radwaste Bldg.
Condensate Storage Tanks	yes (MCR)[b]	Floor Drain Sump T-4	Dike

(a) Radwaste Control Room (RCR)

(b) Main Control Room (MCR)

(c) Hopper mixers are interlocked to stop the centrifuges and shift valve lineups on local high level alarms, refer to FSAR Figure 11.4-1.

(d) Overflow of the centrifuges is routed to the waste sludge phase separator.

11.2 LIQUID WASTE MANAGEMENT SYSTEM

11.2.1 DESIGN BASIS

11.2.1.1 Design Objective

The liquid waste management system is designed to collect, segregate, store and process potentially radioactive liquids generated during normal plant operation and anticipated operational occurrences. The design objective is to keep the radiation dose in unrestricted areas as low as reasonably achievable within the guidelines of Appendix I to 10CFR50, May 5, 1975. The design incorporates the objectives of maximum recycle and minimum release of radioactive liquids without limiting plant operations or availability.

11.2.1.2 Design Criteria

The criteria considered in the design of this system include volume, radioactivity, operational exposure and required quality for recycle of the processed liquid.

The system is designed to treat process liquids with radioisotope concentrations associated with the design basis fuel leakage and produce a quality of water which allows its recycle for plant reuse. Water inventory will at times require the discharge of processed liquids to the environs in which case concentrations of radioisotopes in the effluent (Table 11.2-9) will be significantly less than the values specified in 10CFR20 and within the release limits established in the technical specifications. Radiation exposure to persons in unrestricted areas, resulting from liquid waste discharged during normal operation and anticipated operational occurrences is less than the guidelines specified in 10CFR50, Appendix I. The radwaste building can retain the normal operating capacity of all liquid radwaste tanks in the event an accident occurs as postulated in 15.7.~~3~~2.

The design of the system was accomplished prior to the issuance of Regulatory Guide 8.8. However, the system does incorporate substantially the guidance provided in this regulatory guide. Components radioisotopic inventories are listed in Table 11.2-8. These values are based on the reactor water source term associated with the design basis fuel leakage rate. Allowance is made for concentration, decay and daughter product build-up in filters, demineralizers and tanks. Equipment locations and arrangements are shown in 1.2.

6 of 6

Responses to:

Radiological Assessment Branch Questions

(331.15 - 331.24)

QUESTION: 331.15 Indicate whether the guidance provided by the following regulatory guides has been followed: Regulatory Guide 8.14, "Personnel Neutron Dosimeters" and Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection." If so, describe how. If not, describe the specific alternative methods.

RESPONSE:

Regulatory Guide 8.14

The guidance given in Regulatory Guide 8.14 "Personnel Neutron Dosimeters" has been followed except that an alternate accuracy requirement has been substituted for the Position 2 accuracy requirement. The alternate requirement uses AmBe instead of Cf-252 because of its longer half life and lower gamma dose rate. The alternate requirement is:

> "When exposed to either unmoderated or moderated (4 Poly) neutrons from an AmBe source, the average accuracy of a set of 10 dosimeters exposed in the range of 100 mrem to 3 rems should be $\pm$ 50%."

All remaining guidance and the alternate accuracy requirement have been incorporated in TLD purchase specifications and acceptance criteria, and will be followed in TLD and neutron survey instrument calibration procedures, neutron survey procedures, and dosimetry procedures.

Regulatory Guide 8.15

Respiratory protective equipment used at WNP-2 for radiological purposes is used as stipulated in Regulatory Guide 8.15 with one exception.

The only exception to Regulatory Guide 8.15 is the use of charcoal cartridges for protection from radioiodine. The charcoal cartridges are used only in the interest of maintaining exposures ALARA. No allowance is taken, for the protection provided, when assessing internal exposures.

Management policy and assignment of responsibilities regarding respiratory protection are contained in Supply System Health Physics Policy Manual.

Equipment selection is based on evaluation of the hazard and provides MESA/NIOSH approval where available.

Each prospective respiratory protective device user is provided a medical evaluation, formal training, and a quantitative man-test.

Procedures are provided for equipment selection, control, maintenance and testing.

Routine in-vivo bioassay is employed to evaluate individual exposures and to assess protection provided.

Q. 331.16
(12.1.2)

The following design features are intended to complement the design objective characterized by the phrase "as low as reasonably achievable" (ALARA). Describe how your plant design reflects a consideration of these two features:

a. The use of adequate and quick-service auxiliary lighting in high radiation areas.

b. A clear identification of localized radiation sources.

RESPONSE:

a. The lighting system at WNP-2 is described in detail in Subsection 9.5.3. Lighting consists mainly of fluorescent or mercury vapor lamps. Fluorescent lamps have a mean lifetime in excess of 20,000 hours for a 40 watt bulb at three starts/hour. Mercury vapor lamps have a mean lifetime of 24,000 hours under the same starting conditions. The lights in Zone V areas are generally turned off except for inspection, maintenance or other personnel access operations, therefore, the replacement of these lamps will be rare.

   The emergency lighting system provides lighting in all normally lighted areas to allow a safe and orderly shutdown and personnel egress. In the event that the normal emergency lighting fails, the battery powered emergency lighting system will provide lighting to exits and passageways.

b. The design of WNP-2 identifies, and minimizes exposure from, major localized radiation sources. Examples include the following:

   o Demineralizers and filter demineralizers are contained in individual shielded cells with no auxiliary components (valves, etc.) within the cell.

   o Reactor water cleanup hold pumps are shielded separately from other hold pumps.

   o Radwaste sludge pumps are provided with automatic flushing capability.

Q 331.17
(12.2.2)

Tables 12.2-14 through 12.2-16 of the FSAR indicate the estimated airborne radionuclide concentrations for three areas of the WNP-2 facility which will be normally occupied during operation. Provide similar tables giving the estimated airborne radionuclide concentrations for the solid radwaste handling areas and the liquid radwaste handling areas.

Response:

See revised section 12.2.2 of the FSAR.*

*Draft changes attached

12.2.2.3.7 Effects of Solid Radwaste Handling Areas

The solid radwaste handling equipment contained in rooms number C-127 and C-128 Figure 12.3-7 are designed for remote operation. Entry for maintenance activities will normally entail shut down and flushing of systems and equipment. The solid radwaste handling equipment is operated from room C-126.

The ventilation supply to room C-126 is clean outside air with air flow into surrounding normally unoccupied areas. The only source of airborne radioactivity, the waste compactor, is vented directly to the filtered exhaust ventilation system.

Airborne radioactivity levels in the normally occupied solid radwaste handling areas will be ambient outside air concentrations.

12.2.2.3.8 Effects of Liquid Radwaste Handling Areas

Normally occupied liquid radwaste handling areas include the valve corridor C-218 Figure 12.3-8, Precoat Room C-222 Figure 12.3-8 and the radwaste control room C-220 Figure 12.3-8.

The valve corridor C-218 is supplied directly with outside air. Components which are operated from corridor C-218 and contain radioactive materials are located in normally unoccupied valve and pump rooms which are served by separate ventilated supply and exhaust.

The radwaste control room and the precoat rooms do not house radioactive material containing components.

Although not normally occupied, the possibility exists that entry into pump corridor C-125, Figure 12.3-7 and valve and pump rooms C-217 and C-219 could be necessary while systems are operating.

The pump corridor, C-125, contains the highest concentrations of radioactive material and has the lowest air flow to volume ratio, so is taken as a worst case. Equilibrium airborne radioactivity concentration is calculated as described in 12.2.2.2 assuming a leak rate of 50 gallons per day from the reactor water clean-up phase separator decant pump. The results are shown in Table 12.2-17.

TABLE 12.2-17

AIRBORNE RADIONUCLIDE CONCENTRATION IN LIQUID RADWASTE HANDLING AREA

Radionuclide	Airborne Concentration C_i (Ci/cc)	Maximum Permissible Concentration (Ci/cc)	Ratio of C_i to MPC
Ba-140	5.5×10^{-10}	4×10^{-8}	1.4×10^{-2}
La-140	6.2×10^{-10}	1×10^{-7}	6.2×10^{-3}
Np-239	2.1×10^{-9}	7×10^{-7}	3.0×10^{-3}
Co-58	9.3×10^{-10}	5×10^{-8}	1.9×10^{-2}
Sr-89	4.6×10^{-10}	3×10^{-8}	1.5×10^{-2}
Mo-99	2.5×10^{-10}	2×10^{-7}	1.3×10^{-2}
Te-99M	1.6×10^{-10}	1×10^{-5}	2.0×10^{-4}
Te-132	1.4×10^{-10}	1×10^{-7}	1.4×10^{-3}
I-131	8.8×10^{-10}	9×10^{-9}	9.7×10^{-2}
I-132	2.3×10^{-10}	2×10^{-7}	1.2×10^{-3}
I-133	3.9×10^{-10}	3×10^{-8}	1.3×10^{-2}
I-135	1.7×10^{-10}	1×10^{-7}	1.7×10^{-3}

Q 331.18

Indicate the frequency at which you plan to calibrate your area radiation monitors.

Response:

See revised section 12.3.4.3 of the FSAR.*

*Draft change attached

12.3.4.3 Specification for Area Radiation Monitors

The areas readiation-monitoring system is shown as a funtional block diagram in Figure 12.3-20. Each channel consists of a combined sensor and a converter unit, a combined indicator and trip unit, a shared power supply, and a shared multipoint recorder. All channels also have a local audible alarm auxiliary unit mounted near the sensor.

Each monitor has an upscale trip that indicates high radiation and a downscale trip that indicates instrument trouble. These trips sound alarms but cause no control action. The trip circuits are set so that a loss of power initiates an alarm.

The type of detector used is a Geiger-Mueller tube responsive to gross gamma radiation over an energy range of 80 KeV to 3 MeV. Each detector range covers four decades. Detector ranges are given in Table 12.3-1.

The overall accuracy within the manufacturer's design range of temperature, humidity, line voltage and line frequency variation is such that the actual reading relative to the true reading, including susceptability and energy dependance (100 KeV to 3 MeV) is within 9.5% of equivalent linear full scale recorder output for any decade.

The calibrating frequency is once every 18 months and assures ~~such as to assure~~ that drift does not exceed ± 0.2% of equivalent linear full scale recorder output for a 24-hour period or a ± 2% for a 30-day period.

Facilities for calibrating area radiation monitor units are provided for by means of a test fixture designed for use in the adjustment procedure for the area radiation monitor sensor and converter unit. It provides several gamma radiation levels between 10 and 250 mR/hr. The calibration unit source is cobalt-60. A cavity in the test fixture receives the monitor sensor. A window is located on the back wall of the cylindrical lower half of the cavity through which radiation emanates from the source to the sensor. A chart on each test fixture indicates the radiation levels available from the unit for the various control settings. For checking at higher radiation levels, a Cesium 137 source up to 10 Ci in strength is provided in a shielded test fixture.

Q. 331.19
RSP
(12.3.4)

We require that area radiation monitors located in high noise areas should have visual as well as audible alarms. State how you plan to comply with our position on this matter.

RESPONSE:

Radiation monitors located in high noise areas will be equipped with an audible alarm and a visual alarm (either a rotating beacon or strobe light).

Q 331.20
(12.4.1)

In addition to the job group exposure breakdown in Section 12.4 of the FSAR, provide a profile of the estimated annual man-rem doses at the WNP-2 facility broken down by major functions such as operations, maintenance, radwaste handling, and inservice inspection. Using experience from operating boiling water reactors, provide estimates of doses resulting from non-routine or special maintenance activities. Indicate the estimated dose rates, the expected required number of workers and the occupancy times required for performing such maintenance work which you used in evaluating the estimated annual can-rem doses. Regulatory Guide 8.19 provides guidance in making such an assessment.

Response:

See revised Section 12.4.* The tables in revised Section 12.4 provide a profile of the estimated annual man-rem doses at the WNP-2 facility. The tables are based on operating BWR experience and are formulated in accordance with Regulatory Guide 8.19.

*Draft changes attached.

12.4 DOSE ASSESSMENT

Replace with new Section 12.4.1 attached

12.4.1 DESIGN CRITERIA, OCCUPANCY FACTORS AND PERSONNEL DOSE

The criteria for the dose to plant personnel during normal operation and anticipated operational occurrences, including refueling, are based on the requirements discussed in 10CFR Part 20. The design radiation levels during normal operation and refueling are shown on Figures 12.3-1 to 12.3-6. In areas such as the control room and offices, the maximum dose rate does not exceed 1.0 mrem/hr (Zone I radiation level). For personnel who work in controlled radiation areas, radiation Zone II through V on Figures 12.3-1 to 12.3-6, administrative controls ensure that doses do not exceed the requirements of 10CFR Part 20.

The occupancy factors used in estimating plant personnel radiation exposure are listed on Table 12.4-1 for the six personnel groups.

a. Group 1 - includes maintenance personnel such as mechanical, electrical instrument craftsmen and foremen. There are approximately 46 people in this group.

b. Group 2 - includes control and equipment operators. There are approximately 29 people in this group.

c. Group 3 - includes technicians such as health physics and chemistry technicians and engineering assistants. There are approximately 8 people in this group.

d. Group 4 - includes engineers and technical supervisors. There are approximately 9 people in this group.

e. Group 5 - includes inplant supervisors such as health physics-chemistry supervisor, shift supervisor, etc. There are approximately 14 people in this group.

f. Group 6 - includes administrative and management personnel. There are approximately 11 people in this group.

These occupancy factors are determined by estimating the amounts of time spent by personnel in controlled radiation areas while performing the following functions:

g. Routine patrol

h. Periodic tests and operations

i. Control room operations

j. Refueling

k. Maintenance

l. In-service inspection

Table 12.4-1 also lists the estimated annual dose that will apply to a particular group in a particular area. To find the dose that will apply to a particular group, the dose rate is multiplied by the occupancy factors and the number of people in each group. The results, listed in Table 12.4-1, represent whole body exposure.

Data on personnel exposure from operating BWR plants show that the operation and maintenance requirements of all BWR plants are similar. It is anticipated that the operation and maintenance requirements of WNP-2 will be similar to other BWR plants and therefore the personnel exposure data will be similar. 12.4.2 discusses personnel exposure based on BWR operating experience.

replace with new 12.4.1 attached

12.4 DOSE ASSESSMENT

12.4.1 DESIGN CRITERIA

The criteria for the dose to plant personnel during normal operation and anticipated operational occurrences, including refueling, are based on the requirements discussed in 10CFR part 20. The design radiation levels during normal operation and refueling are shown on Figures 12.3-1 to 12.3-6. In areas such as the control room and offices, the maximum dose rate does not exceed 1.0 mrem/hr (Zone I radiation level). For personnel who work in controlled radiation areas, radiation zone II through V on figures 12.3-1 to 12.3-6, administrative controls ensure that doses do not exceed the requirements of 10CFR part 20.

~~12.4.2 PLANT PERSONNEL RADIATION EXPOSURES~~

~~12.4.2.1 General (Remains the same)~~

12.4.2 PERSONNEL DOSE ASSESSMENT BASED ON BWR OPERATING DATA

12.4.2.1 General

In general, recent data[1] from operating BWR's have shown that the man-rem exposures to plant personnel are primarily due to the corrosion product isotopes. Of the corrosion product isotopes, Co-60 is believed to be the single most important radionuclide.

The variables that have been found to affect plant personnel exposure include the following:

a. BWR plants show an increase in total personnel exposure during the first few years of operation.

b. The need to minimize plant downtime requires that inspection and repair tasks must be started immediately after plant shutdown when the dose rates from short-lived radionuclides can be significant.

c. Plant design and equipment layout has a significant effect on personnel dose. 12.3.1 discussed the design features used to minimize plant personnel exposure.

d. Training and experience of plant workers.

e. The extent of maintenance operations required for a specific year.

f. The extent that a utility uses non-regular or contractor personnel.

12.4.2.2 Personnel Dose From Operating BWR Data

References 1 and 2 provide a tabulation of personnel exposures for operating BWR's. Table 12.4-~~2~~9 tabulates the average personnel exposure for several plants operating for a period of several years ~~[illegible]~~ based on these references. References 4 through 7 provide more recent information. The assessment summarized in section 12.4.2.3 ~~not~~ includes this more recent information.

Replace with new 12.4.2.2 attached

delete

A tabulation of the average fraction (in percent) of the annual plant exposure is listed on Table 12.4-3. This table shows that the jobs listed account for 47% of the total dose received by plant personnel on the average. The remaining exposure can be largely accounted for after considering miscellaneous routine operations and maintenance. Each of the tasks in this category are insignificant as far as radiation dose is concerned. However, the cumulative dose received by plant personnel after performing many of these tasks becomes significant. Data show that the exposure from routine operations is approximately 33% of the total annual exposure. The remaining 20% of the total annual exposure is accounted for after considering miscellaneous work during outages, bias in accounting for exposure, and differences in dosimetry results.

It is concluded that no single source of exposure is dominant at operating BWR's. The largest single sources were the recirculation pumps including clean-up system and work on valves, particularly relief and safety valves. Each of these sources contributed 8% to the total annual exposure. Inservice inspection, liquid waste treatment systems and fuel handling contributed the next highest exposures, 4.9%, 5.6% and 5.5% of the annual exposure, respectively.

None of the above discussion includes the dose received by contractor personnel. Looking at Table 12.4-2, subtracting the column labeled "regular-man-rem" from "total-man-rem" yields the dose received by contractor personnel. In general, a significant fraction (between 25% and 60%) of the total man-rem is received by contractor personnel.

12.4.2.3 Results and Conclusions

As discussed previously, a precise estimate of occupational exposure of specific individuals is not attainable. A gross assessment is provided in the following paragraphs for the six job group classifications defined in 12.4.1.

a. <u>Group 1</u> - Maintenance personnel would receive the largest dose of any of the six groups. Based on the data discussed in the preceding sections, the average annual personnel dose for all plants with a thermal output greater than 50,000 MWD is 140 man-rem per year per plant. The plants considered are listed on Table 12.4-2.

replace with new 12.4.2.3 attached

From Table 12.4-3, it is seen that maintenance operations, which include control rod drive, recirculation pump, valve, turbine, fuel pool and condensate demineralizer maintenance operations, would account for 21% of the average annual personnel dose. If one-half of the dose from routine operations (33%) and miscellaneous work during outages (20%) also is received by plant maintenance personnel, approximately 50% of the total annual average personnel dose, or 70 man-rem, is received by this group. Assuming 46 people are in this group results in a dose of 1.52 rem per person annually. As discussed in 12.3.1, the equipment design and layout and the shielding design are such that the exposure is as low as reasonably achievable.

b. Group 2 - Plant operation personnel can be divided into three groups - supervisors, control room staff and plant equipment operators. These exposures can be estimated using radiation zone limit values. The values given in Table 12.4-1 show that the total personnel dose for operation and shutdown is approximately 22 man-rem or about 0.7 rem/yr per person. As part of this total, the supervisors and control room staff would be expected to receive an exposure of less than 500 mrem/yr if they remain in the control room and administrative office areas. These people will, however, spend time on inspection of plant systems.

c. Group 3 - If the plant health physics/chemistry personnel spent 1% of their time collecting samples in Zone III sampling stations, they will receive a maximum dose of 300 mrem/yr. Assuming that they spend the remainder of their time in Zone I and Zone II areas, their total dose would be between 1 and 2 rem per person. The plant health physics/chemistry personnel also conduct radiation surveys and support maintenance activities which require monitoring and pre-job radiation surveys. The exposure to the health physics/chemistry staff can range from 1 to 3 rem/yr. This is based on experience from operating plants.

replace with new section 12.4.2.3 attached

7 of 32

replace with new section 12.4.2.3 attached

Assuming a dose rate per person of 2 rem/yr and considering the 8 people in this group, the total personnel dose is 16 man-rem.

d. Group 4 - The engineering staff and technical supervisors will spend most of their time in Zone I areas where exposures will be less than 500 mrem/yr. They will spend time in higher radiation areas. However, the resulting dose that they will receive is difficult to estimate accurately. The dose received by the 9 people in this group is taken to be 4.5 man-rem.

e. Group 5 - Plant supervisors will spend time supervising Group 1 and Group 3 personnel. Their dose rate would be about the same as that received by personnel in Group 1 and 3. This value is about 1 rem/yr. With 14 people in this group, the total personnel dose is 14 man-rem.

f. Group 6 - The administrative and plant personnel will spend their time in Zone I radiation areas. Thus, it is expected that their dose rate will be less than 500 mrem/yr. With 11 people in this group, the total personnel dose becomes 5.5 man-rem.

The total personnel dose for all groups is approximately 130 man-rem. Assuming that another 130 man-rem will be received by contractor personnel, as discussed in 12.4.2.2, the total personnel exposure is estimated to be 260 man-rem.

12.4.3 INHALATION EXPOSURES

Airborne radionuclide concentrations in normally occupied areas are, as discussed in 12.2.2, well below the limits set by 10 CFR Part 20 and thus inhalation exposures are negligible.

A summary of the total estimated man-rem dose broken down by major function is given in Table 12.4-1. More detailed breakdowns are presented in Tables 12.4-2 through 12.4-8 for each of the seven major functions given on Table 12.4-1. These tables are based on the more recent information obtained from References 4 through 7. The data from Table 12.4-9 is given for comparison purposes only.

The results of the total estimated man-rem doses will be discussed with reference to six occupational groups as follows:

a. <u>Group 1</u> - This group includes maintenance personnel such as mechanical, electrical, instrument craftsmen and Foremen. There are approximately 46 people in this group. Tables 12.4-4 and 12.4-8 provide the functional breakdown of exposures for this occupational group. As can be seen from the Tables, 373 total man-rem may be expected.

(go to next page)

~~Table 12.4-1 and 12.4-2~~, Routine and special maintenance operations which include control rod drive repairs, Residual Heat Removal (RHR) repairs, Snubber maintenance ~~[illegible]~~, etc. account for approximately ~~approx~~ 62% of the average annual personnel dose.

One to three rem per year per person is projected for the Station maintenance personnel for a maximum total of 138 man-rem per year. Accordingly, the remaining 235 ~~[illegible]~~ man-rem per year ~~will be~~ would be expected to be received by non-station maintenance personnel. As discussed in 12.3.1, the equipment layout and design and shielding design are such that the exposures are as low as reasonably achievable (ALARA).

b. ~~Group I~~ - This group includes plant operations personnel ~~which~~ ~~falls~~ composed of ~~[illegible]~~ ~~three~~ sub ~~groups~~ supervisors, control room staff and plant equipment operators. There are approximately 29 people in this group. Tables 12.4-2, 12.4-3, 12.4-5 and 12.4-6 show the total estimated man-rem ~~dose~~ for ~~this group~~ ~~for operation and shutdown is approx~~. As can be seen, the total is approximately 111 ~~[illegible]~~ man-rem per year or approximately 3.8 rem per year per man. Personnel in this group will be performing routine and non-routine operation and surveillance, waste processing and refueling operations. In-plant operations personnel are expected to receive approx. one to two rem per year per man for a maximum total of 58 man-rem per year. The remaining 51 ~~[illegible]~~ man-rem per year ~~will be~~ may be expected to be received by non-station personnel. As part of this total, the supervisors and control room staff are expected to receive an exposure of less than 500 mrem/yr. ~~If they remain in the control room and administrative office areas. These people will, however,~~

c. Group 3- This group Includes Health Physics / Chemistry technician personnel. There are approximately 8 people in this group. If the plant health physics / chemistry personnel spend 1% of their time collecting samples in Zone III sampling stations, they will receive a maximum dose of 300 mrem / yr. Assuming the remainder of their time is spent in Zone I & II areas, the total dose is between 1 and 2 rem per person. The plant health physics / chemistry personnel also conduct radiation surveys and support maintenance activities which require continuous and pre-job radiation surveys. The exposure to these health physics / chemistry personnel ranges from 2 to 4 rem / yr. This is based on experience from operating plants. Assuming a dose of 3 rem per person per year and considering 8 people in the group, the total is 24 man-rem per year. [INSERT]

d. Group 4- This group Includes engineers and technical supervisors. There are approximately 9 people in this group. Personnel in this group will spend most of their time in Zone I areas where exposures are less than 500 mrem / yr. Table 12.4-7 indicates approximately 129 man-rem per year will be experienced for inservice inspection. Plant technical personnel will have a supervising roll in this operation with non-station personnel performing the inspecting operations. Thus, the projected dose estimate for the 9 people in this group is 4.5 man-rem per year, the balance being accounted for by the non-station personnel

e. Group 5- This group Includes station supervisors such as health physics-Chemistry supervisors, shift supervisors, etc. There are approx.

[INSERT]

Since this group covers virtually all ~~maintenance~~ functions delineated in Tables 2 through 8, this 24 man-rem is considered to be spread out across all the functions.

supervise Group 1 and Group 4 personnel. Their dose is approx. the same as personnel in these groups. With a projected dose estimate of 1 rem per year per and 14 people in the group. the total dose is 14 man-rem per year.

f. __Group 6 -__ This group includes administrative and management personnel. There are approximately 11 people in this group. Personnel in this group spend their time in Zone 1 radiation areas. The projected dose estimates will be less than 500 mrem/yr. With 11 people in this group and a 500 mrem per man per year the total dose is 5.5 man-rem per year.

The total personnel dose for all groups is approx. 647 man-rem per year.

~~12.4.2.3 Personnel Dose from Operating BWR Data~~

~~References 1 and 2 provide a tabulation of personnel exposures for operating BWR's. Table 12.4-9 tabulates the average personnel exposure for several plants operating for a period of several years. Tables 12.4-1 through 12.4-8 were derived from information contained in references 4 through 7.~~

¶ As seen from Table 12.4-1, the total estimated man-rem exposure is 613 man-rem. Groups 3, 5, and 6 are considered to be spread over all the functions. These groups constitute only 7% of the total exposure in any case.

12.4.4 SITE BOUNDARY DOSE

Steam-handling equipment on the turbine operating floor can contribute to the site boundary dose in two ways: through a direct component and through an air-scattered "skyshine" component. Since the N-16 bearing equipment is known, it can be shielded to reduce the direct component. The "skyshine" component reaches the site boundary as a result of those gamma rays which are directed such that they bypass any intercepting shield walls and are scattered by the air to the site boundary.

The calculated results show that the skyshine dose will have its greatest effect on a dose point 1950 meters north of the turbine building. The skyshine dose at this point will be approximately 3.6 mrem/yr. This result is based on a plant capacity factor of 80% at full power operation.

The main contributors to this dose and their contribution (in percent) are the south moisture-separator reheater (MSR) which contribute 60%, the north MSR which contributes 20%, the cross over lines which contribute 10% and the turbines and feedwater heaters which contribute 10%.

The dose estimate was computed from a model that represents the N-16 gamma leakage by point isotopic sources. This model uses the output from the COHORT Code[3] which gives the air-scattered dose as a function of distance and source ray angle.

The site boundary dose from liquid and gaseous effluents are discussed in 11.2.3 and 11.3.3.

12.4.5 REFERENCES

~~12.4-1~~ 12A-1 Atomic Industrial Forum, Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, September 1974.

~~12.4-2~~ 12.4-2 USAEC, A Compilation of Occupational Radiation Exposure From Light Water Cooled Nuclear Power Plant 1969-1973, WASH-1311.

~~12.4-3~~ 12.4-3 Wells, J. B., Collins, D. G., and Neuendorf, W. P., Boundary Dose Rates Due to Gamma Rays at Power Reactor Sites, Report RRA-7202, Radiation Research Associates (November 12, 1972).

12A-4

Vance J, Weaver C. L., Lepper, E. M. A preliminary Assessment of the Potential Impact on operating Nuclear Power Plants of a 500 mRem Occupational Exposure Limit, Atomic Industrial Forum, Washington D. C. April 1978.

12A-5 Murphy, T. D., Dayem, N. J., Bland, O. J., Pasciah, W. J., Occupational Radiation Exposure at light water cooled power reactor 1969-1975, VS. NRC, NUREG 0109, Washington D. C., April 1976.

12A-6

Dickson, H. W., Cottrell, W. D., Jacobs, D. C., Application of ALAD concept to exposure of workers at light water reactors, 2 RNC, TM-5126, Oak Ridge Tennessee, November 1975.

12A-7

Ninth Annual Occupational Radiation Exposure Report, USNRC, NUREG 0322, Washington D. C., October 1977.

TABLE 12.4-1

ESTIMATED ANNUAL DOSE TO PERSONNEL

Personnel Group	Number of People in Group	PLANT IN OPERATION: REACTOR BLDG. Occupancy Factor (hr/man/yr)	REACTOR BLDG. Personnel Dose (man-rem)	TURBINE BLDG. Occupancy Factor (hr/man/yr)	TURBINE BLDG. Personnel Dose (man-rem)	RADWASTE BLDG. Occupancy Factor (hr/man/yr)	RADWASTE BLDG. Personnel Dose (man-rem)
Maintenance Craftsmen (Group 1)	46	20	4.60	20	4.60	60	27.60
Operators (Group 2)	29	50	4.35	50	4.35	50	7.25
Technicians (Group 3)	8	50	1.20	50	1.20	50	2.00
Engineers (Group 4)	9	8	.22	8	.22	4	.18
Plant Supervision (Group 5)	14	35	1.46	35	1.46	30	2.10
Management (Group 6)	11	1	.03	1	.03	1	.05
TOTAL	117		11.86		11.86		39.18

All Table to be replaced with Attached

Table 12.4-1

SUMMARY OF

OCCUPATIONAL DOSE ESTIMATES

AT WNP-2

		Man Rem/Year	
1.	Routine Operation and Surveillance	~~[illegible]~~	38
2.	Non-Routine Operation and Surveillance	~~[illegible]~~	15
3.	Routine Maintenance	~~[illegible]~~	~~[illegible]~~ 275
4.	Waste Processing	~~[illegible]~~	11
5.	Refueling	~~46.5~~	47
6.	Inservice Inspection	~~137~~	~~[illegible]~~ 129
7.	Special Maintenance	~~121.8~~	~~[illegible]~~ 98
	~~Total~~ Total	~~645.2~~	~~[illegible]~~ 613

Table 12.9-2

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE OPERATIONS AND SURVEILLANCE AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
1. Walking	.5	.5	2	1/shift	.54
2a. Checking					
Railroad Access					
Change Rooms					
Relay Room					
Motor Generator Sets					
Battery Room					
Computer Room					
Switch Gear Room					
Air Conditioning Equip.	1	1	2	1/shift	2.2
Recirc. Motor Gen.					
SGTS					
HPCI Turbine & Pump					
RBCCW Heat Exchangers					
Emergency Air Comp.					
[illegible]CU Pumps					
[illegible]W Expansion Tank					
b. CRD Pumps					
CRD Hydrolic Control Units					
Refueling Flear					
CRD Filters					
RUCV Demmo Resin Tanks	10	.5	.2	1/shift	11
RNP Pumps					
SRMP Pumps					
Air Coolers					
IVST Racks					
CRD Storage & Repair					
c. RWCU Heat Exchangers					
RHR Heat Exchangers	50	.1	1	1/shift	5.5
Acid Purple & Turbine					

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
2. Checking (cont.)					
d. Demin Precoat Tank Precoat Pump Waste Sample Pump Floor Drain Sample Room Waste Surge Pump Equip. Drain Sump Pump Waste Surge Pump Equipment Drain Sump Pump Waste Precoat Pump Waste Sludge Discharge Pump Waste Filter Aid Pump Chemical Waste Pump Floor Drain Collection Pump	0.2	0.5	1	1/shift	0.1
e. ...ical Waste Tank ...t Resin Pump Condensate Phase Decant Pump Condensate Phase Sludge Discharge Mixing Pump	50	0.5	1	1/week	1.3
f. Floor Drain Demin. Waste Hopper Floor Drain Filter	8	2	1	1/week	0.8
g. Turbine Instruments & Controls Gen. CO_2 Units Station Air Comp. Heater Feed Pumps Demineralize Pumps & Valves MTG Lubrication System Hatch Area above Demin. Tanks H_2 Seal 2.1 Equip. Health Shell Pull Space ...CW Heat Expansion Pumps	.5	1	2	1/shift	1.1

Table 12.4-3 (page 3)

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
2. Checking (Cont.)					
h. TBCCW Expansion Tank; Ventilation Equip.; Demin. Precoat & Resin. Tanks; Demin. Precoat Pumps; Sump Pumps; Reactor Feed Pump Turbine; Lub. System; MTG Lub Oil Cooler; Main Gen. & Exciter; MTG Utilizer Activators; Stop & Thrittle Valves	5	0.3	1	1/shift	1.6
i. Heater Drain Pumps; Heater Drain Flash Tanks; Condense Water Box; Circ. Water Isolation Valves; Reactor Feed Pumps & Turbines; Drain Coolers; Med. Vacs Pumps; Feed Water Heaters; Reheater Seal Tank; Gland Steam Condenser; Main Turbine; Reheater Separators	2.5	0.5	1	1/shift	13.7
				TOTAL	37.8

Table 12.4-3

OCCUPATIONAL DOSE ESTIMATES DURING NONROUTINE OPERATION AND SURVEILLANCE AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
1. Operation of Equipment:					
a. Traversing In-Core Probe System	2	2	2	3/year	0.02
b. Safety Injection Sys.	5	1	1	1/month	~~0.30~~ 0.06
c. Feedwater Pumps & Turbine	1	1	1	1/week	0.05
d. Instrument Calibration	2	1	1	1/day	0.73
2. Collection of Radioactive samples:					
a. Liquid System	10	0.5	1	1/day	1.83
b. Gas System	5	0.5	1	1/month	0.03
c. Solid System	10	0.5	1	4/year	~~0.02~~ 0.01
d. Radiochemistry	1	1	2	1/day	0.73
e. Radwaste Operation	3	8	3	1/week	3.75
f. Health Physics	5	2	2	1/day	7.30
~~Total~~				Total	14.5

Table 12.4-4

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE MAINTENANCE
AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Numbers Of Workers	Freq.	Dose Man-Rem/Year
~~Maintenance~~					
1. Minor Repairs Reactor Building	.1	20	2	1/week	2.1
2. Ventilation & Air Conditioning	0.5	20	1	1/week	0.5
3. Control Rod Drive Repair*	15	200	6	1/year	18
4. Reactor Water Clean Up Pump*	180	35	3	1/year	19
5. Reactor Water Clean Up Valve & Heat*	110	45	6	1/year	30
6. Exchanger					
7. Residual Heat Removal System*	200	27	8	1/year	43
8. Safety Relief Valves	80	30	5	1/year	~~[illegible]~~ 12
9. Main Steam Isolation Valves	75	100	6	1/year	45
10. Recirc. Pumps	200	50	3	1/year	30
11. Snubber Inspector & Repair	75	100	5	1/year	37.5
12. Misc. Turbine Bldg. Repairs	2	8	1	1/day	5.8
13. Reactor Feed Pumps & Turbine	2	40	2	6/year	0.96
14. Drain Coolers	2	40	2	1/year	0.16
15. Steam Jet Air Ejectors	2	40	2	2/year	0.32
16. Off Gas System	2	40	2	6/year	0.96
17. MTG Actuator	5	40	1	1/year	~~20~~ .24
18. Heater Drain Flash Tanks	2	40	1	1/year	0.08
19. Condensor Water Box	5	20	1	1/year	0.1
20. Annual Turbine Inspection	3	120	10	1/year	3.6
21. Misc. Radwaste Pump Repairs	5	40	2	6/year	2.4
22. Misc. Radwaste Valve Repairs	5	40	2	6/year	2.4
23. Filter & Demin.	65	30	3	1/year	5.9
24. Centerfuge	5	8	2	4/year	0.32

Table 12.4-4 (page 2)

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
Mechanical (Cont.)					
25. Evaporation	85	50	3	1/year	12.8
26. Turbine Instr. & Control	2	10	1	1/week	1.0
27. Waste Solidification	2	40	2	2/year	0.32
28. Area Monitors	20	40	2	2/year	0.32
~~Total~~				Total	~~[illegible]~~ 274.8

Table 12.4-5

OCCUPATIONAL DOSE ESTIMATES DURING WASTE PROCESSING AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
1. Radwaste Control Room	.5	8	1	1/shift	4.4
2. Sampling & Filter Changing	10	4	1	1/week	2.1
3. Panel Operator Insp. & Testing	1	2	1	1/day	.73
4. Operation of Waste & Packaging Equip.	2	16	2	1/week	3.3
~~Total~~				Total	~~10.93~~ 10.5

Table 12.4-6

OCCUPATIONAL DOSE ESTIMATE DURING REFUELING AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
1. Opening/Closing Reactor*	60	40	10	1/year	24
2 Pressure Vessel					
2. Fuel Preparation	10	24	2	1/year	0.48
3. Refueling *	10	100	15	1/year	15
4. Fuel Handling	2.5	100	4	1/year	1.0
5. Fuel Sipping	10	100	6	1/year	6.0
~~Total~~				Total	~~46.8~~ 46.5

Table 12.4-7

OCCUPATIONAL DOSE ESTIMATES DURING INSERVICE INSPECTION
AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
1. Removal/Replacement of Insulation	150	40	4	1/year	24
2. Installation/Removal & Ladders	50	40	4	1/year	8
3. Inspecting Inside Dry Well*	150	80	6	1/year	72
4. Recorder Data	50	80	6	1/year	24
5. Inspecting Outside Dry Well*	5	50	2	1/year	.5
~~Total~~				Total	129

Table 12.4-8

OCCUPATIONAL DOSE ESTIMATES DURING SPECIAL MAINTENANCE AT WNP-2

Activity	Ave. Dose Rate mRem/Hr.	Exposure Time Hrs.	Number Of Workers	Freq.	Dose Man-Rem/Year
Sparger Replacement	800	60	5	Should not be necessary	—
CRD Replacement	260	35	5	1/year	45.5
Turbine Overhaul	3	250	20	1/5 year	3
Servicing In Core Detectors	15	50	3	1/year	2.3
Off Gas Charcoal Sys. Overhaul	100	100	2	1/20 year	1
Special Maintenance Reactor	150	100	4	1/10 year	6
Water Clear Up Sys.					
Misc. Piping Repair's	80	100	5	1/year	[illegible] 40
Total					~~121.8~~ 97.8

(old Table 12.4-2)

TABLE 12.4-9

PERSONNEL EXPOSURE FOR SEVERAL BWR PLANTS

PLANT #3 RATED MWe: 640

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS MAN-REM	TOTAL MAN-REM
73	5	452,708	2263	142	551	1449
72	4	540,877	1548	108	399	651
71	3	486,380	1567	98	140	249
70	2	441,800	1687	88	48	64
69	1	49,806		69	7	13
68				67	.3	.3

PLANT #5 RATED MWe: 630

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS MAN-REM	TOTAL MAN-REM
73	4	457,173	2679	136	310	594
72	3	417,109	2678	130	218	305
71	2	381,082	2798	68	106	206
70	1	247,501	4487	69	26	62

* Regulars - Denotes the number of Regular (Non-Contractor) Plant Employees

TABLE 12.4-9 (Continued) Sheet 4

PLANT #13 RATED MWe: 652

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS* MAN-REM	TOTAL MAN-REM
73	3	58,082	8040	176	225	620
72	2	403,650	3,960	232	255	595
71	1	463,000	3240	244	31	49
70		11,988				

* Regulars: Denotes the number of Regular (Non-Contractor) Plant Employees

TABLE 12.4-9 (Continued) Sheet 2

Plant #8 RATED MWe: U1 200
U2,3 800

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS* MAN-REM (all units)	TOTAL MAN-REM (all units)
73	14	U1 101,353	2332		576	909
73	4	U1 681,174	807			
73	3	U3 495,689	2577			
72	13	U1 156,783	1726	239	368	728
72	3	U2 432,725	3402			
72	2	U3 618,888	1144			
71	12	U1 99,078	3140	225	315	715
71	2	U2 364,023	2669			
71	1	U3 149,510	5944			
70	11	U1 198,835	498	202	127	143
69	10	U1 120,493	3292	182	215	
68	9	133,307	3177	189	303	
67	8	115,362	3855	170	363	
66	7	199,214	368	107	150	
65	6	138,149	1800	103	128	
64	5	138,688	1547	71	90	
63	4	130,757	1992	89	108	

* Regulars - Denotes number of Regular (Non-Contractor) Plant Employees

TABLE 12.4-9. (Continued) Sheet 3

PLANT #8 RATED MWe: U1 200
(Continued) U2,3 800

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS* MAN-REM (all units)	TOTAL MAN-REM (all units)
62	2	168,008	1716	182	86	145
61	2	72,403	4500		105	105
60	1	31,707			64	64

PLANT #9 (1) RATED MWe: U1 800
U2 800

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS* MAN-REM	TOTAL MAN-REM
73	2	U1 646,112	970	259	142	201
73	1	U1 676,909	1074			
72	1	U1 110,733	3502	380	26	64
72		U2 205,080				

PLANT #10 RATED MWe: 548

CAL YEAR	PLANT AGE (YRS)	THERMAL MWD	DOWN HOURS	REGULARS* #	REGULARS* MAN-REM	TOTAL MAN-REM
73	3	405,000	2131**	105	91	156
72	2	445,000	1419	83	63	65
71	1	180,000	4367	82	27	29

**First six months only

* Regulars - Denotes the number of Regular (Non-Contractor) Plant Employees

Q 331.21
(12.4.2)

The annual exposure estimates given for the job group classification in Section 12.4.2.3 of the FSAR do not agree with the total group exposures listed in Table 12.4-1. Clarify this apparent discrepancy.

RESPONSE:

More historical data from operating BWR's has recently become available. Hence, sections 12.4.1 and 12.4.2 are being revised. See the response to question 331.20. In any case, the previous information in Table 12.4-1 was an estimate based on occupancy times and average radiation dose rates. The previous discussion in 12.4.2.3 referred to BWR operating experience then available.

Q. 331.22
(12.5.2)

Discuss the provisions for laundering soiled and radioactively contaminated protective clothing at the WNP-2 facility.

RESPONSE:

Drawing M582 of change room facilities in the Service Building is included. Facilities for the increased number of personnel during major outages will be provided by using mobile dressing facilities.

Q. 331.23
(12.5.2)

Discuss the provisions for local exhaust systems in the hot machine shop when work is being performed on items contaminated by radioactivity.

RESPONSE:

Two provisions are intended for use. They are:

a. The existing HEPA filtered vacuum system (negative blower) permanently installed in the shop. Adapters(longer hoses) will be attached with the restrictive limitations being system efficiency and obstructions to others work in the area.

b. Portable vacuum systems are available and are intended for use. These units are HEPA filtered.

QUESTION: 331.24 Discuss the provisions for laundering soiled and (12.5.2) radioactively contaminated protective clothing at the WNP-2 facility.

RESPONSE:

Washington Public Power Supply System has obtained an agreement with the Department of Energy which will allow the supply system to contract the services of the Rockwell Hanford Operations laundry located on the Hanford Reservation. The RHO laundry has many years of experience in providing complete laundry services.

The Supply System is however investigating all alternatives to determine the most effective and efficient means of laundering contaminated protective clothing. Alternatives under consideration include commercial contractor service and leased or purchased equipment, utilizing both wet and dry methods.

Responses to:

Hydrology - Meteorology Branch Questions
Hydrology Section (371.8 - 371.14)
Meteorology Section (372.8 - 372.17)

Q 371.8

As requested in Item 371.1 of the first acceptance review questions issued on June 24, 1977, provide a detailed post-construction topographic map of the plant area with particular emphasis on the locations of streams, ditches, and drainage structures, Where any drainage structures, including roof openings, are relied upon to convey runoff from the local Probable Maximum Precipitation (PMP), provide pertinent details of these structures, including their size, slope, elevation, and cross-sectional area.

RESPONSE:

A detailed post-construction topographic map of the plant area is not completed at this time. It is expected that the drawings of the final site grading and drainage features will be developed by October 1979 so that a contract for this work can be let in 1980.

All safety-related facilities are located above the elevation of the Probable Maximum Flood (PMF) produced by the PMP. Final site grading will assure that the rainfall is carried away from the plant. Discharge from roof drains, and non-radioactive floor and equipment drains is carried by pipeline to an outfall area approximately 1200 feet from the main plant structure and at an elevation approximately 40 feet lower than the final plant grade.

Q 371.9

Your analysis of the Probable Maximum Flood (PMF) discharge on the stream with the 38.5-square-mile drainage area is incomplete. In particular, provide additional information regarding the analyses which you have performed and the bases for the PMF hydrograph shown in Figure 2.4-12 of the FSAR, including: (1) verification of historical flood flows, if applicable; (2) the time of concentration; (3) the duration of rainfall for the unitgraph; and (4) all other assumptions.

Response:

The Probable Maximum Flood (PMF) generated by the PMP event has been recalculated. Refer to Sections 2.4.3.2 and 2.4.3.3 revised.* Verification of historical flood flows and duration of rainfall for a unitgraph is not possible, since this is an ungaged drainage basin.

*See the attached draft FSAR changes.

A topographic map and contour map of the region surrounding the site are shown in Figures 2.4-2 and 2.4-~~3~~10. Figure 2.1-4 shows details of contours in the main plant area.

The natural drainage features of the surrounding area have not been changed by the construction of WNP-2.

2.4.1.2 Hydrosphere

The Columbia River, the largest river flowing into the Pacific Ocean from North America, is one of this world's greatest sources of hydroelectric power. Its annual discharge of 18,000,000 acre ft. (1 acre-ft = 43,560 cu ft) is exceeded in the North American continent only by the Mississippi, Mackenzie and St. Lawrence Rivers.

The Columbia River drains an area of approximately 258,000 square miles, lying to the west of the Continental Divide in the northwestern part of the United States (85%) and South-western part of Canada (15%). Major tributaries are the Kootenay, Snake, Pend Oreille, Spokane, Okanogan, Yakima and Willamette Rivers.

In determining the Standard Project Flood the drainage area was divided into subbasins. These subbasins can be grouped into six general areas with similar hydrometeorological characteristics.

The six (6) areas are: (1) upper Columbia, which includes the drainage of the area in Canada and the northern part of the United States above Chief Joseph Dam; (2) Middle Columbia, which includes the area between Pasco and Chief Joseph Dam; (3) Upper and Middle Snake River; (4) Lower Snake River, the area between Weiser and Ice Harbor Dam; (5) Lower Columbia, including the area between Bonneville Dam and Pasco; (6) the Columbia below Bonneville Dam, including the Willamette River.

The river basin has five (5) outstanding physical features: the Rocky Mountain System, the Columbia Plateau, the Columbia River Gorge, the Cascade Range and Puget Trough.

The Rocky Mountain System is the major range with elevations from 2,000 to over 12,000 feet. There are permanent glaciers and extensive snow fields at higher elevations and deep valleys that provide the principal drainage for the head-waters of the Columbia, Kootenay and other rivers.

Tri-Cities Service Area

Projected Municipal Water Use

Year	Use (MGD)
1970	22.5
1980	28.5
2000	46.0
2020	66.2

The values for water rights given in Table 2.4-2 for the cities of Richland and Pasco are 37 MGD (57 cfs) and 23 MGD (35 cfs) respectively. These values exceed the ones given by the study made by the Pacific Northwest River Basin Commission for the Tri-Cities Service Area.

At the present time, there are no groundwater users on either side of the river in the vicinity of the site. Consumptive use of water upstream from Hanford reach is primarily associated with irrigation development. The description and sources of groundwater are discussed in 2.4.13.1 and 2.4.13.2, respectively.

2.4.2 FLOODS

2.4.2.1 Flood History

Floods in the Columbia River Basin are grouped as:

a. the interior basin east of the Cascades, caused by melting snowpack and occurring from May through June;

b. the Willamette and other basins, west of the Cascades, caused by direct runoff from intense winter rain occasionally augmented by snowmelt.

There is some overlapping effect within these two groupings. At certain elevations, basins in the interior Columbia drainage area occasionally have significant flood flows resulting from winter rain or snowmelt. These are local floods and do not usually contribute sufficient flow to cause flooding of

the main Columbia River. Major floods on the Columbia River Basin result from rapid spring melting of the snowpack over a wide area, generally augmented by rain, or by above-normal precipitation in May, accompanied by a major chinook wind which causes rapid area temperature rise. The annual spring snowmelt flood of the main interior basin is characterized by relatively uniform distribution over the basin. The snowfall and individual snow storms may vary, but the integration of all storms over the winter period smoothes the irregularities, with the result that the distribution of the flood runoff is reasonably constant from year to year.

The maximum historical flood of record is that of June 7, 1894 which resulted from a combination of hydrometeorologic conditions, including heavy snowpack and rapid melt plus rainfall. The peak discharge at WNP-2 was 740,000 cfs for the Columbia River, as estimated from high water mark at Wenatchee, Washington(2). The largest recent flood, occurring in 1948, had an observed peak discharge of 690,000 cfs at Hanford. These floods were spring floods resulting from the melt of a large snowpack combined with the spring rains(3).

There is no record of ~~major~~ flooding in the immediate site area. ~~due to ice jams, nor have there been any recorded flood conditions due to dam failure.~~

~~2.4.2.1.1 Flood Conditions on the Columbia River at the Site~~

~~The flood conditions for the Columbia River applicable to the site were determined by the Corps of Engineers(3).~~ Water surface profiles for the Columbia River in the vicinity of the site as derived by the Corps of Engineers(3) are given in Figure 2.4-8.

~~2.4.2.1.2 Plant Site Area Flooding~~

The plant site is located approximately 3 miles west of the Columbia River at River Mile 352 with reactor floor elevation of 441.0 ft. MSL, which is 68 feet above the water level estimated for the largest historical flood (approximately 373.0 ft. MSL). ~~The site area is not subject to flood from rain. The plant is located only three miles from the river, and the terrain is sloping in the direction of the river, thus providing good drainage paths to it. The projected post-construction topographic map of the WNP-2 site is shown in Figure 2.1-4 and the general drainage area for WNP-2 is shown in Figure 2.4-9. Details of the locations of streams, ditches, culverts, and other site drainage features are presently under design and will be provided at a future date.~~

~~The entire area where the reactor building and spray ponds are~~ locat~~ed drains~~ to a broad channel that extends in the ~~north-~~ south directi~~on for~~ about seven miles and ~~ranges~~ from about 2,000 feet to over a ~~mile~~ in width ~~(See~~ Figure 2.4-2). The reactor building and spray ~~ponds~~ are located on high ground to the west of the ~~channel~~ and over 30 ~~feet~~ above the channel low point~~s.~~ At a point about six miles south ~~of the~~ reactor ~~site~~, the channel curves towards the east and drains ~~to the Columbia River.~~

~~The project facilities have been designed to be safe from flood based on the criteria outlined in 2.4.2.2.~~

2.4.2.2 Flood Design Considerations

Flood protection of safety-related components is based on the highest calculated flood water level ~~elevation~~, including wave effects, resulting from ~~several hypothetical events.~~ intense local precipitation. Several different probable maximum events were considered, including the Corps of Engineers design-project flood considered to be "the most severe reasonably possible." Wave action caused by storm winds, the effects of failure of upstream dam surge flooding and ice flooding were also considered.

The results of these analyses (described in 2.4.3) indicate that the plant site for WNP-2, located 3 miles west of the Columbia River at a ground elevation of approximately 440 ft/ MSL, the reactor first floor at elevation 441 ft/ MSL, and the spray ponds with a minimum wall elevation of 435 ft/ MSL are safe-from-flood.

~~The analysis for Probable Maximum Flood was done by the Corps of Engineers(3) in accordance with Engineering Circulars 1110-2-27 and 1110-2-34 as described in Regulatory Guide 1.59, Rev. 0 (Issued August 1973) and its Appendix A.~~

2.4.2.3 Effects of Local Intense Precipitation

Intense local summer thunder storms can produce short duration rains which have the potential for causing serious flood. The probable maximum precipitation event for the WNP-2 site has been determined using the methodology developed by the U.S. Weather Bureau and reported in Hydrometeorological report No. 43, "Probable Maximum Precipitation, Northwest States" (4).

The plant ~~entire~~ area drains easterly to a broad channel which is adequate to store and drain the probable maximum precipitation (PMP) flows (See Figure 2.4-3~~8~~). The reactor building and the spray ponds are located at elevations that are safe from the effect of any flood resulting from the maximum precipitation event.

The projected post-construction contours of the WNP-2 site are shown in Figure 2.1-4. Details of the locations of streams, ~~2.4-7~~ ditches, culverts, and other site drainage features are still in the design stage.

Winter precipitation may occur as rain or snow. The winter season snowfall has ranged from less than one-half inch to a maximum of 12 inches in December 1964. There is no ice accumulation at the site.

Roofs of buildings are designed to take, with adequate drainage, any instantaneous or local intense precipitation. Discharge from roof drains is carried by means of a storm sewer system to a manhole located southeast of the reactor building. From that point a pipeline with a northeast alignment transfers the discharge to a low point of disposal about 1,500 ft. away from the plant site.

The roofs of safety-related buildings are concrete beam and slab construction except the high roof of the reactor building which is metal deck on steel framing. The minimum roof slope for all structures is 1/8 inch per foot for adequate drainage and the roof areas are encompassed by curbs or parapet walls up to 3 feet 6 inches high. Roof plans, including details of roof drains and overflow scuppers are provided in Figure 2.4-36. Assuming that the roof drains are completely blocked during the probable maximum precipitation (PMP) event, overflow scuppers limit the depth of water to within the design load carrying capability of the roofs. Those safety-related structures that do not have this relief capability structurally can carry the entire PMP accumulations.

2.4.3 PROBABLE MAXIMUM FLOOD (PMF) ON STREAMS AND RIVERS

Analyses for Probable Maximum Flood on the Columbia River done by the Corps of Engineers[3] are consistent with the requirements of Regulatory Guide 1.59, Rev. 0, and its Appendix A.

The Standard Project Flood (SPF) for the Mid-Columbia Reach of the highly developed and regulated Columbia River has been defined by the Corps of Engineers[5] as 570,000 cfs. The unregulated Standard Project Flood for the same reach is 740,000 cfs.

The unregulated Probable Maximum Flood (PMF) at the site was derived[6] by the North Pacific Division, Corps of Engineers[3]. The adjusted flow for the Hanford region is given in Reference 5 as 1,600,000 cfs, for the unregulated PMF.

The methods of estimating peak flow rate and the controlled PMF hydrograph above Pasco are given in the above referenced report. Adjustment of the flood profiles for the Hanford region reported in Reference 5, results in a regulated PMF of 1,440,000 cfs and a water level of 390 feet at the Seismic Category II makeup water structure. This structure is not designed to function throughout the PMF but is designed on the basis SPF (unregulated) of 740,000 cfs. The water level at the intake structure (R.M. 351.7) was derived by interpolating the Corps of Engineers drawing dated November, 1970, titled "Columbia River Washington Water Surface Profiles, RM 323 to RM 395", sheet 2 of 2, and is given in Figure 2.4-8.

Although assumed to exist for the purpose of flood hydrograph calculations, the Corps of Engineers states[7] that the Ben Franklin dam is not a federally authorized project but as originally planned it would have been a low head dam which would have had only a negligible effect on extreme flood flows.

Presently there are no authorized river projects which affect the water level at the site.

The design basis for the WNP-2 site area results from the probable maximum precipitation event on the adjacent drainage basin and not from flooding of the Columbia River.

2.4.3.1 Probable Maximum Precipitation (PMP)

The Probable Maximum Precipitation (PMP) event which was presented in the WNP-2 PSAR was subsequently reevaluated in the preparation of the PSAR for WPPSS Nuclear Project No. 1 (Docket 50-460). The analysis presented here is consistent with the latter document.

Precipitation in the vicinity of the site has been classified by the U.S. Weather Bureau, Reference 4, as convergence precipitation, orographic precipitation, and thunderstorm precipitation. The methodology for predicting the total amount of precipitation from each of these events, as given in Reference 4, requires the adding together of the convergence PMP and the orographic PMP to obtain a single precipitation for a general storm. A separate analysis is then required for thunderstorms. Thunderstorms in the vicinity of the site can be locally very intense for short periods of time and hence, have the potential for causing serious flooding. The PMP for both a general storm and a thunderstorm were analyzed as given in Chapters 6 and 5, respectively, of Reference 4 for a 38.5 square mile basin at the site. This basin is shown in Figure 2.4-~~8~~ 3 and is described in 2.4.3.3. The calculated general

storm PMP results in a 24 hour and 48 hour precipitation of 7.9 inches and 10.1 inches, respectively. A thunderstorm PMP yields 9.2 inches in a 6 hour period. Therefore, the thunderstorm is considerably more severe. The thunderstorm PMP hydrograph is:

Hour	Inches of Rain
1	0.6
2	1.6
3	5.2
4	0.9
5	0.5
6	0.4
Total	9.2

Insert A → ~~2.4.3.2 Precipitation Losses~~

Infiltration losses have been estimated (7a) in the vicinity of the sites as 1.5-2 inches/hour. Although these rates are extremely high, to insure conservatism, the analysis below does not consider infiltration losses.

2.4.3.3 Runoff and Stream Course Models

The projected post-construction topographic map of the WNP-2 site is shown in Figure 2.1-4 and the drainage basin common to the reactor buildings and spray ponds is shown in Figure 2.4-9. The entire area drains to a broad channel that extends in a north-south direction for about seven miles and ranges from about 2000 feet to over a mile wide. All plant structures are located on high ground to the west of the channel and over 30 feet above the channel low points. At a point about six miles south of the reactor site, the channel curves toward the east and drains to the Columbia River.

The channel is constricted at a point just southeast of the reactor site; see dotted line on Figures 2.4-3 and 2.4-9 (approximate coordinates N10800, E0300). About 70% of the PMP for the 38.5 square mile basin drains into the channel north of the restriction. Figure 2.4-10 shows cross sections of the channel. The storage capacity and the cross sectional area of the channel as a function of the pond surface elevation north of the restricting channel is shown ~~in Figure 2.4-11.~~

INSERT (A)

2.4.3.2 Precipitation Losses

Infiltration losses have been estimated in the vicinity of the sites as 1.5-2 inches/hour (7a). However, for the analysis below, an average antecedent moisture condition (Condition II as defined in Reference 2.4-51) was assumed. As explained in the following section, the 60-minute retention loss rate is 0.15 in/hr.

2.4.3.3 Runoff and Stream Course Models

The projected post-construction topographic map of the WNP-2 site is shown in Figure 2.1-4, and the drainage basin common to the reactor buildings and spray ponds is shown in Figure 2.4-3. The entire area drains to a broad channel that extends in a north-south direction for about seven miles, and ranges from about 2000 feet to over a mile wide. All plant structures are located on high ground to the west of the channel. At a point about 2.8 miles south of the reactor site, the four-lane DOE highway crosses the drainage basin. The area above this section is 33.2 square miles.

To evaluate the effect of the PMP event on the plant area, the peak discharge at the highway crossing, 2.8 miles downstream of the plant, was calculated using the U.S. Bureau of Reclamation procedure for computing design floods on ungaged basins from thunderstorm rainfall in the Western U.S.[51] Important assumptions used in the triangular hydrograph procedure of Reference 2.4-51 are:

1. Hydrologic soil group B
2. Land use and treatment class - poor pasture or range
3. Thunderstorm cover-index is brush-sage-grass combination with 50% or less cover density
4. Thunderstorm minimum 15-minute rentention loss rate of 0.06 in/15 min. and 60-minute retention loss rate of 0.15 in/hr.

Additionally, no credit was taken in the hydrograph analysis for potential storage in the stream channel or upstream sub-basins.

The time of concentration, T_c, for the watershed above the highway crossing was computed to be 7.5 hours. The PMF hydrograph is shown in Figure 2.4-9 for the 33.2 square mile drainage basin. A peak discharge of 21,400 cfs was determined.

Based on this PMF, an upstream water surface profile was determined using the Corps of Engineers HEC Standard-Step Procedure.[52] A total of eleven cross-sections were used (7 downstream, one at the plant, and 3 upstream as shown in Figure 2.4-10). Details of the channel cross-sections are shown in Figure 2.4-11. The Manning roughness coefficient was conservatively taken as n=0.035 in the main channel sections, and n=0.05 in the overbank areas.

Using the computational procedure of Reference 2.4-52, it was determined that the channel restrictions at cross-sections 5 and 7 (Figure 2.4-10) do not control the flow. The still-water elevation at the plant site (cross-section 8) was determined to be 431.1 ft. MSL. The water surface profile is shown in Figure 2.4-12

2.4.3.4 Probable Maximum Flood Flow

The PMF runoff hydrograph produced by the PMP at cross-section 1 (Figure 2.4-10) is shown in Figure 2.4-9. The peak discharge at this location is 21,400 cfs.

2.4.3.5 Water Level Determinations

As discussed in Section 2.4.3.3, the water elevation of a flood at the plant site generated by the PMP event is 431.1 ft. MSL. This flood condition is more severe than any flood of the Columbia River.

2.4.3.6 Coincident Wind Wave Activity

Procedures provided in the Corps of Engineers Shore Protection Manual (Reference 2.4-53) and ETL 1110-2-221 (Reference 2.4-8) were used to determine the wind wave activity.

The effective fetch for the predominant July wind direction (north) is 3450 feet (0.65 mi). The effective fetch diagram is shown in Figure 2.4-1~~2~~3). The calculated extreme 2-year over water wind for the north-to-south direction, based on area data, is 63.5 mph. This wind results in a maximum wave height of 4.0 feet, with the assumption of a water depth of 12 feet (the average depth in cross-sections 8, 9, and 10). The other potential wind directions ENE and ESE were evaluated but found to be less severe.

The wind setup has been computed to be 0.3 ft., and the maximum wave runup is 1.9 ft. on a smooth, 1 on 8 slope of compacted naturally occurring sands and gravels. Therefore, the design water surface elevation is 433.3 ft. MSL. This is less than the east spray pond elevation of 435.0 ft. MSL.

~~In order to evaluate the flow through the restricting~~ channel, a conservative analysis was performed using the PMP hydrograph given in 2.4.3.1. In this analysis it was assumed that no infiltration losses to the ground occurred. It was secondly assumed that the runoff to the pond area was instantaneous, that is, as soon as the rain fell according to the PMP hydrograph, that volume was added to the pond. The third assumption was that there was no scouring of the outlet channel shown in Figure 2.4-10. As discussed in 2.4.3.2, the assumption of no infiltration losses is very conservative.

Flow through the restricting channel depends upon a number of factors, which include the energy grade line, the effective hydraulic radius and the roughness of the channel. For purposes of this analysis, a very conservative energy grade line was assumed. The energy grade line was assumed to be a 3 ft. drop in 6500 ft. for a slope of .000462 ft/ft. This slope was determined by subtracting the bottom of the channel (417 ft.) from the 420 ft. contour, 6500 north of the channel. The effective hydraulic radius was taken as the depth of the water existing in the channel during the various time increments throughout the PMP. The channel roughness was assumed to be described by a Manning coefficient of 0.025. Flows through the channel were calculated using Mannings equation:

$$Q = \frac{1.49}{n} A R^{2/3} S^{1/2}$$

Where:

Q = Flow in CFS

n = Manning's coefficient

A = channel cross sectional area ft^2

R = Hydraulic radius ft.

S = Energy grade line slope ft./ft.

The calculations for rising flows near the restricting channel were performed using calculated pond volume and elevation for the previous time increment. For falling flows the calculations were performed in an interactive ~~manner to approximate the proper flow for that time~~

~~increment. The resulting hydrograph for the restricting~~ flow channel and the water surface elevations for the pond formed north of the channel for the PMF based on the PMP are shown in Figures 2.4-12 and 2.4-13 respectively.

The above analysis shows that the maximum pond elevation at the restricting channel location is 431.2 ft. Based on the assumed slope of 3/6500 ft/ft and the fact that WNP-2 is 1500 ft. north of the restricting channel, the water surface elevation would be approximately 431.9 ft. at the WNP-2 site.

As shown in Figures 2.4-9 and 2.4-10 the DOE mainline railroad and the four lane DOE highway both cross the drainage basin, but neither would affect the flood levels or channel discharge capacity adversely.

In summary, the natural channel east of the site has the capacity to store the entire PMP at a water surface elevation of 426 ft. In addition, at the most restricting channel the discharge capacity is sufficient that the water surface elevation would not exceed 431.9 ft. at WNP-2. The lowest safety related structure is the top of the spray pond walls at EL. 435 ft. Hence, the PMP would not flood any safety related structures.

2.4.3.4 Probable Maximum Flood Flow

The probable maximum flood flow for the Hanford region of the Columbia River noted in 2.4.3 is 1,440,000 cfs (regulated). The flood peak could occur between mid-May and mid-July (See Figure 2.4-5).

The U.S. Corps of Engineers[3] have developed a system diagram for flood estimates. All comparisons and consistency checks indicate the derivation of the PMF is technically sound and that the hydrographs that have been developed are in agreement with the definitions of those floods.[3]

2.4.3.5 Water Level Determinations

The Corps of Engineers prepared water surface profiles for ~~artificial and real stage flows for the Columbia River Mile~~

~~323 to River Mile 395 in November 1970. A portion of that~~ information is presented in Figure 2.4-8. The information was used to develop Flow Rating Curve for River Mile 350 and is shown in Figure 2.4-14.

2.4.3.6 Coincident Wind Wave Activity

The computation of wind wave activity at the site is based on the Corps of Engineers' publication No. ETL 1110-2-8, "Computation of Freeboard Allowances for Waves in Reservoirs".[8] This document provides the method for estimating the most critical effect of wave action.

For the Design Basis Flood which results from the postulated PMP event, an analysis was performed assuming a 430 ft. elevation pond level. In the analysis the effective fetch was computed with an 18° deviation from either side of a central radial. The effective fetch distance was then exaggerated from 1.2 to 1.5 miles to assure that the computed value for the maximum wave height was conservative.

The wind generated maximum wave height above still water level coincidental with the possible ponding (See 2.4.3.3), using standard techniques for computation is 1.0 ft. for an over water wind velocity of 45 mph. The runup of the wind generated waves is included in the previously defined shore~~line wave height and was calculated to be 0.6 feet~~.

2.4.4 POTENTIAL DAM FAILURES, SEISMICALLY INDUCED

Analyses of floods resulting from potential dam failures were investigated by the Corps of Engineers for the Columbia River. These studies are consistent with Regulatory Guide 1.59, Rev. 0 and its Appendix A. The flood resulting from the breaching of Grand Coulee Dam is considered in lieu of a seismically induced flood.

In 1951, the Seattle District Corps of Engineers made a confidential study (now declassified) to determine artificial flood hydrographs and the flood profile in the Columbia River Valley resulting from breaching the Grand Coulee Dam. The studies covered a spectrum of conditions in terms of breach openings and hydrologic conditions that might prevail at the time of an enemy attack. Although this criteria (enemy attack) does not apply to nuclear power plants, the "Artificial Flood No. 1" provides a "limiting case" assessment of the conservatism of WNP-2 elevation. This flood would have an

2.4-15 REFERENCES

2.4-1 "Municipal and Industrial Water Supply," Appendix XI of Columbia-North Pacific Region Comprehensive Framework Study of Water and Related Lands, Pacific Northwest River Basins Commission, P. 65.

2.4-2 Woods, V.W., "A Summary of Columbia River Hydrographic Information Pertinent to Hanford Works - 1894 to 1954," HW-30347, February 24, 1954.

2.4-3 Memorandum Report Columbia River Basin Lower Columbia River Standard Project Flood and Probable Maximum Flood, U.S. Army Engineers, North Pacific Division, Portland, Oregon, September, 1969.

2.4-4 "Probable Maximum Precipitation," Hydrometeorological Report No. 43, U.S. Weather Bureau (now NOAA), Northwest States, 1966.

2.4-5 Corps of Engineers, North Pacific Division, letter to V.C. St. Clair, Atomic Energy Commission, Richland, Office, 2 November, 1970.

2.4-6 Corps of Engineers, North Pacific Division, letter to M.J. Hroncich, Burns and Roe Inc., 7 February, 1972.

2.4-7 Corps of Engineers, North Pacific Division, letter to M.J. Hroncich, Burns and Roe Inc., 14 February 1972.

2.4-7a Final Report Hydrology Studies of the WNP-2 Site, by Battelle Northwest, Richland, Wa., to Burns and Roe Inc., July, 1971.

2.4-7b Addendum I to Final Report on Hydrology Studies of the WNP-2 Site, by Battelle Northwest, Richland, Wa., to Burns and Roe, Inc., July, 1971.

2.4-8 ~~"Computation of Freeboard Allowances for Waves in Reservoirs," Engineering Technical Letter No. 1110-2-8, Department of the Army, Corps of Engineers, August 1, 1966.~~

2.4-9 Artificial Flood Possibilities on the Columbia River Corps of Engineers, Washington District, Washington, D.C., 20 November, 1951.

"Wave Runup and Wind Setup on Reservoir Embankments" Engineering Technical Letter No. 1110-2-221, Department of the Army, Corps of Engineers, November 29, 1976.

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2.4-45 Parker, G.G., and Piper, A.M., "Geologic and Hydrologic Features of the Richland Area, Washington, Relevant to Disposal of Waste at the Hanford - Directed Operations of the Atomic Energy Commission," Interior Report 1, U.S.G.S. Report to Atomic Energy Commission, 101 pages, 5 Illus., 1949.

2.4-46 Newcomb, R.C., Strand, J.R. and Frank, F.J., "Geology and Ground Water Characteristics of the Hanford Reservation of the U.S. Atomic Energy Commission," Professional Paper # 717, U.S.G.S., Washington, 1972.

2.4-47 Kipp, K.L. and Mudd, R.D., "Selected Water Table Contour Maps for the Well Hydrographs and Hanford Reservation, 1944-1973," BNWL-1797, Battelle, Pacific Northwest Laboratories, Richland, Washington.

2.4-48 Means and Parcher, "Physical Properties of Soils," Charles E. Merrill Publishing Co., Columbus, Ohio, 1963, Pages 186 - 189.

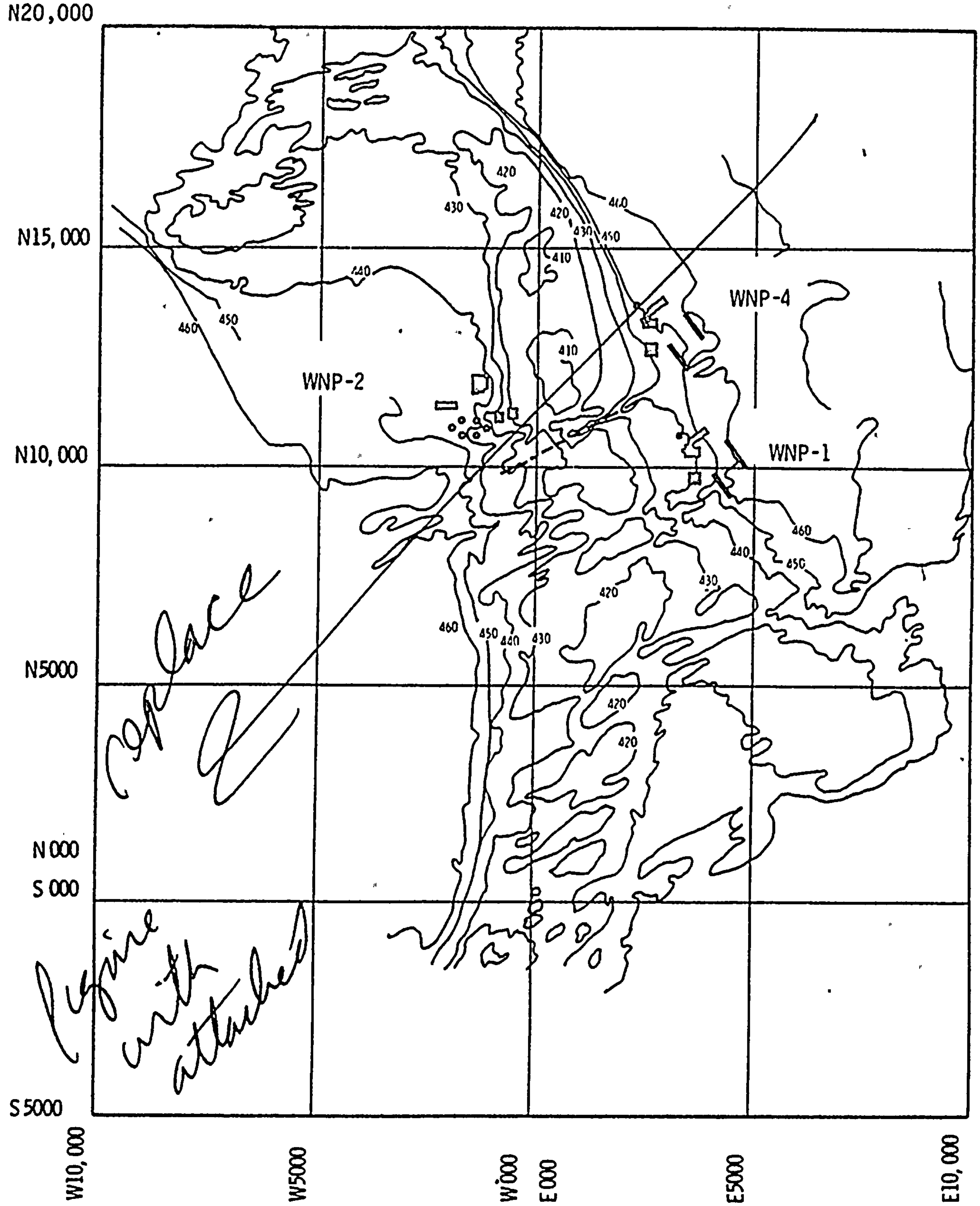
2.4-49 WNP-1 and 4 PSAR (Docket Nos. 50 - 460 and 50 - 513), Volume 1, section 2.4.

2.~~5~~4-50 Letter from A. Brandstetter to J.J. Verderber on "Columbia River Water Users", dated January 5, 1978.

2.4-51 Design of Small Dams, U.S. Bureau of Reclamation, 1977.

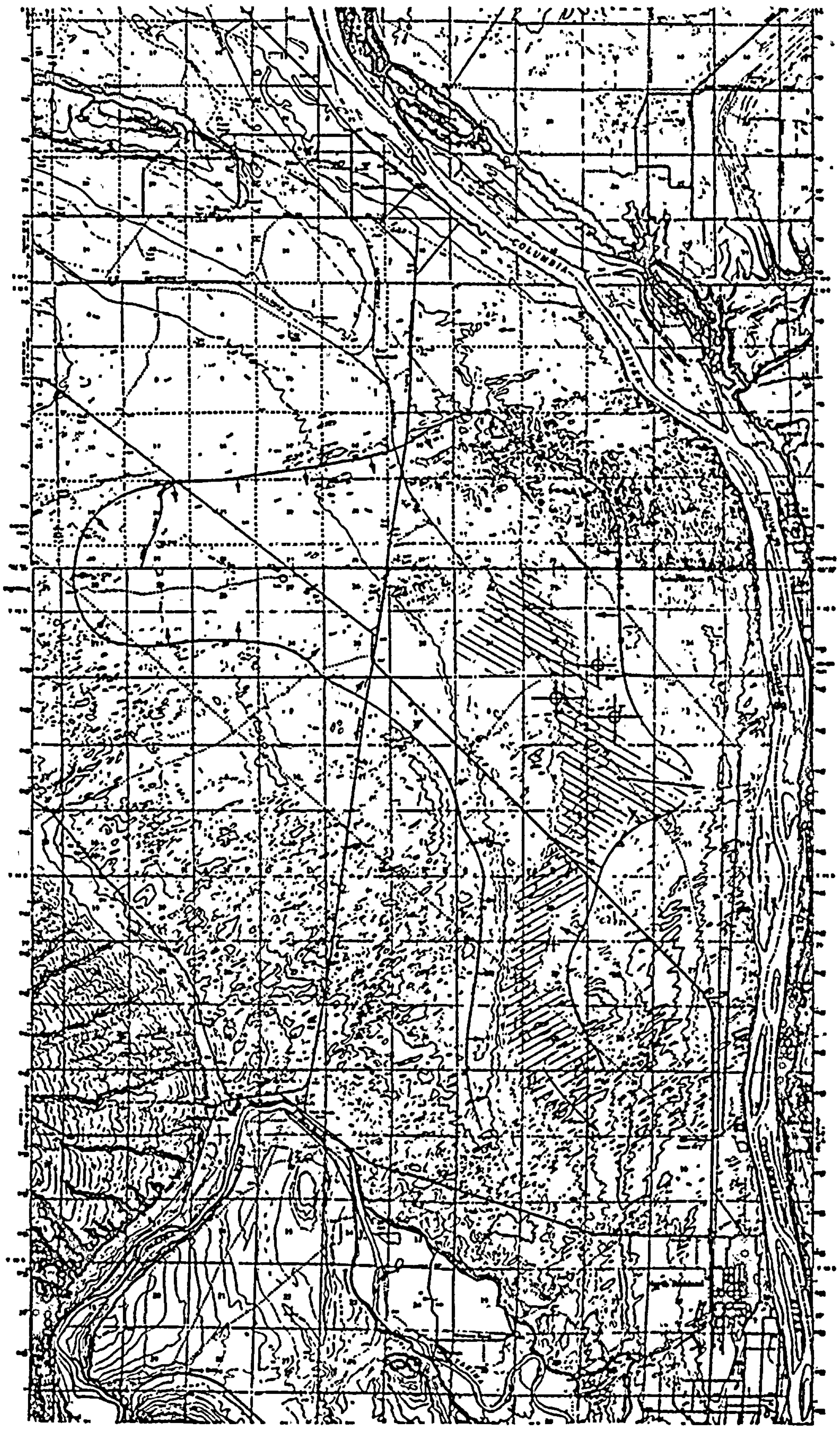
2.4-52 "Water Surface Profiles", Vol. 6 Hydrologic Engineering Methods for Water Resources Development, U.S. Army Corps of Engineers, Hydrologic Engineering Center, July 1975.

2.4-53 Shore Protection Manual, U.S. Army Corps of Engineers, Coastal Engineering Research Center, 1975.

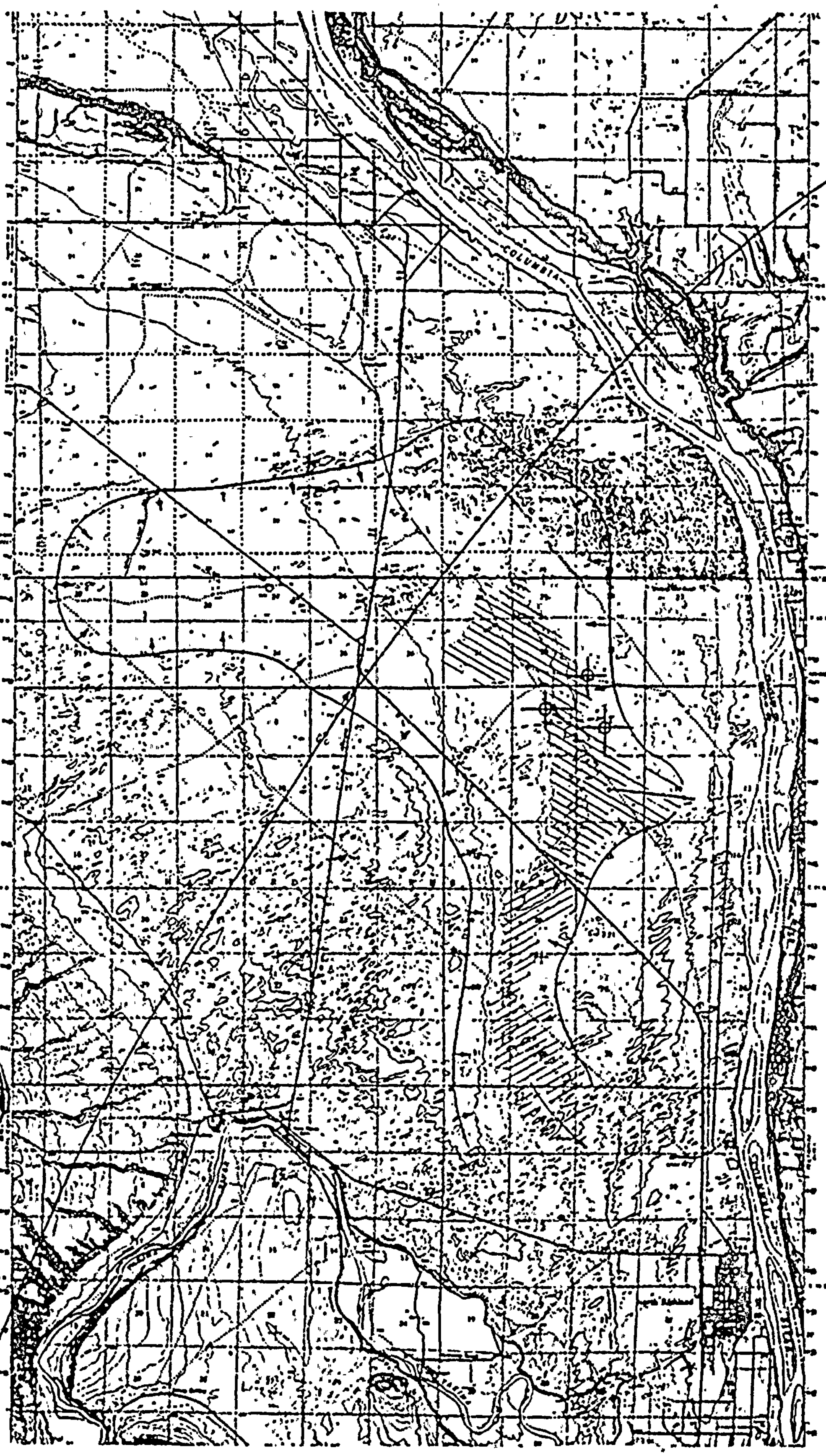


Replace figure with attached

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	DETAILED CONTOURS NEAR THE SITE	FIGURE 2.4-3

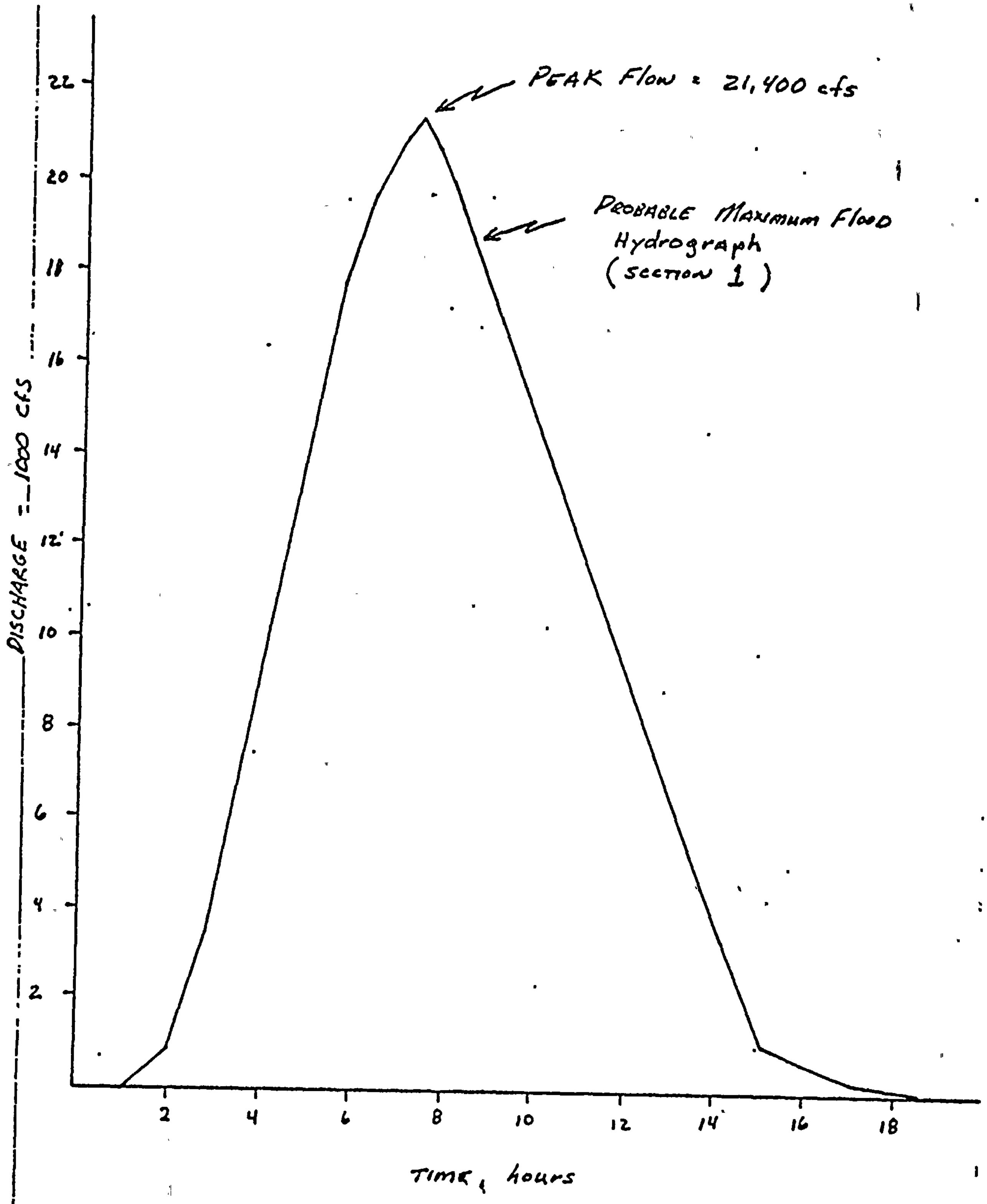


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROBABLE MAXIMUM PRECIPITATION DRAINAGE BASIN	FIGURE 2.4-~~8~~3

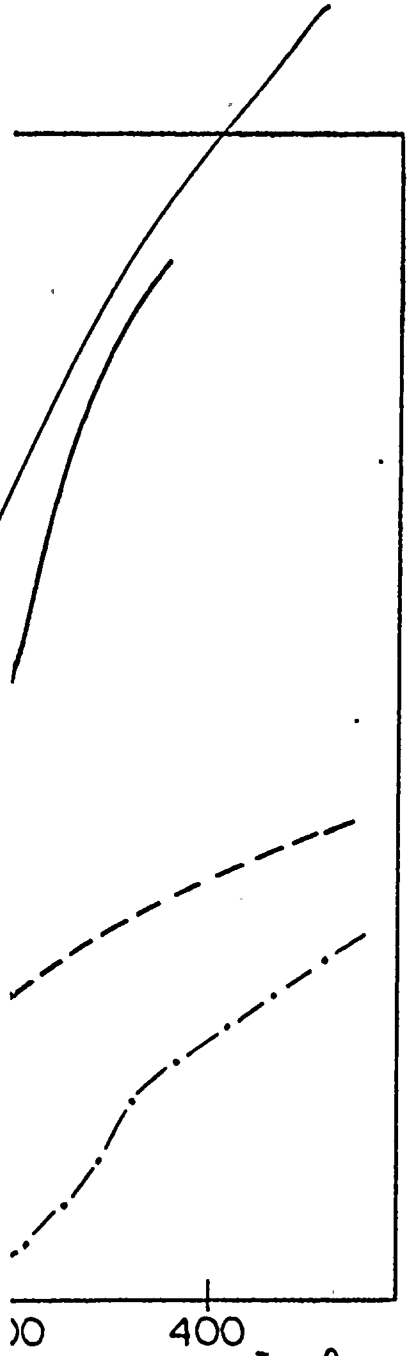


Replace figure with attached

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROBABLE MAXIMUM PRECIPITATION DRAINAGE BASIN	FIGURE 2.4-9



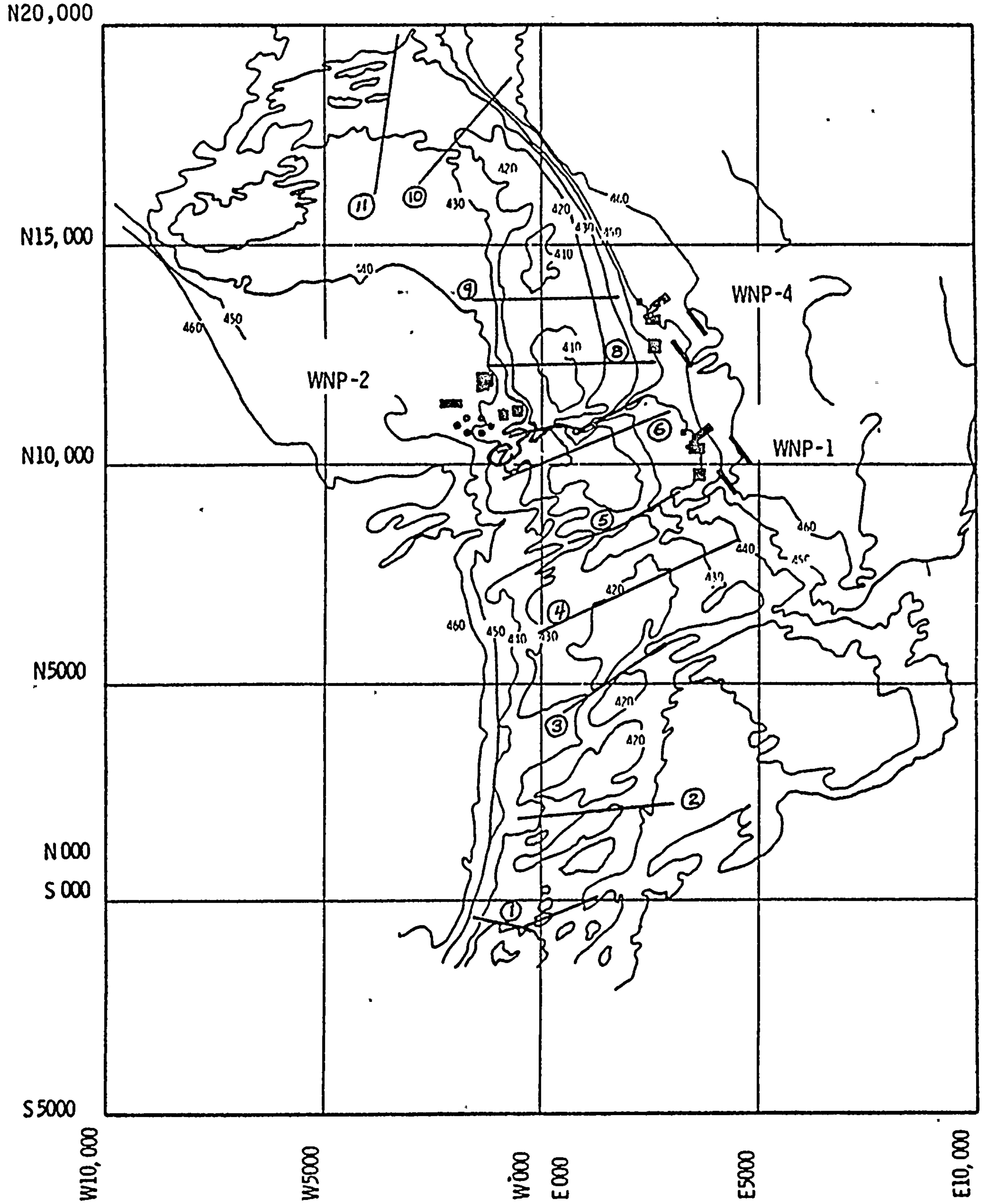
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PMF HYDROGRAPH DUE TO THUNDERSTORM PMP	FIGURE 2.4-~~12~~ 9



replace figure with attached

FLOW CHANNEL CROSS SECTIONS

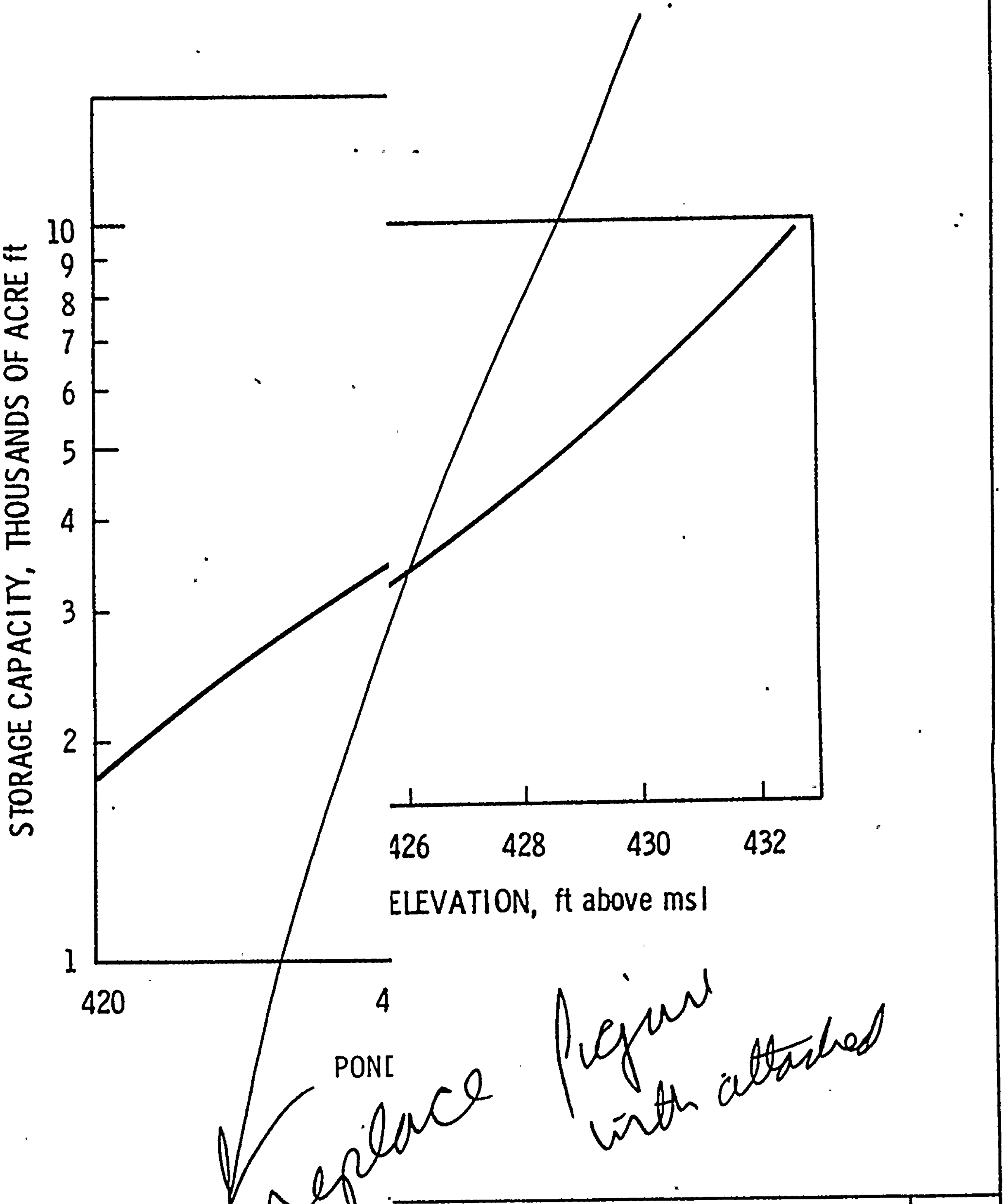
FIGURE 2.4-10



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

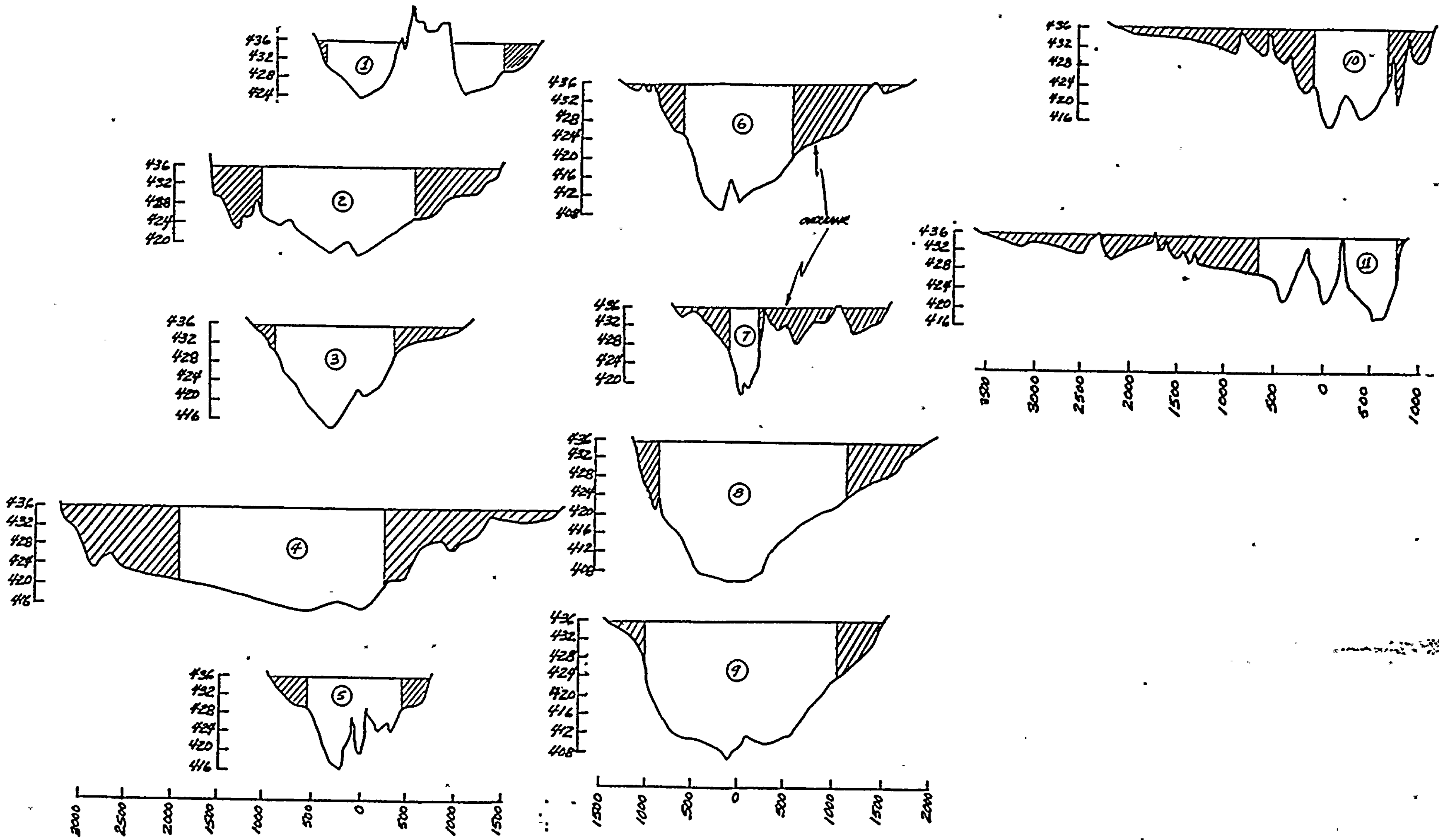
~~DETAILED CONTOURS NEAR THE SITE~~
CHANNEL CROSS-SECTION LOCATIONS

FIGURE
2.4-10

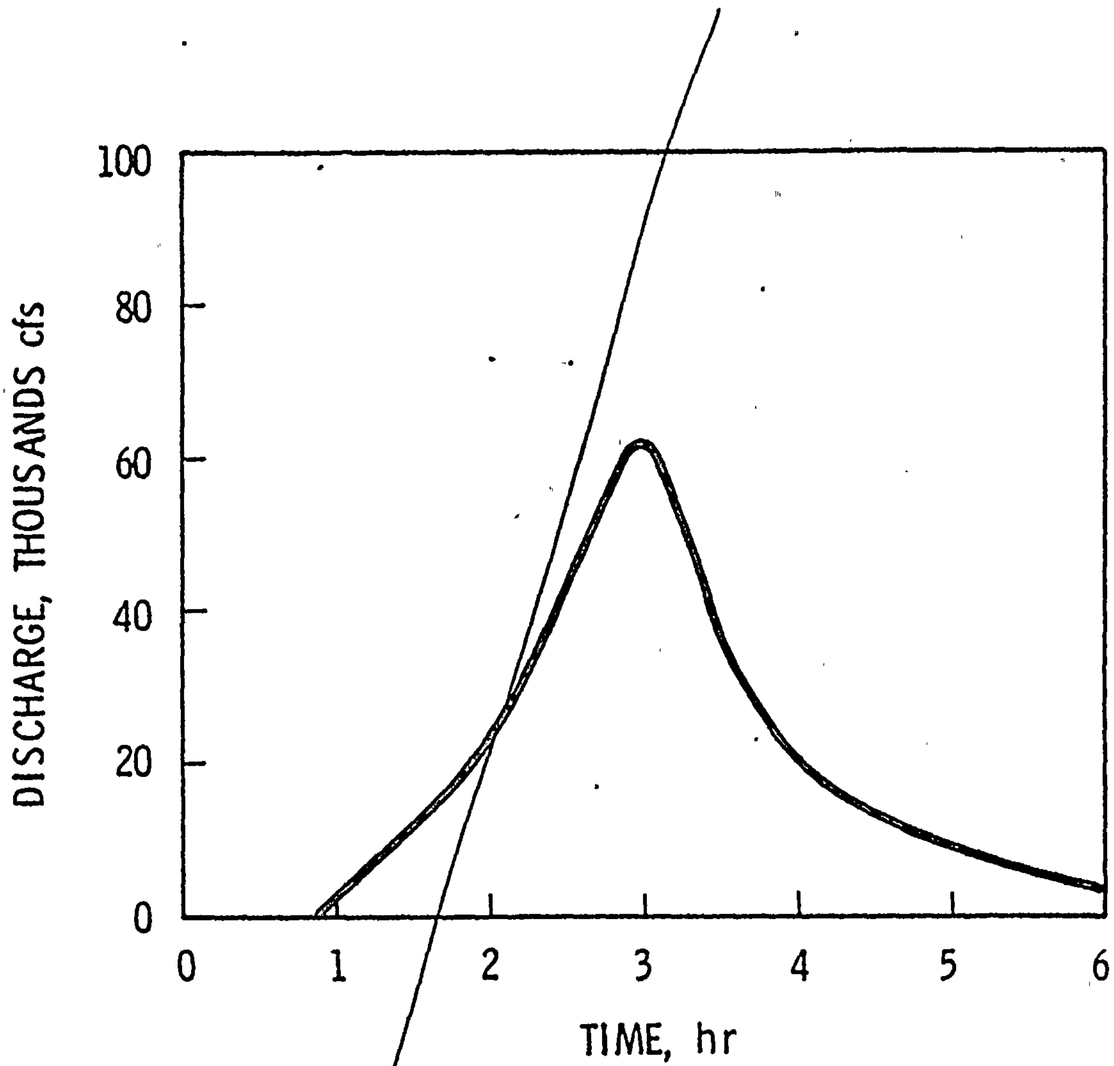


CHANNEL STORAGE CAPACITY AND CROSS SECTIONAL AREA VS POND SURFACE ELEVATION

FIGURE 2.4-11



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PMP CHANNEL CROSS SECTIONS	FIGURE 2.4-11



replace figure with attached

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PMF HYDROGRAPH DUE TO THUNDERSTORM PMP	FIGURE 2.4-12

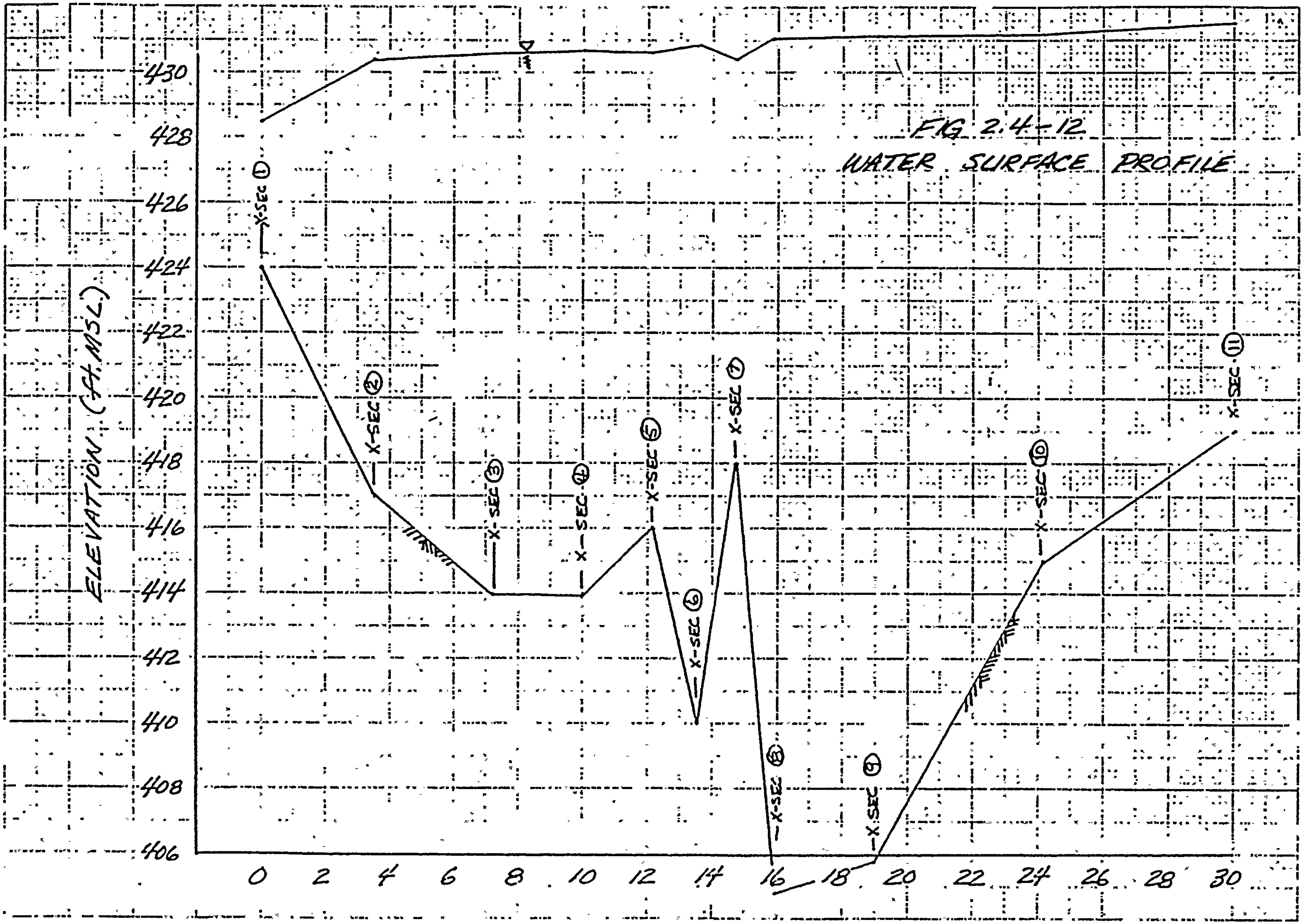
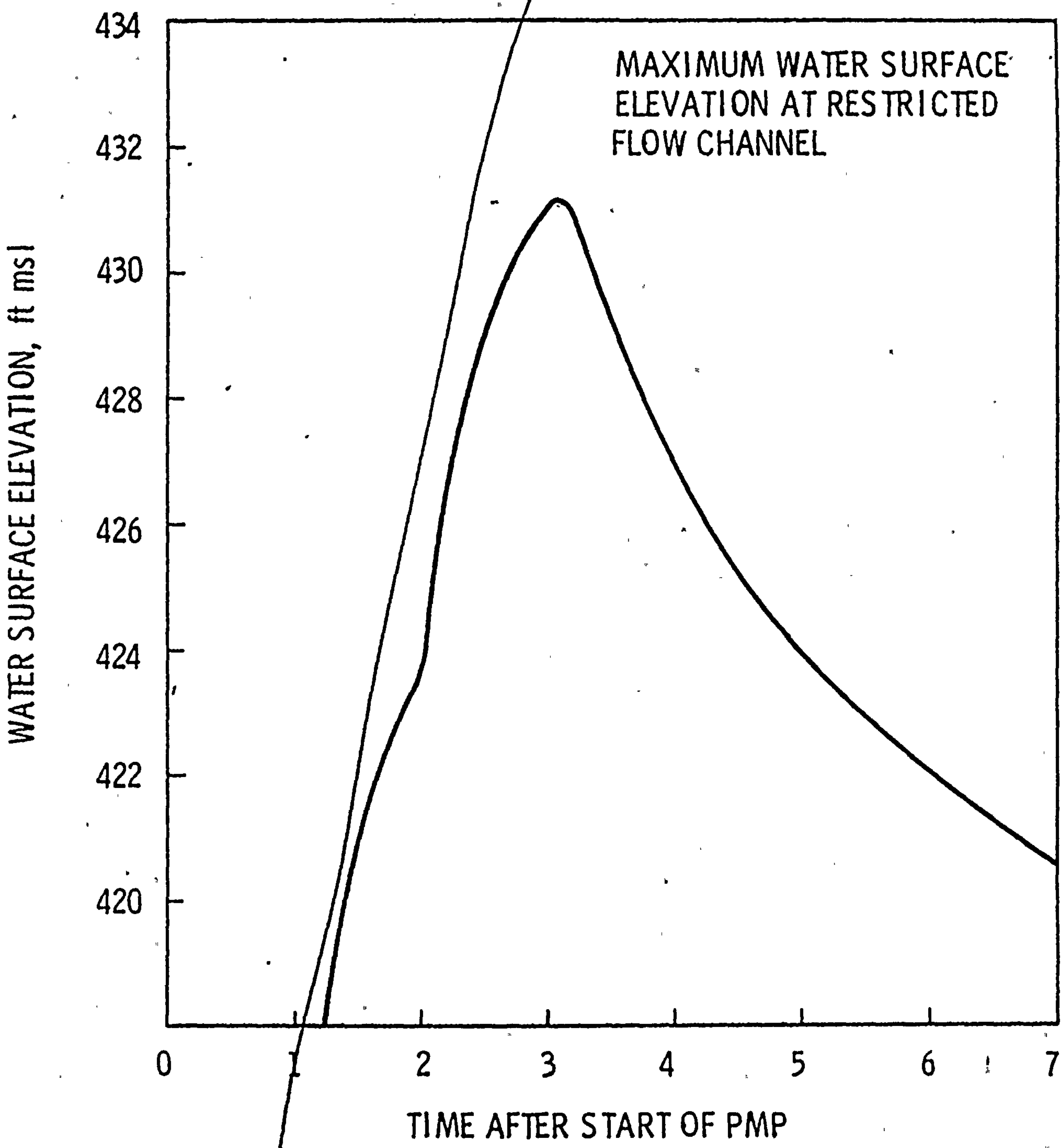


FIG 2.4-12
WATER SURFACE PROFILE
ELEVATION (ft. MSL)
430
428
426
424
422
420
418
416
414
412
410
408
406
0
2
4
6
8
10
12
14
16
18
20
22
24
26
28
30
X-SEC 1
X-SEC 2
X-SEC 3
X-SEC 4
X-SEC 5
X-SEC 6
X-SEC 7
X-SEC 8
X-SEC 9
X-SEC 10
X-SEC 11



replace figure with attached

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAXIMUM POND SURFACE ELEVATION VS. TIME AFTER START OF PMP	FIGURE 2.4-13

Q 371.10

Your analysis of the PMF water suface profiles for the 38.5-square-mile basin is not clear and uses some assumptions which may not be conservative. Specifically, slope-area computations are not necessarily conservative when computing water-surface profiles. Since the design basis flood (i.e., within three feet of safety-related structures) is produced by a PMF on this stream, provide water surface profiles computed using a method which utilizes standard step computational solutions to gradually varied flow (HEC-2, Corps of Engineers is one acceptable method). More than three cross-sections (in addition to those on Figure 2.4-10) will be needed for this revised analysis.

Response:

The PMF hydrograph and water surface elevations were recalculated using more conservative assumptions. Refer to Section 2.4.3.3 revised.*

*See the draft FSAR changes attached to question 371.9.

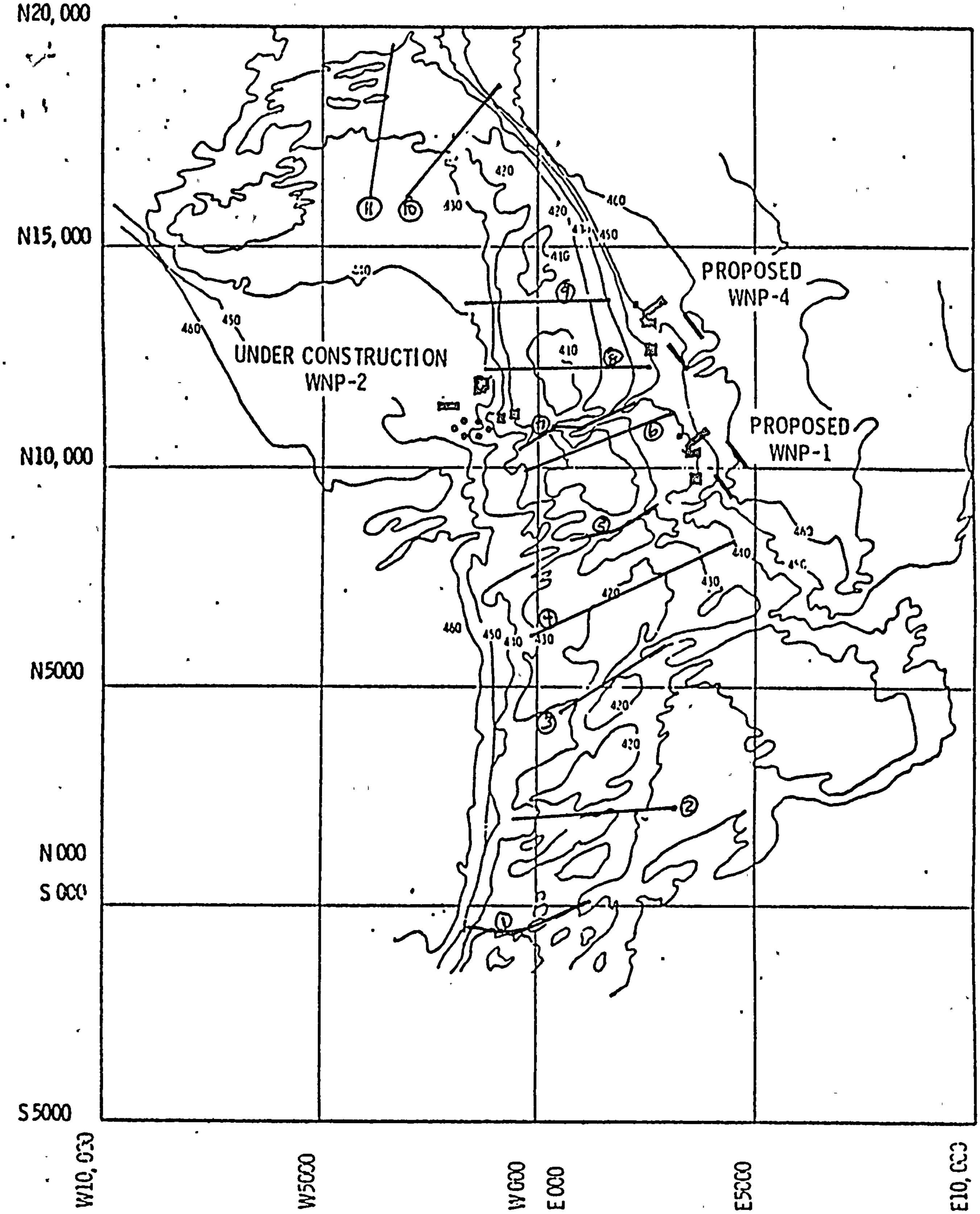


FIGURE 371.10-1

Q 371.10

Your analysis of the PMF water suface profiles for the 38.5-square-mile basin is not clear and uses some assumptions which may not be conservative. Specifically, slope-area computations are not necessarily conservative when computing water-surface profiles. Since the design basis flood (i.e., within three feet of safety-related structures) is produced by a PMF on this stream, provide water surface profiles computed using a method which utilizes standard step computational solutions to gradually varied flow (HEC-2, Corps of Engineers is one acceptable method). More than three cross-sections (in addition to those on Figure 2.4-10) will be needed for this revised analysis.

Response:

The PMF hydrograph and water surface elevations were recalculated using more conservative assumptions. Refer to Section 2.4.3.3 revised.*

*See the draft FSAR changes attached to question 371.9.

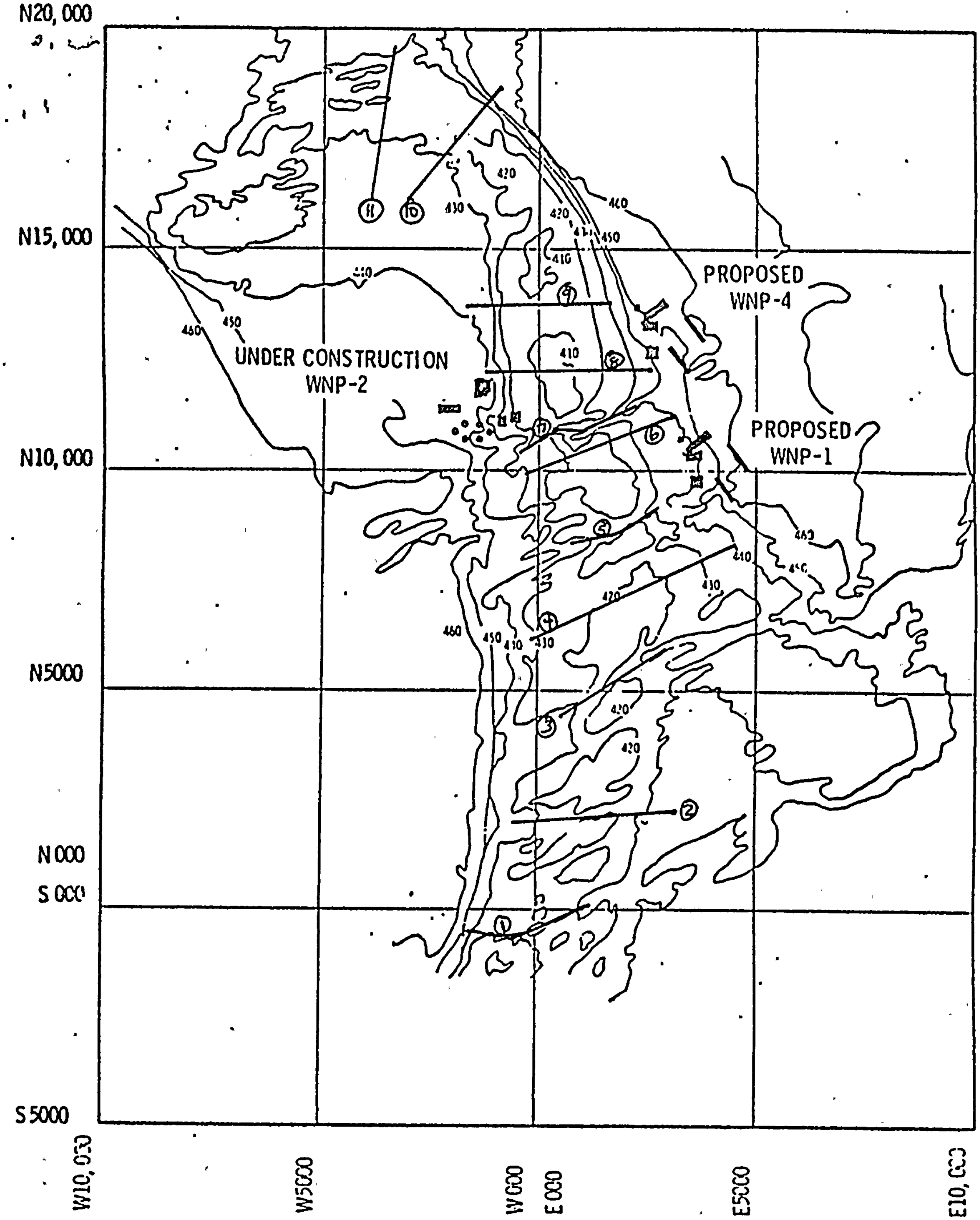


FIGURE 371.10-1

371.11 We require that you use an 'n' value of 0.05 when calculating the water surface elevations of the stream cited in Item 371.9. Additionally, we require that you perform sensitivity studies to determine the change in the water surface profile with respect to a change in the 'n' value assumed in the analysis. Alternatively, you can provide justification for the appropriate conservatism of a Manning 'n' value of 0.025.

Response: The PMP hydrograph and water surface elevations were recalculated using conservative assumptions. Refer to Section 2.4.3.3 revised.*

A Manning's roughness coefficient, n, of 0.05 was used for the overbank areas of each cross-section. Values of n = 0.035 were used for all main channel sections. These values are deemed conservative based upon the following references:

1) U.S.G.S. Water Supply Paper No. 1849, "Roughness Characteristics of Natural Channels," 1967, pp. 10-13, 30-33.

2) U.S.B.R., Designs of Small Dams, Table B-6, Revised Reprint of Second Edition, 1977, p. 577.

3) U.S. Corps of Engineers, Hydrologic Engineering Center, "Water Surface Profiles," Volume 6, Hydrologic Engineering Methods for Water Resources Development, Table 4.01, July 1975, p. 4.08.

4) Chow, V.T. 1959. Open-Channel Hydraulics, Table 5-6 (pp. 110-113 and pp. 115-123).

* See the draft FSAR changes attached to question 371.9.

371.12 If credit is to be taken for reservoir and stream storage in your calculations of the PMF profile, as discussed on Pages 2.4-11 and 2.4-12 of the FSAR, we require that you provide the bases for the flow routings including all applicable routing coefficients, inflow/outflow relationships, and area capacity curves. Alternatively, you may assume that the computed peak flow occurs without any attenuation attributable to the storage capacity of streams and reservoirs.

Response In the revised analyses discussed in response to Question 371.10, credit is not taken for reservoir and stream channel storage in the calculations for the PMF water surface profile. Therefore, routing coefficients, inflow/outflow relationships, and area-capacity curves are not applicable. We have assumed that the peak flow occurs without any attenuation attributable to the storage capacity of streams and reservoirs.

Q 371.13

Since it is not apparent to the staff how a maximum wave height of only one foot was computed using an effective fetch of 1.5 miles and an overwater wind speed of 45 miles per hour as indicated on Page 2.4-13 of the FSAR, provide additional information on this matter. In addition, provide the effective fetch diagram and the depth of water assumed in your analysis.

Response:

The effective fetch and maximum wave height have been recomputed. Refer to Section 2.4.3.6 revised.*

*See the draft FSAR changes attached to question 371.9.

Q 371.14

Provide the bases for your estimate of a wave runup of 0.6 feet. Indicate the slope assumed in your analysis and the material composition of the slope (e.g., either riprap or a concrete wall)

Response:

Wave setup and runup have been reevaluated. Refer to Section 2.4.3.6 revised.*

*See the draft FSAR changes attached to question 371.9.

372.8 (2.3.1) In developing your analysis of the potential for dust storms, you examined 28 years of Hanford data and estimated dust loadings using an empirical relationship correlating dust loadings with visibility. Your purpose was to establish the appropriate climatology for dust storms at the WNP-2 site and from that to develop the design bases for dust loadings potentially affecting the operational functioning of the WNP-2 facility. However, the empirical relationship which you used is based on data from the Great Plains. Accordingly, we request that you:

a. Provide justification which demonstrates that this relationship is applicable to the WNP-2 site.

Response: The relationship between the mass loading, M, and visibility, V is a theoretically derived relationship that should apply to any situation:

$$MV = C$$

In application to specific cases, the value of the constant C will depend on the suspended particle size distributions. Patterson and Gillette (1977) point out that if the suspended particle size distribution changes in a consistent way with increasing or decreasing visibility, the exponential constant included by Hagen and Woodruff (1973) will allow for an empirical fit to shifts between visibility and particle size distributions. This relationship with an exponential constant γ is,

$$MV^{\gamma} = C$$

Even with this exponential correction for shifts in particle size distributions, the constant C still has a relatively wide range of reported values. The variation is attributed to changes in local surface conditions.

Patterson and Gillette (1977) reviewed the constants for a number of sites and conditions. The largest values of dust loadings for a given visibility were in the Great Plains data (Hagen and Woodruff, 1973) which were obtained under drought conditions with local erosion over agricultural lands. The variability in the Great Plains data suggests that the annual value of C may range up to a factor of two larger than the best fit value in * ~~the table~~. Data from other studies without local wind erosion have consistently smaller relative mass loadings. The table below summarizes these values. Larger values of the constant, C, implies larger dust loadings for a given wind speed and visibility.

Table 372.8-1

576-

TABLE ~~3~~. Best Fit Values of Constants in Visibility-Dust Relationship

	C $kg\ m^{-3} km$	γ
Charlson, 1969; Urban	1.8	1.00
Bertrand, et al., 1974; 2×10^3 km downwind from source region	1.4	1.05
Patterson and Gillette, 1977; rural Texas, no local erosion	20	1.07
Chepil & Woodruff, 1957; drought with local erosion from agricultural lands	56	1.25

The applicability of any of these to the WNP-2 site depends on the similarity of the surface conditions. Patterson and Gillette (1977) point out the large potential variation from year to year at a single site as surface conditions change. Hence, no one set of constants will apply to a site for all conditions. The surface conditions in the region surrounding the WNP-2 at any given time depends largely on the preceding weather although agricultural activities and fire are also major factors. The range of values in Table 1 may variously apply depending on conditions upwind of the WNP-2 site. However this is a region where local wind erosion is not unusual and the Great Plains data appear to be a suitably conservative basis for computing the potential dust loading levels at WNP-2. Only during worst conditions are the surface conditions in the WNP-2 region expected to approach the unstabilized agricultural surfaces for which the Great Plains constants were derived . Hence depending on source surface conditions, the actual dust loading values at the WNP-2 site are expected to range up to the computed values.

References

Bertrand, J., J. Baudet and A. Drochon, "Importance des aerosols naturels en Afrique de L'ouest," J. Recherches Atmos., 8, pp. 845-860, 1974.

Charlson, R. J., "Atmospheric Visibility Related to Aerosol Mass Concentration," Environ. Sci. Technol., 3, pp. 913-918, 1969.

Chepil, W. S. and N. P. Woodruff, "Sedimentary Characteristics of Dust Storms - II. Visibility and Dust Concentration," Am. J. Sci., 255, pp. 104-114, 1957.

Patterson, E. M. and D. A. Gillette, "Measurements of Visibility vs Mass-Concentration for Airborne Soil Particles," Atmos. Envir., Vol. 11, pp. 193-196, Pergamon Press, 1977.

Hagen, L. J. and N. P. Woodruff, "Air Pollution from Duststorms in the Great Plains," Atmospheric Environment, 7, pp. 323-332, 1973.

Patterson, E. M., D. A. Gillette and G. W. Grams, "On the Relation Between Visibility and the Size-Number Distribution for Airborne Soil Particles," J. Appl. Met., 15, pp. 470-479, 1976.

b. Provide estimates of the dust concentrations for the dust storms of April 1972 and August 1955 using the Great Plains empirical relationship so that we may compare the measured values with predicted values.

Response: The case studies of mass loading in this region have limited utility in the derivation of a representative visibility-mass loading relationship. There are only a few data points and the field data do not include concurrent visibility observations. For the August 1955 dust storm, the adjacent weather observation records at the Hanford Meteorological Station may be used to estimate the appropriate visibility value corresponding to the field tests. Using a reference height of 1 m, the predicted value is 7 mg m^{-3} compared to an interpolated measured value of 17 mg m^{-3}. This is as good a agreement as can be expected considering the sources of variability in this single point measurement. Estimates cannot be made for April 1972 due to the lack of definition of the exact time period for the test.

Although direct comparisons between computed and measured dust loading values in this region are limited by the lack of concurrent visibility observations, the magnitude of the observed dust loadings in Figure 2.3-6 are in general comparable to dust loading values computed by the subject visibility-mass loading relationship.

c. Provide the dates and the peak and average wind speeds at the 50 foot level of the WNP-2 tower for the six worst storms listed on Page 2.3-18 of the FSAR. Provide cross-references to the appropriate sections of the FSAR for your statement that "the worst case dust storm ... was used in other FSAR chapters for ... plant design and operation parameters."

Response: The dates and hourly peak and average wind speeds for the six worst dust storms listed in the FSAR are given in the following table. Wind data is given for the 50' level on the Hanford Meteorological Tower since all of these events precede the onsite meteorological data collection at the WNP-2 site.

Dust Storm Number (FSAR p 2.3-18)	Date	Average Wind Speed m/s	Peak Hourly Wind Speed m/s
1	6/05/57	26.8	26.8*
2	4/12/69	17.9	17.9*
3	4/11/56	8.0	13.0
4	4/10/65	14.3	17.0
5	4/30/68	15.2	16.1
6	1/20/62	7.6	10.7

*Same as average for one hour duration dust storm.

See ~~attached~~ revised ~~draft~~ FSAR page 2.3-18 for changes to respond to second part of question.* In addition, see the response to question 40.26.** Also, more explicit responses to this question will be given in the response to ASB question 10.16 due to be submitted by April 9B. Appropriate additional FSAR changes will accompany that response.

d. Provide a copy of Reference 2.3-33.

Response: See attached report.
Droppo, J. G., 1978: Hanford Dust Storm Climatology, to Burns and Roe, Inc. for Washington Public Power Supply System, Battelle, Pacific Northwest Laboratories, Richland, Wa., January 1978.

This report expanded on reference 2.3-33 and represents the final report of the investigation.

* See the attached draft change.

** copy submitted with this set.

The April 1972 dust storm has higher mass loadings near the surface in all size ranges. The August 11, 1955 dust storm had higher dust loadings above 1 to 2 m heights in the ranges greater than 5μ diameter. One interpretation of these profiles is that the 1972 storm had a source nearby and the 1955 data represents advection of airborne dust from more remote sources [33].

2.3.1.2.1.5.2 Hanford Dust Storm Climatology for Design and Operating Bases

The Hanford climatological study of dust storms for 1953-1970[30] (discussed in the previous subsection) was re-examined for the purpose of establishing the "worst case" dust storm which may have occurred during that period [33]. The worst case dust storm, i.e., that storm which had the largest calculated time integrated dust loading (mg-hr/m^3), ~~was used in other FSAR chapters for the purpose of ascertaining the effect of that storm on WNP-2 plant design and operation parameters~~ is considered in FSAR Sections 9.4 and 9.5 in evaluating the design and performance of HVAC systems and diesel generators. Results of this worst case dust storm investigation are listed below. As discussed in the previous subsection, these loadings would apply for a height of 5-6 feet above the ground.

<u>Detailed Estimates of the Dust Loadings for the Six Worst Storms Based on Surface Observations of the Hanford Meteorology Station, 1953-1970</u>

Storm Number	Total Dust Loading (mg-hr/m^3)	Actual Duration (hrs)	Average Dust Loading (mg/m^3)
1	40*	0.67	60
2	100	1.0	100
3	160	18**	8.9
4	44	2.6	17
5	90	3.1	29
6	80	7	11

*Value is less than actual dust loading as a result of less than 1 hour duration.

**The detailed investigation yielded 18 hours as opposed to a range of 1-16 hours given in Table 2.3-40 of 2.3.1.2.1.5.1 for the range in duration of dust storms using the same 1953-1970 data.

DW Dragnich
JG Droppo
RK Hadlock
AB/Corr. File
AB/Task 5.4 File

Battelle
Pacific Northwest Laboratories
Battelle Boulevard
Richland, Washington 99352
Telephone (509) 946-2412
Telex 32-6345

19 January 1978

Mr. J. J. Verderber
Project Engineering Manager
Burns and Roe, Inc.
185 Crossways Park Drive
Woodbury, NY 11797

Attention: Mr. J. M. Raymont

SUBJECT: HANFORD SANDSTORMS
ENVIRONMENTAL STUDIES FOR WNP-2

Dear Mr. Verderber:

Enclosed are revised copies of the report "Hanford Dust Storm Climatology" by Dr. James G. Droppo of our Atmospheric Sciences Department. Please call Dr. Droppo on (509) 946-2166 if you need additional information.

Very truly yours,

Albin Brandstetter, Ph.D
Senior Development Engineer
Environmental Management Section
Water and Land Resources Department

AB:pl/19

In triplicate

Enclosure

cc: R. A. Chitwood, WPPSS
R. K. Woodruff, WPPSS

HANFORD DUST STORM CLIMATOLOGY

to
Burns and Roe, Inc.
for
Washington Public Power Supply System

by
James G. Droppo, Ph.D

January 1978

Battelle
Pacific Northwest Laboratories
Richland, Washington 99352

HANFORD DUST STORM CLIMATOLOGY

A climatological summary of Hanford dust storms is given in Table 1 for 1953-1970 (Orgill, et al., 1974). Dust dependence on wind speed and direction (50 ft) at the Hanford Meteorology Station is given in Table 2 for the same period (Orgill, et.al., 1974). Approximate values of dust concentrations are computed based on an empirical relationship using visibility observations (Hagen and Woodruff, 1973). The relationship is

$$C_6 = \frac{56}{V^{1.25}} \, [\mathrm{mg/m^3}] \qquad (1)$$

where V is horizontal visibility in km. This is based on data from the Great Plains with visibilities 7 to 9 miles and wind speeds greater than 12 mph. Using hourly weather observations at the Hanford Meteorological Station as input Orgill, et al., used the following criteria to define a wind resuspension or dust storm period:

1. Visibility less than 7 miles and dust, blowing dust or sand reported.
2. Visibility 7 to 14 miles, wind speed greater than 5.8 ms^{-1} (13 mph) and relative humidity less than 70%. Dust is assumed.

TABLE 1. Frequency of Wind Resuspension Periods by Hanford (1953-1970)

Total Dust Hours	476
Total Dust Days	142
Number of Dust Storms	150
Average Dust hr/yr	26.4
Average Dust days/yr	7.9
Average Dust Storms per year	8.3
Range in Duration of Dust Storms (hr)	1-16
Average Duration of Dust Storms (hr)	3.2

To assure that all relevant observations were included, certain additional criteria were established to include beginnings and endings of storms.

TABLE 2. Dust Concentration Dependency o[illegible]nd Speed and Direction at Hanford, 1953-19[illegible]
(Orgill, et al., 1974)

Wind Direction	Predicted Concentration From Visibility, mg/m^3											
	Wind Speed Class (MPH)											Overall Average
	1-3	4-7	8-12	13-18	19-24	25-31	32-38	39-46	47-54	55-63	64-Up	
SE	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
SSE	0.00	0.00	0.00	2.71	1.38	1.25	0.00	0.00	0.00	0.00	0.00	1.78
S	0.00	0.00	7.83	1.62	1.38	5.70	15.92	0.00	0.00	0.00	0.00	7.17
SSW	0.00	0.00	0.00	2.48	1.62	2.21	3.86	4.13	0.00	0.00	0.00	2.95
SW	0.00	0.00	2.71	6.34	3.54	2.75	8.83	13.87	0.00	988.88*	0.00	19.40
WSW	0.00	0.00	1.74	1.81	4.96	4.13	12.95	48.31	0.00	0.00	0.00	7.67
W	0.00	1.74	1.74	1.83	2.89	5.37	2.71	0.00	0.00	0.00	0.00	3.54
WNW	0.00	0.00	3.49	2.64	1.77	1.99	3.29	4.13	0.00	0.00	0.00	2.39
NW	0.00	3.29	1.88	2.58	1.50	1.98	2.23	0.00	0.00	0.00	0.00	2.08
NNW	0.00	2.02	2.60	2.58	4.80	0.00	0.00	0.00	0.00	0.00	0.00	2.77
N	0.00	3.29	2.92	3.50	5.06	12.99	0.00	0.00	0.00	0.00	0.00	3.81
NNE	0.00	1.74	3.38	3.41	6.08	7.04	7.83	0.00	0.00	0.00	0.00	4.77
NE	0.00	4.38	4.60	3.33	4.54	2.81	0.00	0.00	0.00	0.00	0.00	3.84
ENE	0.00	0.00	0.00	3.05	2.19	0.00	0.00	0.00	0.00	0.00	0.00	2.43
E	0.00	0.00	3.29	2.44	3.60	2.71	0.00	0.00	0.00	0.00	0.00	2.73
ESE	0.00	2.71	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	2.71
Overall Average	0.00	2.84	3.00	3.15	4.15	3.71	8.57	22.22	0.00	988.88	0.00	6.77

* Visibility 0 to 1/16 Mile Due to One-hour Duststorm
0.00 No data

The frequency of the hourly data satisfying the dust storm criteria at Hanford is given in Table 3. The average visibilities are given on the last line of the table.

Care must be taken in interpretation of Tables 1, 2 and 3 to allow for certain limitations. Estimates based on visibilities and/or wind speeds outside the range using in formulation of Equation 1 will be an interpolation of data points. Table 3 indicates that the average visibilities are within the range excepting the high wind cases. This method assumes a fixed relationship between visibility and dust loading, ignoring other factors which can affect visibility. This can result in considerable variability on a case to case basis. Finally, the visibility observations are taken at specific times and are not hourly averages. Considering these limitations Tables 1, 2 and 3 may be taken to represent overall aspects of the Hanford dust storm climatology, but individual values must be considered approximate estimates - particularly those based on only a few data points.

The dust storm episodes were evaluated to minimize some of these limitations. This resulted in some changes in the climatological statistics quoted above. These changes resulted primarily from definition of the exact starting and stopping times. Although the number of dust storms were reduced by fourteen storms, the total number of dust hours increased from 476 to 479. The longest storm changed from 16 to 18 hours duration and the shortest storm still had one hour duration.

The six worst storms in this revised summary from the viewpoint of time integrated dust loading levels are given in Table 4.

Care must be still taken in interpolation of individual values derived from this method. Table 5 contains the result of a recomputation of dust loadings based on the surface data observations over the actual duration of the storms. Although, these are still subject to variabilities from the lack of continuous visibility data, these numbers are better estimates of the dust loadings than the values based on single hourly observations. A significant change occurs in the first two cases which illustrate the problem of direct application of the hourly climatology for individual cases.

TABLE 3. Hours Satisfying Dust Storm Criteria at Hanford 1953-1970 (Orgill, et al., 1974)

Wind Direction	Hours with (1) Visibility <7 Mile and Dust Reported or (2) Visibility 7 to 14 Miles, Windspeed >5.8 MW Sec; RHK 70% Dust Assumed — Wind Speed Class (MPH)											Total Hours
	1-3	4-7	8-12	13-18	19-24	25-31	32-38	39-46	47-54	55-63	64-Up	
SE	0	0	0	0	0	0	0	0	0	0	0	0
SSE	0	0	0	1	1	1	0	0	0	0	0	3
S	0	0	1	3	1	3	3	0	0	0	0	11
SSW	0	0	0	4	3	11	13	2	0	0	0	33
SW	0	0	1	3	13	24	26	6	0	1	0	74
WSW	0	0	1	7	17	39	13	4	0	0	0	81
W	0	1	1	3	5	7	1	0	0	0	0	18
WNW	0	0	5	5	11	6	1	1	0	0	0	29
NW	0	1	6	4	3	5	2	0	0	0	0	21
NNW	0	2	8	6	2	0	0	0	0	0	0	18
N	0	1	12	34	10	1	0	0	0	0	0	58
NNE	0	1	3	31	23	7	1	0	0	0	0	66
NE	0	2	3	19	15	5	0	0	0	0	0	48
ENE	0	0	0	3	6	0	0	0	0	0	0	9
E	0	0	1	6	2	1	0	0	0	0	0	10
ESE	0	1	0	0	0	0	0	0	0	0	0	1
Total Hours	0	9	42	129	112	110	60	13	0	1	0	476
Visibility, Miles	0.00	7.44	7.68	8.53	8.02	7.96	6.25	4.48	0.00	0.06	0.00	7.77

0.00 No Data

TABLE 4. Six Most Severe Dust Storms Based on a Time Integrated Dust Loading Criteria for Hourly Meteorological Observations at the Hanford Meteorology Station, 1953-1970

Storm Rank	Total Dust Load $mg\text{-}hr/m^3$	Average Dust Load mg/m^3	Storm Duration (hr)
1	990	990	1
2	175	175	1
3	160	8.8	18
4	115	38	3
5	90	13	3
6	80	11.4	7

Typically at the onset of a dust storm very low visibilities with high winds occur for a few minutes. The very limited visibility and high winds for such a period occurred for the first storm. The station log reveals that five minutes after this onset of the dust storm with near zero visibilities and winds 55 to 63 mph, the winds had dropped to 37 mph and visibility was 3/8 mile. This phenomenon was generated by a thunderstorm passing close to the station. The dust loading value was based on the first five minutes and represents an occurrence of very short duration which was part of the normal sequence of onset of a dust storm. Using a time weighted average, based on the visibility observations in the station log for the hour, gives an average dust loading of 60 mg/m^3 for the forty minutes duration of the dust storm. Such magnitudes can only be approximate since the empirical model is not validated for so low visibilities.

The second storm was associated with a cold front passage and had a duration of one hour. The initial visibility observation was 0.25 mile changing to 1 mile twenty-five minutes later. Thirty minutes after the second observation, the visibility was 12 miles indicating the dust storm was over. These visibilities give a weighted average of 100 mg/m^3 for this dust storm. The dust loadings for the third, fifth and sixth storms are not significantly changed.

TABLE 5. Detailed Estimates of the Dust Loadings for the Six Worst Storms Based on Surface Observations of the Hanford Meteorology Station, 1953-1970

Storm Number	Total Dust Loading mg-hr/m3	Actual Duration (hr)	Average Dust Loading mg/m3
1	40*	0.67	60
2	100	1.0	100
3	160	18	8.9
4	44	2.6	17
5	90	3.1	29
6	80	7	11

* Value is less than actual dust loading as a result of less than one hour duration.

The worst storm of these was actually storm #3. Storms #1 and #2 had overestimates of dust loadings as a result of using very low visibilities for longer periods than they actually occurred.

The reoccurrence time of dust storms for Hanford was summarized using the 136 dust storms identified in the reevaluation of climatological summary. Figure 1 contains the plot of the probability of time intervals between storms for periods up to thirty days. The log-normal relationship is nearly linear in this plot. One hour is the smallest duration that was allowed by the current definition of dust storms. This graph shows that once a given dust storm has stopped there is about a 5% probability that another one will occur within ten hours, and 50% probability that another one will occur within thirty days.

Table 6 contains a summary of the frequencies of occurrence of dust storm durations using the same 136 dust storms.

Sehmel (1976) and Hilst and Nickola (1959) report mass loadings as a function of particle size for two different dust storms. These data when considered with the climatological estimates of dust loading, provide an indication of the possible ranges of dust loadings at Hanford.

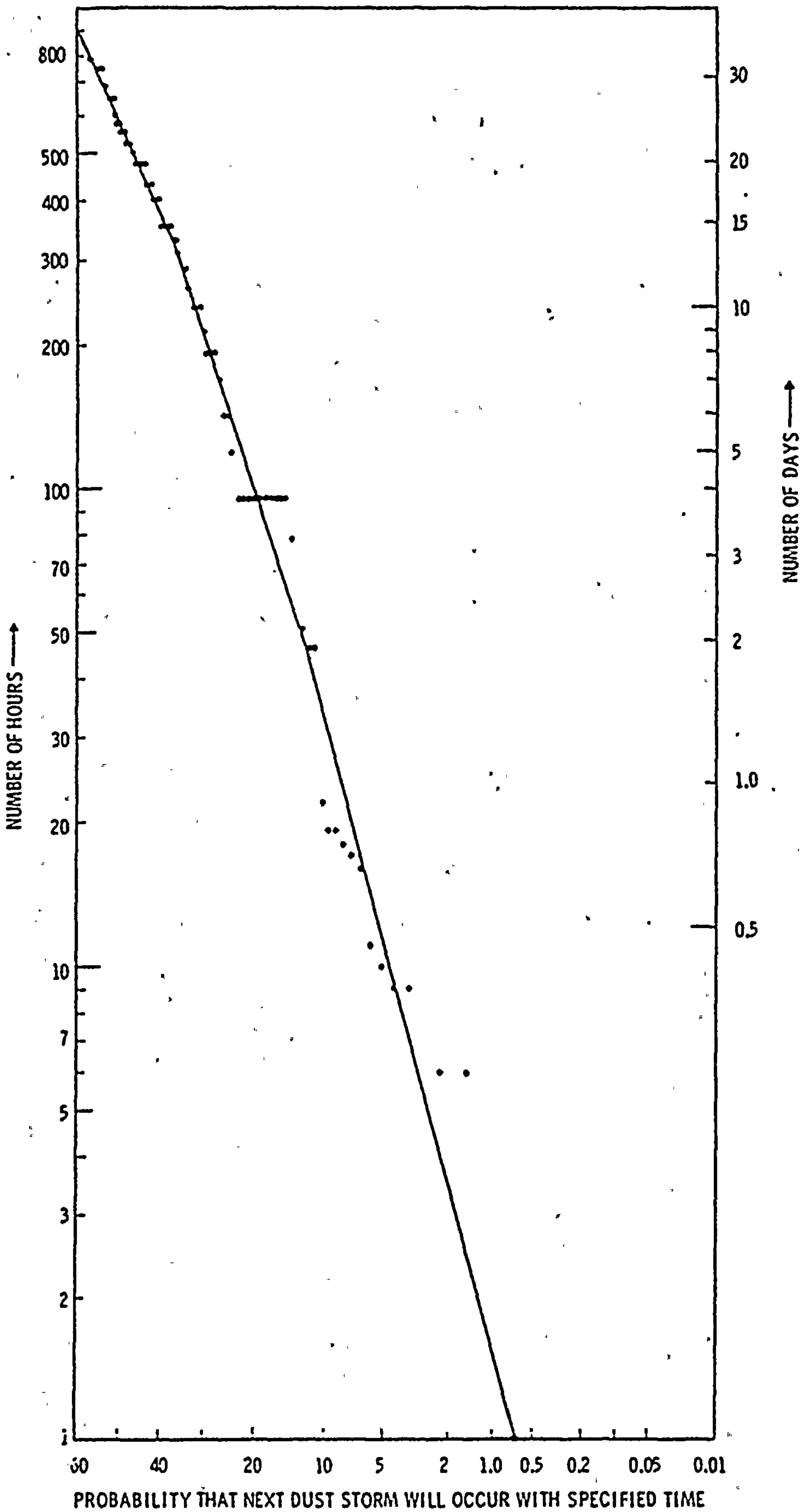


FIGURE 1. Reoccurrence Time Intervals for Hanford Dust Storms

TABLE 6. Frequency of Occurrence of Dust Storm Durations Based on the Revised Summary with 136 Dust Storms

Duration (hr)	No. of Cases	Duration (hr)	No. of Cases
1	50	10	1
2	23	11	2
3	19	12	1
4	14	13	2
5	5	14	2
6	5	15	1
7	5	16	1
8	3	17	0
9	1	18	1

Tables 7 and 8 contain profiles of airborne dust concentrations as a function of the larger particle diameters for both storms. A representative particle density of 2.0 g/cm^3 is used for both storms to produce comparable results. Optical measurements during the April 1972 storm at 0.9 m height for smaller particles (0.16 to 1μ diameter) gave mass loadings between 7.2×10^{-5} and 9.2×10^{-3} mg/m^3. It is clear that the mass loading is dominated by the particle diameter ranges given in Tables 7 and 8.

TABLE 7. Airborne Dust Loadings of Particles Greater Than 0.9μ for April 1972 Dust Storm (mg/m^3)

Particle Diameter Range (μm)	Height (m)					
	0.3	1.0	2.0	3.0	10.0	32.0
0.9 - 5.0	0.21	0.11	*	*	0.058	0.0038
5 - 20	0.83	0.28	0.26	0.25	0.070	0.0056
20 - 60	14.0	4.4	2.9	1.5	0.81	0.29
60 - 240	220.0	6.6	2.8	1.3	0.19	0.11
Total	235	11.4	∿6.0	∿3.1	1.13	0.41

* No value in reference.

TABLE 8. Airborne Dust Loadings of Particles Greater Than 0.9μ for August 11, 1955 Dust Storm (mg/m^3) Based on a Particle Density of 2.0 g/cm^3

Particle Diameter Range (μm)	Height (m) 0.38	2.0*	15.2	30.5	61.0	122
0.9 - 5.0	0.015	0.012	0.012	0.0079	0.0089	0.0052
5 - 20	0.39	0.32	0.23	0.17	0.17	0.123
20 - 60	3.1	2.2	1.1	0.77	0.61	0.36
60 - 240	18.0	12.0	2.3	0.78	0.26	0.52
Total	22.0	14.5	3.7	1.7	1.1	1.0

* Interpolated from data in paper.

The April 1972 dust storm has higher mass loadings near the surface in all size ranges. The August 11, 1955 dust storm had higher dust loadings over 1 to 2 m heights in the particle diameter ranges greater than 5 diameter. One interpretation of these profiles is that the 1972 storm has a source nearby and the 1955 storm represents advection of airborne dust from more remote sources.

The mass loadings at 2 m height for these two dust storms are comparable with the more severe mass loadings given in the climatological dust storm summary (Tables 2 and 3). Although having shorter durations, the mass loadings for the 1972 storm are comparable with the most severe dust storm as defined in Table 5.

It is important to understand that the climatology of dust storms depends both on weather and on the configuration of dust and sand sources in the region. The surface conditions change as a function of time--agriculture and fire being the dominant influences in the region surrounding the site. The sand dune area immediately upwind of the site is a potential local source of dust, as are the agricultural areas on the other side of the Columbia River. The dust loading profile will depend on the distance to the source of dust. Very high loadings near the ground are typical of nearby sources (Table 7) and more uniform profiles of remote sources (Table 8).

If a significant change occurs in the surface stabilization upwind of the site, then a resultant change will occur in the dust magnitudes and frequencies.

The dust storm climatology and dust loading profiles provide estimates of the magnitude and frequency of dust storm occurrences over the period 1953 to 1970 for the Hanford region based both on changes in weather and surface stabilization.

REFERENCES

1. Orgill, M. M., G. A. Sehmel, and T. J. Bander, 1974, "Regional Wind Resuspension of Dust," Pacific Northwest Laboratory Annual Report for 1973, to the USAEC Division of Biomedical and Environmental Research, Part 3, Atmospheric Sciences, BNWL-1850, pp. 214-219.

2. Hagen, L. J. and N. P. Woodruff, 1973: "Air Pollution from Dust Storms in the Great Plains," Atmos. Environ., 7, pp. 323-332.

3. Sehmel, G. A., 1976: "The Influence of Soil Insertion on Atmospheric Particle Size Distributions," Pacific Northwest Laboratory Annual Report to ERDA Division of Biomedical and Environmental Research, Part 3, Atmospheric Sciences, BNWL-2000, pp. 99-101.

4. Hilst, G. R. and P. W. Nickola, 1959: "On the Wind Erosion of Small Particles," Bull. American Meteorology Soc., 40, pp. 73-77.

Q 372.9
(2.3.3)

Describe how you determined the vertical temperature differences (e.g., either temperature subtraction or an electrical bridge system).

Response:

See revised page 2.3-33.*

*Draft attached.

Q 372.9
(2.3.3)

Describe how you determined the vertical temperature differences (e.g., either temperature subtraction or an electrical bridge system).

Response:

See revised page 2.3-33.*

*Draft attached.

Q 372.10
(2.3.3)

Describe how "corrections to the data have been applied per the quarterly calibration findings." (Refer to Section 2.3.3.1 of the FSAR.)

Response:

The reference stated in the above should be FSAR 2.3.3.2.2 rather than 2.3.3.1. See revised Section 2.3.3.2.2 for the response.*

*Draft of revised pages attached.

during the second annual cycle of monitoring, except possibly during December 1975 when a sand plug was discovered in the rain gauge funnel. As a result, it was estimated that less than 0.1 inch of precipitation was not recorded during that month. For the first annual cycle of monitoring, the data recovery rate was 96% or better for all meteorological quantities; this rate was 97% or better during the second annual cycle of monitoring.

2.3.3.2.2 Maintenance and Calibration

Assurance of quality data rests primarily with the calibrations performed at quarterly intervals and reported for July and October 1974; January, April, July and October 1975; and January and April 1976.

All evidence to date obtained through formal calibrations and routine daily and weekly inspection had demonstrated that the meteorological system remained electronically stable in terms of obtaining data of sufficient quality to meet the requirements in Regulatory Guide 1.23. ~~Corrections to the data have been applied per the quarterly calibration findings and all data have been summarized in the form of monthly reports.~~

2.3.3.2.3 Data Processing and Analysis

For the two years of onsite FSAR meteorological monitoring at WNP-2, all data (magnetic tape and strip chart where required) were run through computer edit programs. No data were found to be unreasonable except for known causes as documented in Nonconformance Reports. Data corrections, per the Calibration Reports, were applied in the computer programs. Summarization of data has been accomplished only at such times as calibration information was available to bracket in time the acquired data.

For each hour of collected meteorological data, a representative half hour interval was chosen to compute the hourly value of the selected parameters. All data products were based on these "hourly" averages. Wet bulb from the permanent tower was obtained from standard psychrometric formulas presented in the Smithsonian Meteorological Tables, 1971 issue.

The calibration corrections required are tabulated in the response to NRC question 6.4 on the FER. The calibrations established any system inaccuracies by comparisons to standards. These inaccuracies were corrected by appropriately adjusting the data at the data processing stage as opposed to adjusting the system electronics. The calibrations before and after each calibration period were used to determine if corrections were required to account for drift or if offset had occured. No drift corrections were required. The offsets were discussed in the FER response. For corrections that were not constant throughout the range of a given parameter, a calibration table or curve was used to correct the data. Calibration corrections were applied as part of the computer programs used to edit and translate the data from the original raw-data tapes to a master-file of hourly values.

Q 372.11
(2.3.3)

You state in Section 2.3.3.2 of the FSAR that "for each hour of...data, a representative half-hour interval was chosen..." Describe how this representative half-hour was chosen. Since your primary data recording system was digital, explain why the entire hour wasn't used.

Response:

See revised section 2.3.3.2.3.*

*Draft attached.

during the second annual cycle of monitoring, except possibly during December 1975 when a sand plug was discovered in the rain gauge funnel. As a result, it was estimated that less than 0.1 inch of precipitation was not recorded during that month. For the first annual cycle of monitoring, the data recovery rate was 96% or better for all meteorological quantities; this rate was 97% or better during the second annual cycle of monitoring.

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All evidence to date obtained through formal calibrations and routine daily and weekly inspection had demonstrated that the meteorological system remained electronically stable in terms of obtaining data of sufficient quality to meet the requirements in Regulatory Guide 1.23. Corrections to the data have been applied per the quarterly calibration findings and all data have been summarized in the form of monthly reports.

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~~For each hour of collected meteorological data, a representative half hour interval was chosen to compute the hourly value of the selected parameters.~~ All data products were based on these "hourly" averages. Wet bulb from the permanent tower was obtained from standard psychrometric formulas presented in the Smithsonian Meteorological Tables, 1971 issue.

The data for each hour is represented by an average of the data for the last 30 minutes in the hour. The averaging period of 30 minutes was selected for consistency with 1) the data used to formulate the Hanford Diffusion Model used for routine and accident dose calculations, 2) the recommendations in Regulatory Guide 1.23, and 3) computational economy. The only exception was wind direction which was averaged over one-hour to facilitate the formulation of wind direction persistence summaries.

One thirty-minute period per hour is considered adequate for climatological summaries consisting of averages of many hours. In addition, X/Qs based on thirty-minute averages will be conservative for estimates of the one-hour averages.

Q 372.12
(2.3.3)

Provide in Section 2.3.3.2 of the FSAR, a description of the proposed display for monitoring meteorological parameters in the control room, including a description of how the data will be displayed (e.g., strip-chart recorders and/or digital readout). Indicate whether the displayed data will be time-averaged or will be instantaneous.

Response:

See revised section 2.3.3.2.4.*

*draft attached.

In several of the monthly summary reports, the computer programs as applied to dummy data have been compiled as called for in the Quality Assurance Manual[23] for the purpose of documenting proper programming and proper computer performance.

These computer computations have been verified with hand calculations made with the dummy data. The computational programs for χ/Q were similarly tested.

2.3.3.2.4 Meteorological Monitoring Program During Plant Operation

The WNP-2 operational meteorological program will commence at fuel load. The system will be put in operation at least two months prior to fuel load to ensure reliable operation at fuel load. System measurements will include wind speed and direction and temperatures at 245 and 33', Δt between 245 and 33', and dewpoint at 33'. A quality assurance program will be utilized to ensure measurement accuracies within those recommended by Regulatory Guide 1.23. Those parameters which will be multiplexed to the control room include wind speed and direction from 245' and 33' and the Δt between 245' and 33'.

2.3.3.3 Other Meteorological Measurement Programs Considered for the Data Comparisons

2.3.3.3.1 WNP-2 Temporary Tower

A temporary 23 foot onsite tower was used during the period April 1, 1972 through August 31, 1974 to obtain data input for WNP-2 environmental studies and to provide a comparative overlap with the initially measured permanent tower data.

The temporary tower was located in the vicinity of the permanent towers with its base at approximately 448 feet MSL. Wind data from the temporary tower were obtained at the 23 foot level while temperature data were acquired at the three foot level. Wet bulb data from the temporary tower were established from techniques and data contained in the U.S. Department of Commerce, Weather Bureau Office Document: Relative Humidity and Dewpoint Table. As a special quality assurance program was not initiated for the temporary tower installation, it is not possible to assert that this tower's data complied with the requirements contained in Regulatory Guide 1.23.

2.3.3.3.2 Hanford Meteorological Station

The Hanford Meteorology Station (HMS) is on a plateau in south central Washington. The plateau slopes downward toward the Columbia River, about 10 miles north of the station and about 300 feet lower in elevation. Elevation of the station is 733 feet MSL. The nearest city is Richland, about 25 miles southeast. There is no local vegetation which can interfere with the HMS.

The control room displays will consist of instantaneous analog (strip chart) values of each of the multiplexed parameters.

Q 372.13
(2.3.1)

Indicate the period of record you examined to choose your design meterological data for performing your analysis of your ultimate heat sink. Describe the selection process you used to establish the limiting one- and thirty-day periods.

Response:

For the selection of the worst one day for thermal performance, data from the Hanford Meteorology Station for January 1955 through December 1975 were scanned for episodes of high daily average wet-bulb temperature. Three episodes were identified (see FSAR Section 2.3.1.2.3) which were then scanned for the period with the maximum 24-hour average wet-bulb temperature. The 24-hour period identified was from July 10, 1975 at 0700 hours through July 11 at 0600 hours as measured by the permanent onsite tower. The conditions of this period were assumed to persist during the initial period of pond loading as described in Section 9.2.5.

The worst 30-day period for pond thermal performance was selected by calculating running 30-day averages of the daily maximum dry-bulb temperatures for the 1955-1970 HMS data. This resulted in the identification of the July 9 - August 8, 1961 period which was used to model the remainder of the 30-day pond thermal performance as described in Section 9.2.5.

The worst period for water loss was selected by searching the HMS 1955-1970 data for the 30-day period with the maximum average difference between wet-bulb and dry-bulb temperature. This was done by calculating 30-day running average differences for the data period. The meteorology of the resulting 30-day period was the basis for the water loss modeling described in Section 9.2.5.

372.14 Based on our review of your use of the Hanford diffusion coefficients (σ_y and σ_z as classified by $\Delta T/\Delta Z$, σ_Θ, and $\overline{U}$) for short-term diffusion estimates, we find that the calculational scheme you propose is not sufficiently detailed to evaluate those periods which have the least atmospheric dispersion and which are important in establishing design values of χ/Q. In particular, we compared the Hanford diffusion parameters and the associated classification schemes with those: (1) contained in the draft regulatory guide discussed in Appendix A to this section; (2) contained in Section 2.3.4 of the Standard Review Plan (SRP); and (3) derived by Markee for a desert site.* The last of these three schemes included data from the National Reactor Test Station and from the Hanford reservation.

a. Based on this comparison, we find that for vertical dispersion, the Hanford parameters you propose compare well with the Markee desert parameters, except that your proposed use of the Hanford stability classification scheme does not define parameters for the extremely stable atmospheric conditions labelled as Class G (a vertical temperature gradient greater than 4.4°F for a 200 foot differential height). Thus, when your proposed diffusion parameters are used in the models contained in the first two calculational schemes cited above (i.e., the model of draft Regulatory Guide 1.xxx using the Markee desert coefficients instead of the Pasquill-Gifford coefficients with meander and the model contained in Section 2.3.4 of the SRP , we conclude that your proposed parameters will not provide a sufficient description of those periods with the poorest vertical dispersion, thereby underestimating the calculated χ/Q values used in evaluating dose rates. For example, Stability Class G occurred 13 percent of the period of record measured on the WNP-2 meteorological tower between April 1974 and March 1976. To correct this significant omission in your methodology, we require that you provide constants for Equation 5 of your proposed model for these extremely stable periods and to add a stability class identified as "extremely stable" to Tables 2.3-28a through 2.3-32b of the FSAR.

b. We also find that the σ_Θ can be a predictor to estimate σ_y during those periods when the wind sensors properly represent the actual windflow. However, during periods of low windspeeds (i.e., windspeeds less than about 1.5 meters/second), most windvanes do not respond properly to directional changes of the windflow. Accordingly, during these periods, the vertical temperature gradient should be used to estimate lateral dispersion. We require you to: (1) state that Equations 3 and 4 of your proposed methodology are applicable to windspeeds greater than 1.5 meters/second; and (2) to use Regulatory Guide 1.23 or your Hanford scheme (with Class G) to define lateral dispersion for low windspeeds. Alternatively, provide justification for using σ_Θ to predict σ_y for low windspeeds. Amend your values of χ/Q to reflect any changes resulting from your responses to Items 372.14 and 372.15.

Reference

*Yanskey, Markee, and Richter, 1966. "Climatography of the National Reactor Testing Station," IDO-12048, Air Resources Field Research Station, Idaho Falls, Idaho.

a. A review of the Hanford model for σz (Eq. 5, FSAR pg 2.3-43) and Hanford experimental diffusion data indicates that in fact this formulation conservatively represents vertical plume growth under restrictive dispersion conditions including stability Class G. This conclusion is demonstrated below by 1) comparing Eq. 5 (FSAR pg. 2.3-43) to experimental diffusion data for Hanford tests conducted under G thermal stability, and 2) comparing Eq. 5 to Pasquill - Gifford and Markee approaches.

The fact that Eq. 5 predicts a conservative σz during extremely stable thermal conditions (G) is illustrated in Figure 372.14-1. In the figure, σ_Z as predicted by Eq. 5 (very stable) is plotted versus σz's determined experimentally during five ground-level tracer experiments conducted during G thermal stabilities. The actual experimental measurements were of σy and plume centerline concentrations and the σz's were back-calculated from these measurements assuming a Gaussian model as described by Eq. 1 (FSAR page 2.3-42). For the σz's characteristic of the exclusion radius and LPZ distances, Figure 372.14-1 indicates that the predicted σz's are conservative up to a factor of about 20, and more for greater distances. Therefore, the field measurements do not support the need for an "extremely stable" classification in the Hanford diffusion model. The σz values computed for the Hanford "very stable" category already display a significant conservatism even for Class G thermal stabilities. If any adjustments are warranted they would be to reduce this conservatism.

Figure 372.14-2 compares σ's calculated by Eq. 5 of the FSAR for the Hanford very stable class with the comparable values calculated with Pasquill-Gifford G and Markee's most stable desert σz's. Since in the Hanford model σz is a function of both distance and wind speed, at a given distance, σz is a continuous function depending on the wind speed. Thus the Hanford model provides the greater ability to identify and treat the worst case limited dispersion conditions than any of the suggested alternative schemes for selecting σz.

In the figure, it can be seen that the Hanford σz very stable model will be more conservative than Pasquill-Gifford G for all wind speeds greater than about 0.5 m/s. From the dispersion climatology tables in the FSAR it can be determined that this was the case more than 95% of the very stable hours. Similarly, the Hanford very stable model will be more conservative than the most stable Markee desert curve whenever the wind exceeds about 2.0 ms. This occurred slightly more than 40% of the very stable hours.

b. This question addresses 1) the validity of Equations 3 and 4 at low speeds and 2) the validity of using windvane measurements of $\sigma\Theta$ at speeds near and below the vanes threshold as input to Equations 3 and 4 to estimate σy under this conditions.

With respect to Item 1), the Hanford σy model is usable at low wind speeds because its development and the evaluation of model coefficients were not limited by the response of instruments used during the Hanford dispersion experiments. The basic σy model relating σy to $\sigma\Theta\ \overline{u}$ and travel time was formulated theoretically by GITaylor (1921). The model is

$$\sigma y^2 = A(t - \alpha(1 - e^{-t/\alpha}))$$

where t is the travel time between source and receptor, and A and α are to be determined. Fuquay, Simpson and Hinds (1964) developed the following expressions for A and α:

$$A = 13 + 232\ (\sigma\Theta\ \overline{u})$$

and,

$$\alpha = \frac{A}{2(\sigma\Theta\ \overline{u})^2}$$

These relationships, combined with Taylor's basic equation form the Hanford σy model.

The constants in the relationship between A and $\sigma\Theta\ \overline{u}$ are related to general characteristics of atmospheric turbulence (Slade, 1978). As such, they are related to terrain. The constant values given are appropriate to terrain typical of Hanford. They are reported to be appropriate for the National Reactor Test Site in Idaho, but a factor of 2 too large for very flat terrain found in the mid-west (Watson and Simpson, 1967).

Evaluation of the constants was made using dispersion data collected during wind speeds generally ranging from 1.5 to 7.2 m s^{-1}, with one experiment conducted in a wind speed of 0.9 m s^{-1}. Response of the wind sensors used during the experiments (Aerovanes) at wind speeds below 1.5 m s^{-1} did not significantly affect the evaluation of the constants. The data plotted in Figure 372.14-3 were obtained during "very stable" atmospheric conditions. Wind speeds during the diffusion tests in which these data were collected ranged from 1.2 to 2.5 m s^{-1}. The data from the tests with mean wind speeds of 1.5 m s^{-1} and below, shown by x's in the Figure, are as well modeled as the data from higher wind speed tests. Thus the limitation on applicability of the model in low wind speed conditions is not inherent in the model but is solely a function of the response characteristics of the instruments used to determine $\sigma\Theta$ and $\overline{u}$.

The instantaneous response threshold of the wind vanes on the WNP-2 meteorological tower is about 0.3 m s^{-1}. In the two years meterological data collection period the reported mean wind speed for the hour (30-minute average) fell below the value only 16 times, and only 6 of these occurrences were during very stable atmospheric conditions. These 16 cases account for only 0.09% of the time. Therefore the χ/Q values computed during these 16 hours are not a significant factor in estimating either the 5% or 50% χ/Q values for the data set in question.

With respect to the conservatism of the method, the following observations are important. In considering instrument threshold, it is important to recognize that the threshold value is not a half-hour or hourly average wind speed but an instantaneous value. Rather wind instruments respond to essentially instantaneous winds. If the speed during a gust exceeds the instrument threshold, there will be a response. As the fraction of the time in which the wind speed is below instrument threshold increases, the error in estimates of $\bar{u}$ and $\sigma\Theta$ will increase. The mean wind speed will always be underestimated and $\sigma\Theta$ should tend to be underestimated. Both factors contribute to increasingly conservative diffusion estimates at low wind speeds.

Figure 372.14-4 demonstrates the tendency to underestimate the average value of $\sigma\Theta$ at low average wind speeds for the WNP-2 site. As the wind speed decreases the values of $\sigma\Theta$ become larger as long as the wind speeds are above the vane threshold speed. The average $\sigma\Theta$ for wind speeds below the vane threshold are less than would be expected from a projection of the relationship at higher speeds. This projection is supported by the increasing tendency for plume meander with decreasing wind speed reported by Vander Hoven (1976), Gifford (1976), Nizkola, Clark and Ludwick (1975) for example. The difference can be largely attributed to lack of vane response. These data support the use of the measured $\sigma\Theta$ values in computing χ/Q even when the average wind speed falls below the vane threshold and indicates that this practice is conservative.

In summary, our review of the issues raised indicates that the model and methods used to calculate routine and worst case χ/Qs are technically justified and conservative. If any adjustments are warranted, they would be in the direction of reducing the amount of conservatism.

<u>References</u>

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Gifford, F. A., "Turbulent Diffusion Typing Schemes: A Review," <u>Nuclear Safety</u>, Vol. 17, No. 1, pp. 68-86, 1976.

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Slade, D. H. (Ed.), <u>Meteorology and Atomic Energy - 1978</u>, USAEC Report TID-24190, pp. 135-142, Environmental Science Services Administration, 1968.

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Van der Hoven, I., "A Survey of Field Measurements of Atmospheric Diffusion under Low-Wind Speed Inversion Conditions," Nuclear Safety, Vol. 17, No. 2, pp. 223-230, 1976.

Watson, E. C. and C. L. Simpson, "Effects of Wind Variability on Environmental Consequences of Prolonged Release of Radioactive Contaminants," BNWL-SA-1058, Battelle, Pacific Northwest Laboratory, Richland, WA 99352, February 20, 1967.

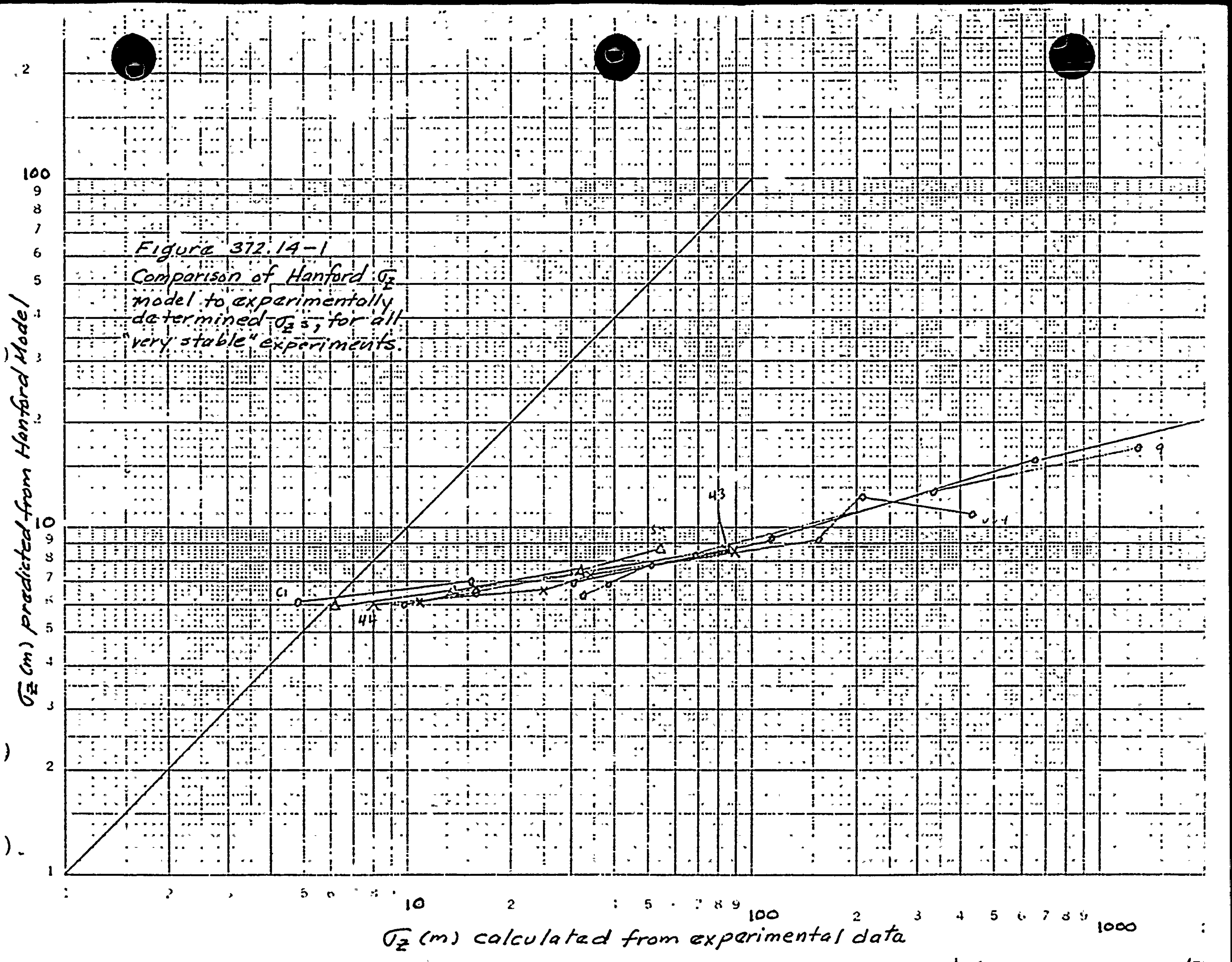

Figure 372.14-1
Comparison of Hanford σ_z model to experimentally determined σ_zs, for all "very stable" experiments.
σ_z (m) predicted from Hanford Model
σ_z (m) calculated from experimental data
100
10
1000
43
44
9

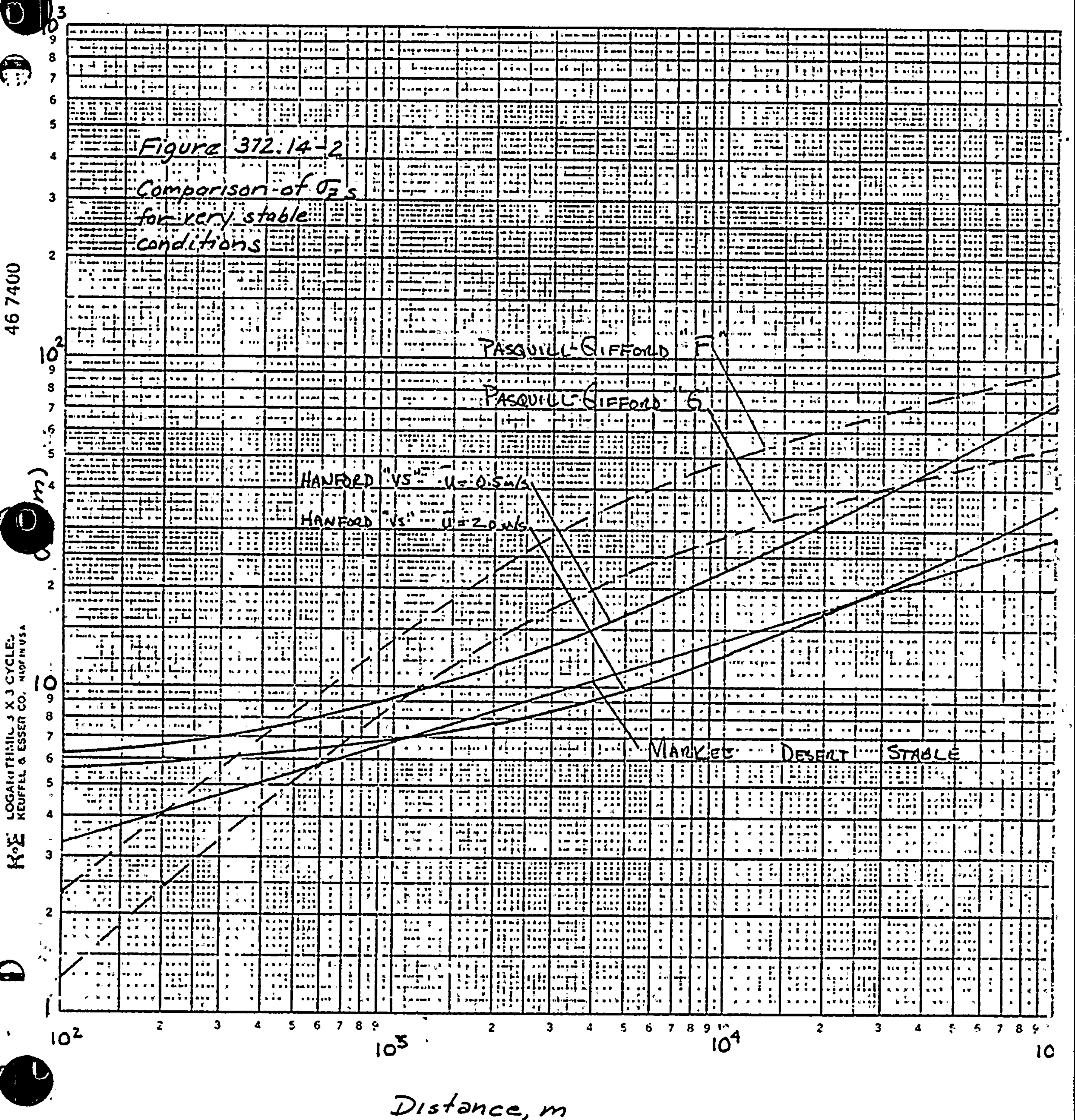

Figure 372.14-2
Comparison of σ_z's
for very stable
conditions
PASQUILL-GIFFORD F
PASQUILL-GIFFORD G
HANFORD "VS" U = 0.5 m/s
HANFORD "VS" U = 2.0 m/s
MARKEE DESERT STABLE
Distance, m
46 7400
K&E LOGARITHMIC 3 X 3 CYCLES
KEUFFEL & ESSER CO. MADE IN U.S.A.

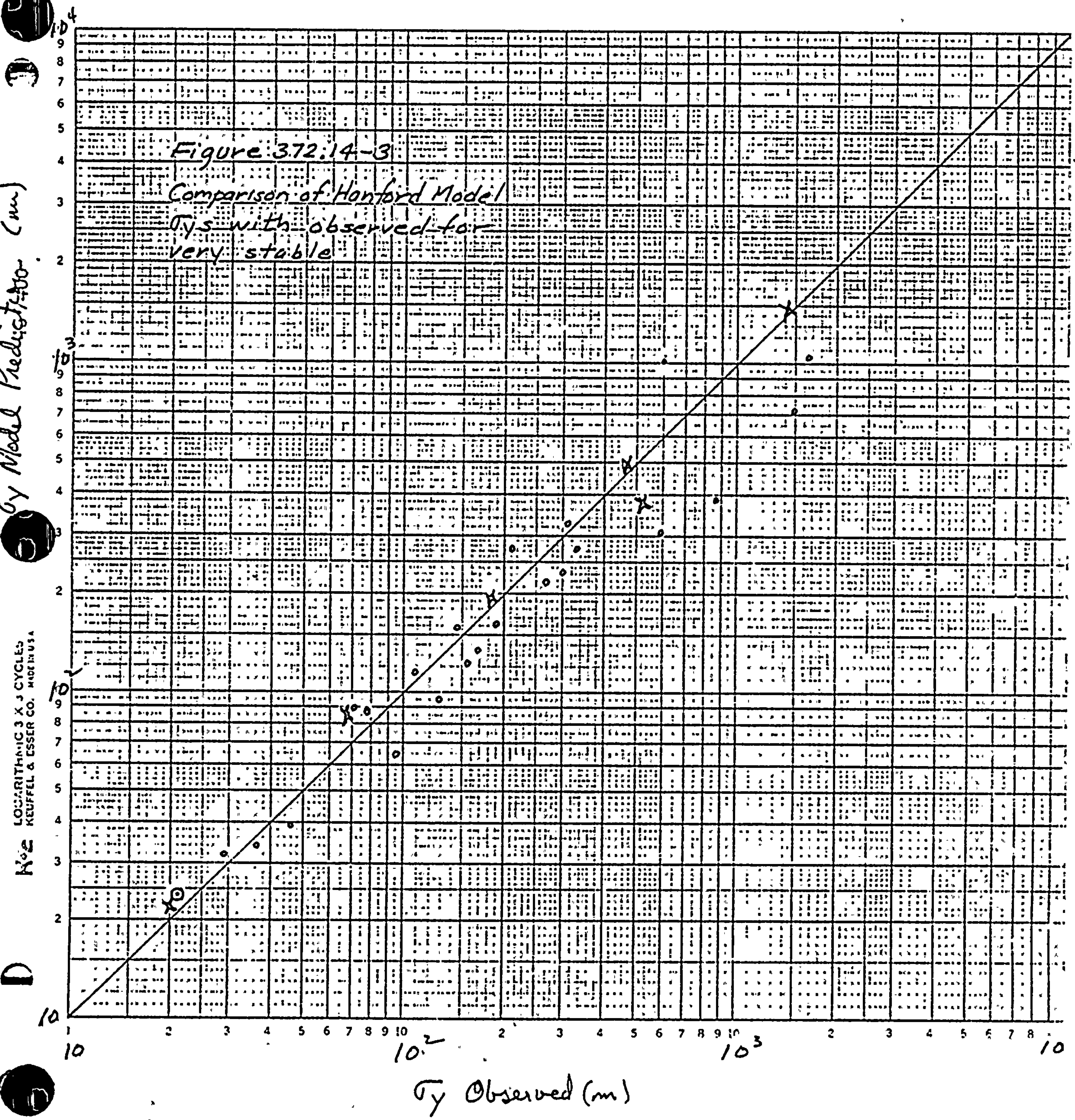

Figure 3.72.14-3
Comparison of Hanford Model σy's with observed for very stable

46 1510

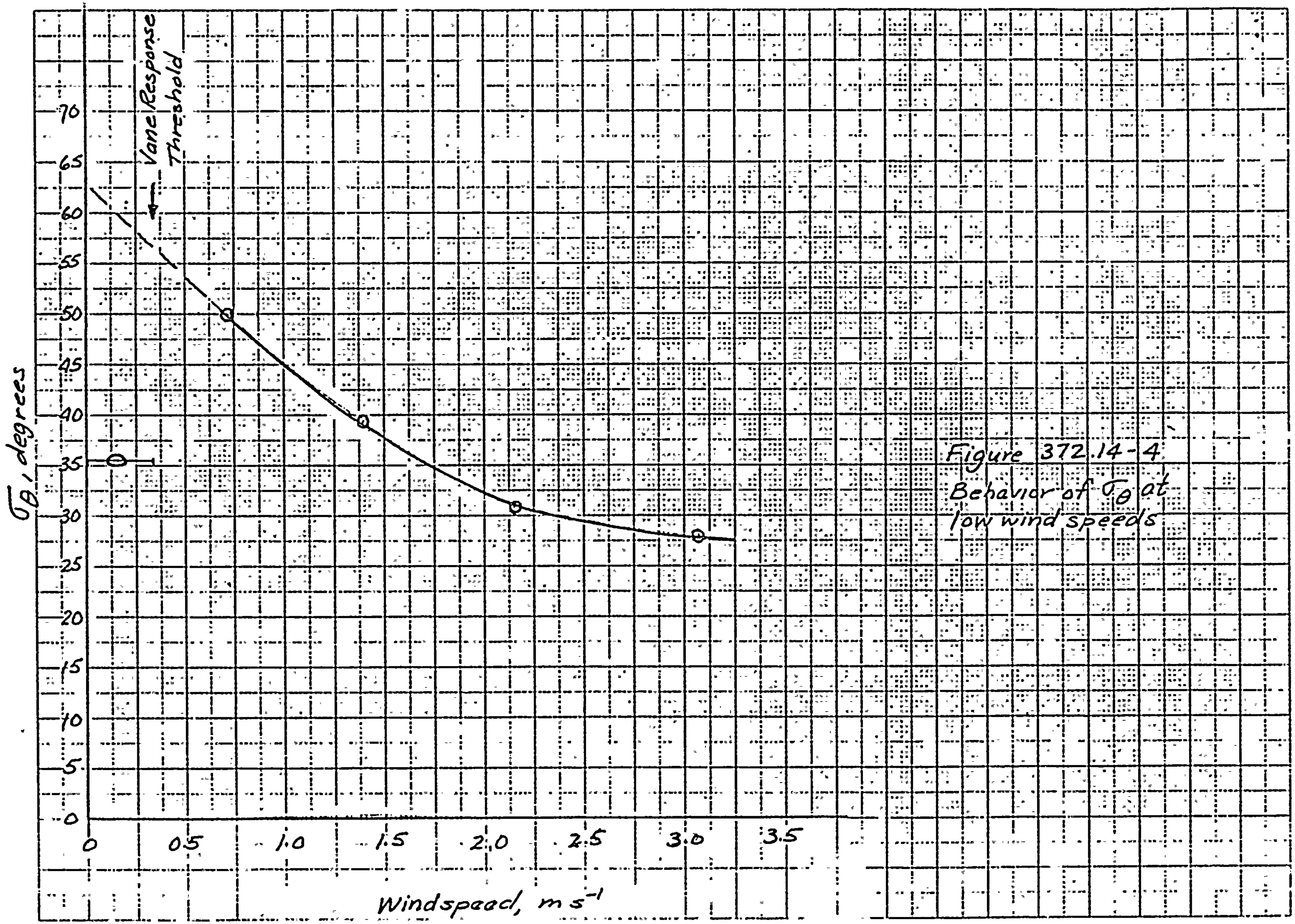

Figure 372.14-4
Behavior of σ_θ at low wind speeds

372.15 Discuss the meteorological instrumentation, including instrument response characteristics, used during the field tests from which the Hanford dispersion coefficients were derived. Compare the response characteristics of the instrumentation on the WNP-2 meteorological tower with the instrumentation of the Hanford field tests. Discuss whether use of the WNP-2 data is compatible with the Hanford diffusion model. For example, are the response characteristics of the windvanes similar and would they produce similar σ_Θ's. Your response to Items 372.15 and 372.16 will permit us to determine the acceptability of σ_Θ as discussed in Item 372.14(b).

Response: The wind instruments used during the Hanford dispersion experiments were Aerovanes with a threshold of 1 to 1.5 m/s, a distance constant for wind speed of about 5.3 m, a gust wave length for direction of about 15 m and a damping ratio of about 0.3. These response characteristics are estimates typical of Aerovanes; the response characteristics of the actual instruments used has not been reported. The values of σ_Θ used in the analysis leading to the Hanford σ_y model were computed from 20 s direction averages and the 30 min mean direction. The 20 s averaging, not anemometer response, was the factor in determining the high frequency limitation on wind direction oscillations included in σ_Θ. The total length of record analyzed and the initial averaging of the wind direction prior to computing σ_Θ form a band pass filter that results in underestimates of σ_Θ. Using a spectral model appropriate for low wind speed, very stable conditions at Hanford and the filter function described by Pasquill (1974), the underestimate of σ_Θ has been determined to be about 25%. It results from filtering out small, high frequency turbulent fluctuations. The fluctuations filtered out have little affect on diffusion at distances beyond the first few meters from the source.

sec

The wind direction instruments on the WNP-2 tower are Climet Instruments, Model 012-10 vanes with a response threshold of about .34 m/s, a gust wave length of about 1 m, and a damping ratio of about .4. These instruments are much more responsive than those used in the experiments leading to development of the Hanford σ_Θ model.

The WNP-2 values of σ_Θ were computed using a scheme that differs significantly from that used for the σ_Θ's reported with the Hanford diffusion data. The total wind direction variance (σ_Θ^2) is estimated in two parts, and the contributions combined. The standard deviation of wind direction fluctuations is computed electronically and recorded each 5 minutes along with the logarithmically averaged 5 minute mean direction. The average of the square of the first six electronic σ_Θ's was taken to represent the variance of wind direction fluctuations associated with the high frequency portion of the turbulence spectrum. The variance

associated with the low frequency portion of the spectrum (meander) was estimated from the variation of the first six 5 min average wind directions each hour about the 30 min average direction. Again applying the by pass filter approach described by Pasquill (1974), it is estimated that using this technique, σ_Θ computed for WNP-2 could be underestimated by about 2%. However, the 5 min average directions are not totally independent because of the electronic averaging circuitry. As a result, the computed values of σ_Θ will be underestimated somewhat more than 2%.

Comparing the σ_Θ estimates in the two cases, the Hanford experimental values underestimate primarily by failing to incorporate the high frequency wind direction fluctuations, while the WNP-2 values underestimate by failing to account for low frequency fluctuations. However the Hanford model does not differentiate between low and high frequency wind direction fluctuations. It is only sensitive to the magnitude of σ_Θ.

Combining the characteristics of both methods of computing σ_Θ, it is estimated that

$$1.0 \leq \frac{(\sigma_\Theta)\text{WNP-2}}{(\sigma_\Theta)\text{Hanford}} \leq 1.3$$

for typical stable atmospheric conditions for a height of 10 m and a wind speed of 2 m/s. The most probable ratio is somewhat greater than 1.0 but less than 1.3. As a result, the use of WNP-2 in the Hanford model will tend to over predict σ_y and therefore lower χ/Q estimates.

The effect of a 20% difference in estimates of σ_Θ on estimates of σ_y for a wind speed of 2 m/s^{-1} is shown in Figure 372.15-1. The solid curve is for $\sigma_\Theta = 2°$ and represents σ_y based on the original Hanford σ_Θ values, and the dashed curve is for $\sigma_\Theta = 2.4°$ and represents σ_y based on probable WNP-2 σ_Θ values. The overestimate in σ_y resulting from instrumentation and computational differences decreases from about 20% at x = 100 m to about 6% at x = 100 Km.

The striking feature apparent in the figure is the small change in position of the curve resulting from a 20% change in σ_Θ. This change in position is insignificant when compared with the uncertainty associated with the placement of the Pasquill G curve.

When considered with the conservative estimates of σ_Z produced by the Hanford model, the very slightly non-conservative effect of using WNP-2 σ_Θ values in estimating σ_y is not significant. When used together with the WNP-2 meteorological data the Hanford models for σ_y and σ_Z will result in conservative estimates of χ/Q in very stable atmospheric conditions. The maximum change of about 20% that would occur in χ/Q values if a 20% correction were applied to WNP-2 σ_Θ values is less than the uncertainty in the χ/Q estimates due to basic problems with all diffusion models including the Gaussian and more sophisticated models.

Reference

Pasquill, F., Atmospheric Diffusion, 2nd Ed., Halsted Press, New York, NY, 1974.

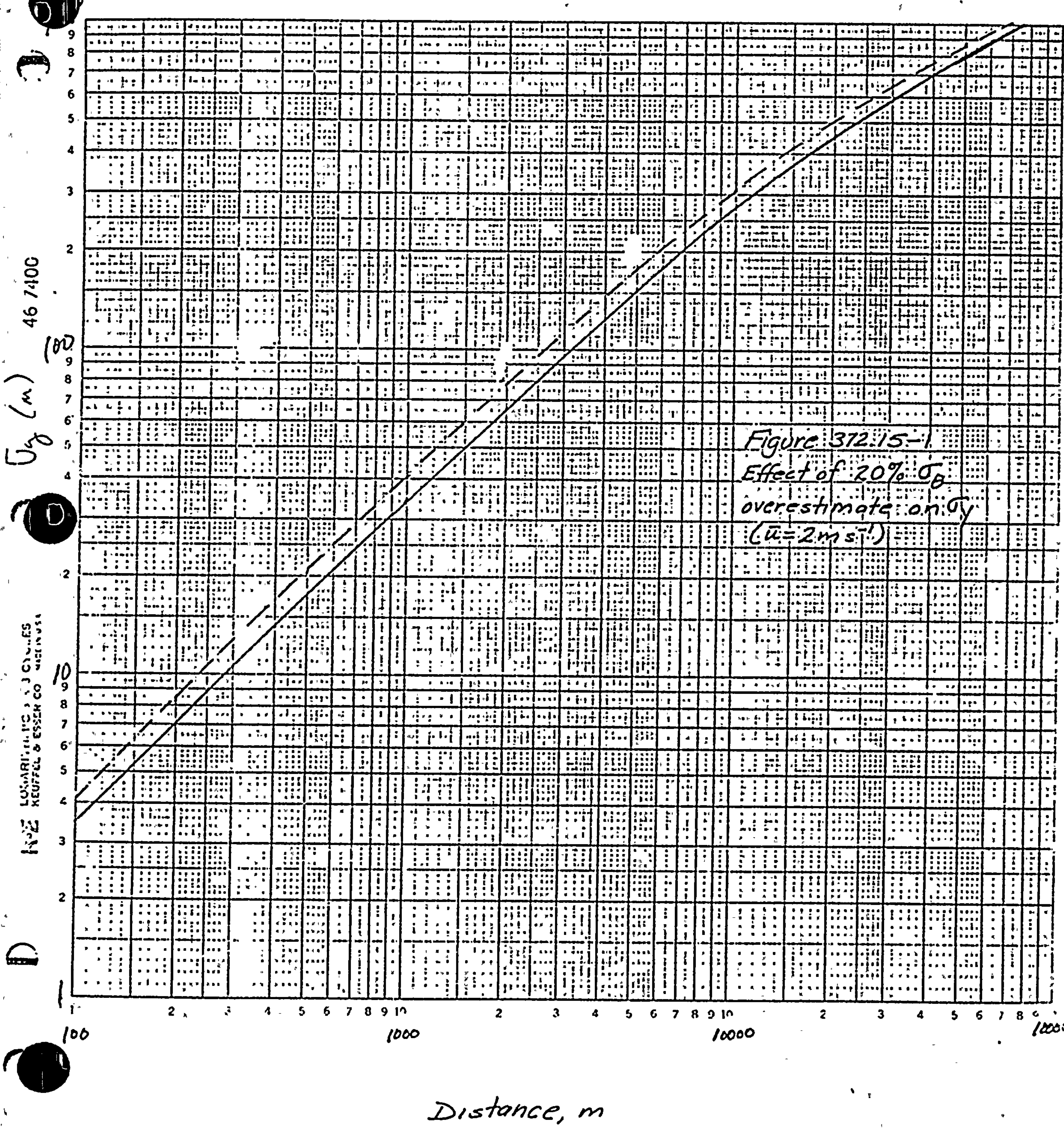

Figure 372.15-1
Effect of 20% σ_θ overestimate on σ_y
($\bar{u}$=2 m s^{-1})
σ_y (m)
10
100
100
1000
10000
Distance, m
46 7400
KEUFFEL & ESSER CO.

372.16 Discuss whether the Hanford test data covers the same range of meteorological conditions for the parameters observed on the WNP-2 tower (i.e., windspeed, vertical temperature gradient and σ_Θ) and whether the Hanford model is valid for all meteorological conditions observed on the WNP-2 tower.

Response: The field data used in development of the Hanford diffusion model were collected during the period 1957 to 1962. Forty-six of the approximate seventy field experiments used in development of that model are summarized in a journal article by Fuquay, Simpson and Hinds. Although it is known that five of the remaining experiments were in the moderately stable to very stable range, and eighteen were in the neutral to unstable range, the experiments were not documented in a form from which it is possible to extract the meteorology in a form responsive to question 372.16. Hence all response to this question with respect to the Hanford test data is based on the 46 experiments reported by Fuquay, et al. The WNP-2 data to which comparison is made is based on the two-year period of record documented in the FSAR.

The mean wind speed observed at the WNP-2 tower was 8.1 mph. Only one instance of calm (for a "representative half hour") was recorded during these two years. Wind speeds of ≥ 25 mph were reported 1.2% of the time, and speeds >19 mph were recorded 4.4% of the time.

The mean wind speed observed during the 46 Hanford tests was 7.4 mph. The lowest speed observed during a test was 1.6 mph, and the highest was 16.1 mph.

During very stable conditions ($\Delta T > 3.5F°/200$ ft), the wind speeds observed at WNP-2 averaged 4.5 mph, the same mean speed observed during the five Hanford tests under the same stability conditions. At WNP-2 winds ranged in speed from 1 to less than 19 mph during very stable conditions. More than 99% of the WNP-2 very stable cases occurred during winds less than 13 mph, and about 90% occurred during winds less than 8 mph. The very stable Hanford tests were conducted during speeds ranging from 3.4 to 5.6 mph.

During moderately stable conditions ($3.5° > \Delta T > -0.5°$), the wind speed averaged 8.5 mph at WNP-2 during the two-year period. The range in speed was from calm (1 case) to greater than 25 mph. About 97% of the speeds during moderately stable conditions averaged less than 19 mph. The mean wind speed during the 25 Hanford moderately stable tests was 6.6 mph, with a range from 1.6 mph to 12.5 mph.

Vertical temperature gradients applicable to the layer between 233 and 33 feet are summarized in the following tables. WNP-2 data result from measurements made at 250 ft and 33 ft. The Hanford data result from smoothed temperature profiles graphed to measurements made at elevations of 3, 50, 100, 150, 200, 250, 300 and 400 feet.

Percent of Data in Specified Stability Class

	Pasquill-Gifford Class						
	A	B	C	D	E	F	G
WNP-2+	3.6	4.0	9.3	26.7	26.7	16.2	13.5
Hanford*	4.3	3.3	5.4	17.4	43.4	19.6	6.5

	Hanford Stability Class				
	Very Unstable	Unstable	Neutral	Moderately Stable	Very Stable
WNP-2+	0.4	19.7	23.7	38.5	17.8
Hanford*	2.2	12.0	18.5	57.7	9.8

For conveniences, a graphical comparison of the Hanford and Pasquill-Gifford stability increments is given on Figure 372.16-1.

Values of σ_Θ computed for the Hanford test periods ranged from 3.6 to 18.3 degrees. The distribution of σ_Θ during the Hanford tests is given in the following tables.

Percent of Data in Specified σ_Θ Interval

	σ_Θ, degrees			
	0-5	5-10	10-15	15-20
Hanford*	17.4	54.3	19.6	8.7

Since the parameter $\sigma_\Theta\ \bar{u}$ is used in the prediction of σ_y in the Hanford model, it is instructive to examine the $\sigma_\Theta\ \bar{u}$ distributions for the WNP-2 and Hanford test data. The following table shows that

Percent of Data in Specified $\sigma_\Theta\ \bar{u}$ Intervals

	$\sigma_\Theta\ \bar{u}$, degree-miles/hour						
	0-20	21-40	41-100	101-200	201-300	310-400	>400
WNP-2+	3.4	10.7	46.2	31.5	7.4	0.6	0.3
Hanford*	8.7	21.7	58.7	10.9	-	-	-

although the range of $\sigma_\Theta\ \bar{u}$ was more restricted during the Hanford tests, the missing categories were at the larger values of $\sigma_\Theta\ \bar{u}$ (and hence larger values of σ_y) where the impact of associated concentrations would be relatively low.

In addressing the question of whether the Hanford model is valid for all meteorological conditions observed at the WNP-2 tower, perhaps the most instructive comparison is of distributions of WNP-2 and Hanford data within the classification scheme used by the Hanford diffusion model. The following table makes that comparison.

Percentage of Data as a Function of Stability Class and σ_Θ (Hanford*/WNP-2+)

σ_Θ	Stability Class				
(degrees)	Very Unstable	Unstable	Neutral	Moderately Stable	Very Stable
>22	0/0.1	0/7.0	0/5.6	0/8.7	0/6.5
17.5-22	0/0.02	0/2.4	0/2.0	2.2/3.0	2.2/1.8
12.5-17.5	0/0.07	1.1/3.2	2.2/3.5	5.4/4.9	0/2.7
7.5-12.5	2.2/0.1	8.7/3.6	10.9/8.6	19.6/11.0	4.3/3.7
3.75-7.5	0/0.01	2.2/0.5	5.4/3.3	28.3/9.1	3.3/2.2
2.1-3.75	0/0	0/0.01	0/0.03	2.2/0.4	0/0.2
<2.1	0/0	0/0	0/0	0/0	0/0.2

Of the 35 possible combinations of stability and σ_Θ, 19 were observed at WNP-2 with a frequency greater than 0.5%. Hanford test data embraced 13 of these combinations. The missing categories are restricted to the large values of σ_Θ, categories which in general should result in relatively low concentrations.

In summary, the Hanford test data embrace most of the range of meteorological conditions observed at the WNP-2 tower. The frequency of Hanford test data is relatively high for stable atmospheric conditions and relatively small σ_Θ values. The Hanford diffusion model should, therefore, be valid for characterizing diffusion at the WNP-2 site.

Reference

Fuquay, J. J., C. L. Simpson and W. T. Hinds, "Prediction of Environmental Exposures from Sources near the Ground Based on Hanford Experimental Data," Journal of Applied Meteorology, Vol. 3, pp. 761-770, December 1964.

+Data for period April 1, 1974 to March 31, 1976.

*Data from 46 Hanford field experiments documented by Fuquay, Simpson and Hinds (1964).

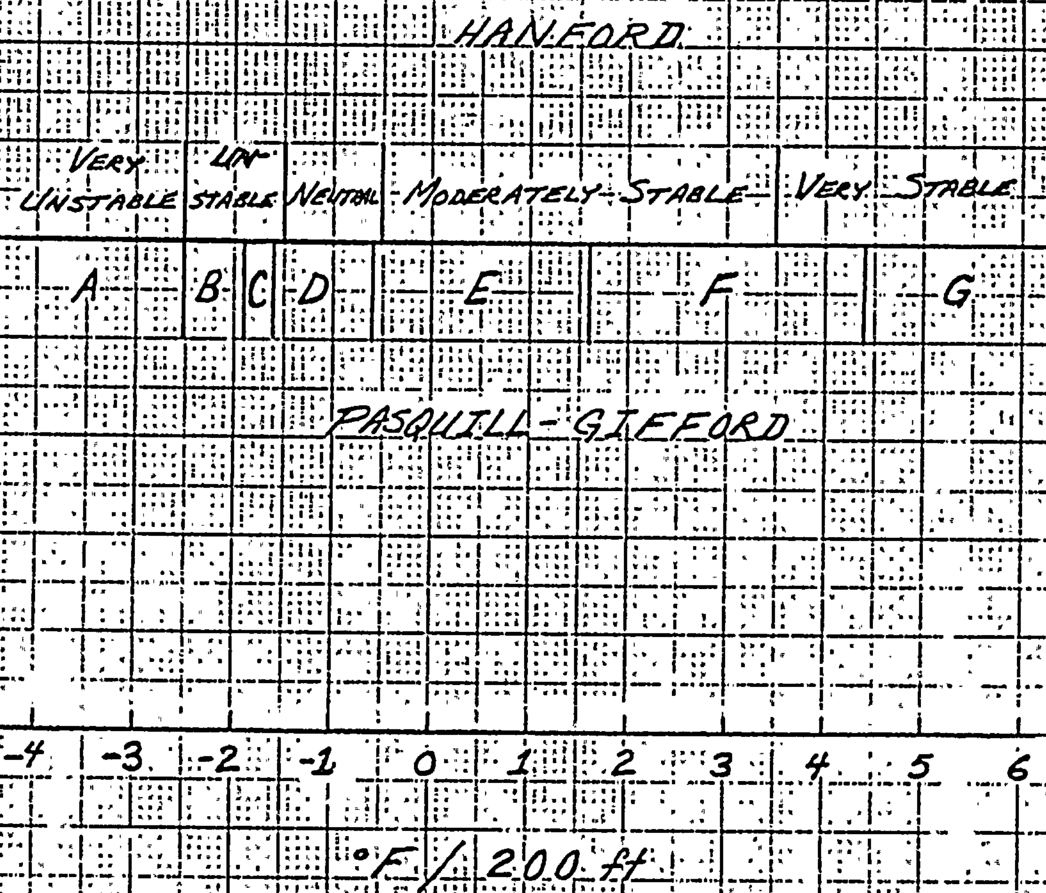

FIGURE 372.16-1

Q 372.17
(2.3.2)

On August 10, 1977, you provided us with a magnetic tape containing meteorological data in hourly form for the period April 1974 through May 1976. Provide the definition of "variable wind" as used on this tape.

Response:

Page 2.3-23 is being revised to show the definition more accurately.*

*See attached draft.

HMS (hist): Data obtained from the Hanford meteorology tower at the Hanford Meteorological Station, used here for various periods identified in the data comparison listings. The source is AEC Research and Development Report "Climatography of the Hanford Area", June 1972, BNWL-1605.

Wind Variable: At WNP-2 an hour of data which contains less than 15 minutes of any one direction sector; at Hanford, the same but for 20 minutes.

Wind Calm: At WNP-2, an hour of data for which the average speed is 0.22 miles per hour or less; at Hanford, average speed less than 1 mph (as decided by weather observer, corresponds to no motion of strip-chart recorded pen).

Sense of Delta T: Positive values imply relative stability, negative values imply relative instability.

The first annual cycle of WNP-2 onsite meteorological data which covered the period April 1, 1974 through March 31, 1975 has been presented in detail. Local meteorological data collected during the second annual cycle (April 1, 1975 through March 31, 1976) generally portrayed the same characteristics as indicated by comparison with the first annual cycle data. Except for the high wet bulb episode experienced at WNP-2 during July 1975 (refer to 2.3.1.2.3 and 2.3.2.3), no monitored onsite data proved to be more severe in terms of the design and operation of WNP-2 than those data presented in 2.3.1.2. Hence, only the second annual cycle monthly averages have been presented in Table 2.3-8a which summarizes the two years of monitored onsite data with concurrently measured and historical HMS data. Any significant differences noted between first and second annual cycle onsite data and concurrently measured WNP-2 and HMS data are discussed in 2.3.2.1. Otherwise, conclusions stated herein for the first annual cycle of data similarly apply to the second annual cycle data. It is observed in Table 2.3-8a that any year to year differences in the summarized monthly mean meteorological data at WNP-2

Responses to:

Geosciences Branch Questions
Geology-Seismology Section (360.4 - 360.5)
Geotechnical Engineering Section (362.5 - 362.9)

Q 360.4

In the Weston Geophysical Research, Inc., report, "Qualitative Aeromagnetic Evaluation of Structures in the Columbia Plateau and adjacent Cascade maountain Area," March 28, 1978, Figure 13 shows several north to northwest trending aeromagnetic linears in the vicinity of Badger Mountain and Jump Off Joe Anticline. However, the Weston report does no discuss the origin or interpretations of these particular linears. The north trending linear crossing the Columbia River at the junction with the Snake River has an apparent offset of the magnetic low defining the Rattlesnake Hills anomaly. Since these aeromagnetic linears trend toward the WNP-2 site, provide: (1) an interpretation of these features, including but not limited to the potential for their continuation to the north to near site area: and (2) a discussion of the fault parameters, if such an interpretation is proposed.

Response:

The Supply System is presently in the process of reviewing information relative to these questions. We will provide a schedule for the submission of a response prior to April 30, 1979.

Q 360.5 Some of the data and discussions in the FSAR of those Columbia Plateau structures relevant to the WNP-2 site are slightly different from the information provided in Amendment 23 to the WNP 1 & 4 PSAR (Docket Nos. 50-460 and 50-513). For example, with regard to the Wallula Gap Fault, your FSAR states that the "...probable fault movement occurred after the deposition of the Touchet beds, and thus less than 12,000 years ago." However, in Appendix 2RH.4 of the WNP 1& 4 PSAR (Amendment 23), you indicate that the fault is older than the Quaternary Kenneweck fanglomerate based on trenching. Additionally, in this same amendment to the WNP 1 & 4 PSAR, you indicate that the faulting along the Horse Heaven Hill Anticline occurred about 3.5 million years before the present (mybp). The WNP-2 FSAR does not discuss this particular point but, rather, questions the existence of faulting along the Horse Heaven Hill Anticline and indicates that it could be the sole result of folding. Clarify these apparent discrepancies and provide cross-references in the WNP-2 FSAR to the appropriate sections of the WNP 1 & 4 PSAR.

Response:

The Supply System is presently in the process of reviewing information relative to these questions. We will provide a schedule for the submission of a response prior to April 30, 1979.

Q 362.5
(2.5.4)

During earthquake loading, the lateral pressures on walls below grade can approach passive pressures. Demonstrate that the lateral pressures calculated using the dynamic coefficient (K_D) are acceptably conservative by comparing them with the lateral loads you calculated using the model shown in Figure 3.7-14a of the FSAR for evaluating the structural response to dynamic loading.

RESPONSE:

The lateral earth pressures on the walls below plant grade during seismic events are calculated using the sum of the at-rest and dynamic pressure coefficients as presented in Figure 2.5-69. The value of the dynamic pressure coefficient is based on the rationale given in 4.10 of Appendix 2.5F.

The mathematical model shown in Figure 3.7-14a assumes that the foundation mat is rigid and rests on the surface of an elastic half-space as described in 3.7.2.4. The effect of the embedment of the foundation is neglected in the seismic analysis, based on the recommendation given in Reference 3.7-6.* Consequently, the lateral dynamic earth pressures on the walls are assumed to be zero in the dynamic analysis using the model. Therefore no comparison can be made with the lateral pressures calculated using the dynamic coefficient (K_D). However, use of the dynamic coefficient K_D in design of the foundation walls is considered to conservatively account for the actual dynamic loading condition, as discussed in 4.10 of Appendix 2.5F.

*Reference 3.7-6 is the same as reference 2.5-126 which was submitted to you with the response to question 362.3.

Q 362.6
(2.5.4)

The values of Poisson's ratio provided in Section 2.5.4.2 and Table 2.5D-1 of the FSAR are not consistent. Clarify this apparent discrepancy.

Response:

The values of Poisson's ratio included in Table 2.5D-1 were calculated from seismic shear wave and seismic compressional wave velocities. The calculated values range from 0.33 to 0.35 in the upper 55 feet with greater values (0.36 to 0.39) below this depth.

Poisson's ratio values provided in Section 2.5.4.2 were developed from engineering judgment based on published data (Reference 2.5-125), subsurface conditions developed from field explorations (Appendices 2.5E and 2.5F), and geophysical surveys (Appendix 2.5D).

The calculated values for the upper 55 feet represent the less dense sand encountered at the site while the values below this depth represent the very dense Ringold Formation materials. However, since the presence of groundwater (encountered at about depth 62 feet, Reference 2.5E and 2.5F) influences seismic compressional wave velocities, the calculated Poisson's ratio values below 55 feet would be affected.

Using the calculated values as a guide, an average lower Poisson ratio value of 0.3 was selected as being more appropriate for the overall very dense soils profile at the WNP-2 site. Also, a Poisson's ratio of 0.3 is closer to an average value for determining the response of the very dense materials under both static and dynamic loading conditions. Based on published data (Lamb and Whitman, 1969), granular soil during the early stages of static loading, when particle rearrangements are important, typically has Poisson's ratio values of about 0.1 to 0.2. During cyclic loading, Poisson's ratio becomes more of a constant with values from about 0.3 to 0.4. The 0.3 value selected for use in design at the WNP-2 site falls midway between the early static loading and cyclic loading range.

Q 362.7

Provide the report referred to on Page 2.5-144 of the FSAR, which contains the studies relating to the backfill under the condensate storage tanks.

Response:

The referenced report of studies performed on backfill under the condensate storage tanks was submitted as part of the response to Question 362.3.*

*See also attached draft FSAR changes for clarification.

Based on the study for the WNP-2 backfill (Ref. 2.5-127A), it was concluded that the reduced vibrating table amplitudes did not significantly alter the maximum density and that the placed fill meets specified compaction criteria.

From surveillance of contractor testing records during this study, it was discovered that valid laboratory maximum density tests may not have been performed for all field density tests taken in the backfill placed beneath the Seismic Category I Condensate Storage Tanks. ~~Studies of this situation are continuing and conclusions will be presented in a future amendment.~~

replace by

2.5.5 STABILITY OF SLOPES

There are no slopes, either natural or manmade (both cut and fill), the failure of which could adversely affect the safety of WNP-2. Therefore, 2.5.5.1 thru 2.5.5.4 do not apply.

2.5.6 EMBANKMENTS AND DAMS

There are no embankments and dams at the WNP-2 site for flood protection or for impounding cooling water required for the operation of the nuclear power plant. Therefore, 2.5.6.1 thru 2.5.6.10 do not apply.

2.5.7 GLOSSARY OF TERMS

Operating Basis Earthquake - That earthquake which could reasonably be expected to affect the plant site during the operating life of the plant (United States Atomic Energy Commission, 10 CFR, Part 100, Appendix A, November 1973).

Safe Shutdown Earthquake - "that earthquake which produces the vibratory ground motion for which structures, systems, and components important to safety are designated to remain functional" (United States Atomic Energy Commission, 10 CRF, Part 100, Appendix A, November, 1973).

Studies of this situation have been made and the conclusion reached that the test results demonstrate that adequate compaction was achieved for proper support of the condensate Storage Tanks (reference 2.5-127A).

2.5-118 Hamilton, Warren and Myers, W.B., 1966, Cenozoic Tectonics of the Western United States, in The World Rift System-- Internat. Upper Mantle Comm. Symposium, Ottawa, 1965: Canada Geol. Survey Paper 66-14, p. 291-306.

2.5-119 Peck, D.L., Griggs, A.B., Schilicker, H.G., Wells, F.G. and Dole, H.M., 1964, Geology of the Central and Northern Parts of the Western Cascade Range in Oregon: U.S. Geol. Survey, Prof. Paper 449, 56 p.

2.5-120 Portland General Electric, 1975, Preliminary Safety Analysis Report (PSAR), Vol. 2, Sec. 2.5, Pebble Springs Site: Docket Nos. 50-514 and 50-515.

2.5-121 U.S. Geological Survey, 1971, Written Communication.

2.5-122 Newcomb, R.C., 1965, Geology and Ground-water Resources of the Walla Walla River Basin, Washington-Oregon: Washington Div. Water Resources, Water Supply Bull. 21, 151 p.

2.5-123 U.S. Atomic Energy Commission, Directorate of Licensing, 1972, Safety Evaluation of the Fast Flux Test Facility.

2.5-124 Couch, R.W., 1976, Written Communication to Shannon & Wilson, May 10.

2.5-125 Shannon & Wilson, Inc., and Agbabian-Jacobsen Associates, (SW-AJA), 1972, State of the Art, Evaluation of Soil Characteristics for Seismic Response Analyses, Soil Behavior Under Earthquake Loading Conditions: National Technical Information Service, Springfield, Va., TID-26444.

2.5-126 Burns & Roe, Inc., 1972, Letter Report - Soil Structure Interaction, Washington Public Power Supply System, Hanford No. 2 Nuclear Station, Richland, Washington, by E. D'Appolonia Consulting Engineers, Inc., dated August.

2.5-127 Burns & Roe, Inc., 1976, Soil Compaction Evaluation of Quality Class I Backfill, Washington Public Power Supply System, Nuclear Project No.2, Benton County, Washington, by Shannon & Wilson, Inc., Attachment to Letter, D. L. Renberger, Washington Public Power Supply System, to R.H. Engelken, Nuclear Regulatory Commission, dated May 11, 1976: Docket No. 50-397.

2.5-127A Shannon and Wilson, Inc., "Soil Compaction Evaluation Quality Class I Backfill Condensate Storage Tanks Area WPPSS Nuclear Project No. 2 Benton County, Washington", Feb. 15, 1977.

Q 362.8
(App. 2.5H)

Provide a summary of the gradation tests performed on backfill material under all Category I structures. Indicate how many such tests were performed.

Response:

The specifications required daily sieve tests on fill material that was to be used for backfilling to verify that the gradation met specification requirements. The specifications required that all backfill (or fill) soil had to be a granular soil with a maximum particle size of 3 inches and not more than 5 percent passing the No. 200 sieve. In order to better define the grain size distribution of the backfill (or fill) soil, 32 additional samples were obtained at random locations from the stockpile area designated for placement beneath the reactor building. A range of typical gradation is presented in Figure 2.5H-2, and further information is contained in References 2.5-127 and 2.5-127A.*

*See revised FSAR page with question 362.7.

Q 362.9
(2.5.4)

Describe the method used to obtain the wet unit weight values in the Ringold Formation, as shown on Figure 2.5-64 of the FSAR. Provide a summary of the test results and identify their source.

Response:

Wet unit weights shown on Figure 2.5-64 were developed from engineering judgment based on published data (Reference 2.5-125) and subsurface explorations performed at the WNP-2 site (Appendices 2.5E and 2.5F). Substantiation that the wet unit weight of the Ringold Formation presented on Figure 2.5-64 is appropriate can be demonstrated from results of in situ density tests performed during construction. Attachment 1 presents the results of 12 density tests performed in Ringold Formation material in the bottom of the WNP-2 Containment excavation. The in situ densities were determined in accordance with procedures described in ASTM Designation: D-1556 using a large sand cone with a 12-inch diameter hole in the base plate. The test results presented in Attachment 1 indicate that the average situ wet density is 146 pcf compared to 141 pcf and 145 pcf included in Figure 2.5-64.

DENSITY TESTING OF SUBGRADE MATERIALS

Test No.	Date of Test	Test Location	Elev. feet	Field Moist. Cont. %	In-Place Density, pcf Wet	In-Place Density, pcf γ_d = Dry	Void Ratio e (Sp. G. = 2.66)
3	11-7-72	B+10 @ 0+75 (Section 1)	391.0	2.5	147.5	143.9	0.15
4	11-7-72	B+10 @ 0+25 (Section 1)	391.0	2.0	144.2	141.3	0.17
5	11-8-72	C+10 @ 0+25 (Section 1)	391.0	2.7	143.8	140.0	0.19
7	11-8-72	C+7 @ 1+35 (Section 1)	391.0	2.6	147.8	144.0	0.15
23	11-16-72	F+12 @ 0+65 (Section 2)	390.2	2.6	145.7	142.0	0.17
28	11-21-72	J+0 @ 0+30 (Section 3)	390.7	3.4	149.0	144.1	0.15
57	12-21-72	C+0 @ 1+90 (Section 4)	391.0	3.1	148.6	144.1	0.15
75	12-29-72	C+0 @ 2+50 (Section 5)	380.4	3.5	143.1	138.3	0.20
76	12-29-72	A+10 @ 3+10 (Section 5)	382.7	3.6	144.8	139.8	0.19
79	1-2-73	A+10 @ 2+40 (Section 5)	381.0	6.0	153.7	145.0	0.15
113	1-19-73	H+0 @ 2+00 (Section 6)	390.4	4.3	147.8	141.7	0.17
128	1-24-73	H+12 @ 3+25 (Section 7)	386.0	11.4	140.7	138.8	0.20

Avg. 146 pcf

TESTING:

1. Density tests conducted on rolled subgrade (as per specifications) using large sand cone with 12" diameter hole in base plate (in accord with ASTM 1556).
2. Specific Gravity (Sp. G.) of subgrade materials = 2.66 (average of 3 tests).
3. Solid (zero voids) weight of subgrade materials = 165.98 pcf (based on Sp. G. = 2.66).
4. Void ratio calculated using 165.98 value and in-place density (γ_d) values from density tests.

$$\text{Void Ratio, } e = \frac{165.98}{\gamma_d} - 1$$

REQUIREMENTS

1. CCR No. 1 - Rev. 3 (attached) states that criteria for acceptability will be void ratio (e) equal to or less than listed for comparible soils.
2. On table attached to CCR No. 1 - Rev. 3, item 5 "Glacial till, very mixed-grained" (e = 0.25) is a soil mixture of comparible density properties. The void ratio values of subgrade soils (highest e = 0.20) are all less than 0.25.

Responses to:

Previous Questions

Containment Systems Branch (22.7 and 22.048)

I & C Branch (31.076)

Power Systems Branch (40.15 and 40.26)

Reactor Systems Branch (212.003)

Q. 22.7

Provide the secondary containment pressure time response for the design basis accident. List and discuss all assumptions made in this analysis.

Response:

See the response to Question 312.18.

022.048 You state in Section 6.2.5.3.1.3 of the FSAR that the corrosion of aluminum, zinc, and zinc base paints located either in the drywell or in the suppression chamber were determined to be insignificant. However, we have determined that a potential hydrogen release from the corrosion of zinc following a postulated loss-of-coolant accident should be considered in the analysis of the total hydrogen production and accumulation within the containment. Accordingly, provide the following information:

a. Provide the corrosion rate, as a function of temperature, for all materials in the containment that could become a source of hydrogen due to corrosion.

b. Describe how the corrosion rates assumed for the materials identified in Item (a) were established. Identify the experimental data base, including the appropriate references, and discuss the conservatism in the applicability of the data in view of the calculated environmental conditions following a postulated loss of coolant accident.

c. Provide the mass and surface area of zinc paint and galvanized steel and other corrodible materials in both the drywell and the wetwell.

d. Provide a graphic representation of the total hydrogen concentration inside the containment as a function of time with: (1) no hydrogen recombiners operating; (2) one recombiner operating; and (3) both recombiners operating.

Response:

See response to Question 312.16.

Q. 031.076
(RSP)
(6.7)
(7.3.2.3)
(Q.031.019)

It is the staff's position that neither the information which is provided in response to Item 031.019 nor the information which is presented in 6.7 and 7.3.2.3 provides sufficient information on the main steamline isolation valve leakage control system. Describe this system in accordance with the guidance provided in Section 7.3 of Regulatory Guide I.70, including a process and instrumentation drawing, an electrical schematic, and a failure mode and effects analysis which is sufficiently detailed to address failures at the component level. For example, describe the consequences of a spurious closing of the contacts of relay K4 under all plant-operating modes, including testing.

RESPONSE:

In addition to FSAR sections 6.7 and 7.3.2.3 and the MSIV leakage control system instrumentation and controls is described in section 7.3.1.1.3. The system is shown diagramatically in P&ID form in FSAR figure 3.2-25 with logic diagrams shown in FSAR figures 7.3-18a-g.

Electrical schematics for the MSIV leakage control system (drawings, E519, shts 30 & 31) have previously been submitted as part of FSAR section 1.7.

A failure modes and effects analysis addressing worst-case failures and consequences concerning the safety function aspects of the MSIV Leakage Control System, i.e. following a LOCA, was submitted as part of the FSAR Section 6.7.3.1. This analysis was submitted previously to the NRC in reference 1 as a response to a post-construction permit item. In reference 2 the NRC requested additional information and WPPSS responded in reference 3. In reference 4, the NRC stated the design was acceptable subject to the provision of an interlock preventing actuation of the MSIV-LCS if the inboard MSIV were open. In reference 5 WPPSS agreed to provide this interlock. In any case, to respond to what is felt the intent of Question 031.076 is, an additional FMEA was performed which addressed failures which could occur during other plant operating modes.

The following is a description of those failures identified having undesirable consequences.

Failure Mode	Equip. Effected by Failure Mode	Undesirable Effects	Remarks/ Results
1. Spurious closing of pressure switch PS-25 contacts	MOV's MSLC-V-9, & MSLC-V-10	Both valves open simultaneously	Reactor pressure steam admitted into low pressure system piping and into the reactor building. Possible piping damage and/or plant personnel hazard.
2. Spurious closing of relay CR-3 contacts	Same as item 1 above	Same as item 1 above	Same as item 1 above
3. Spurious closing of relay CR-1 contacts	MOV's MSLC-V-4, & MSLC-V-5	Same as item 1 above	Same as item 1 above

In order to prevent the events described in items 1, 2 and 3 from occurring the logic design will be modified. An additional interlock will be added in series with the contacts of pressure switch PS-25, relay CR-3, and relay CR-1. The interlock will be provided from the system initiation control switch. Thus, two active component failures would be required to cause a similar occurrence.

In addition it was noted from the results of FMEA that events similar to those described in items 1, 2, and 3 above might occur from localized events in control panels and wireways. This is due to inherent system designs requiring the several pairs of series motor operated valves to be controlled from a common safety division to meet single failure criteria. To preclude such situations certain key control devices and wiring associated with each valve of a series pair will be separated from the other. This will prevent localized events within a control panel, instrument rack, wireway, or motor control center from causing simultaneous opening of both valves during normal operating modes.

The MSIV leakage control system does not contain a relay K4.

References

1. WPPSS to NRC letter, G02-74-73, "Post Construction Permit Item - Transmittal of Report WPPSS 74-2-RG, Concept for Main Steam Isolation Valve Leakage Control System", dated Dec. 3, 1974.

2. NRC to WPPSS letter, Butler to Stein, dated March 18, 1975.

3. WPPSS to NRC letter, G02-75-238, "Response to Request for Information - MSIV-CCS", dated Aug. 18, 1975.

4. NRC to WPPSS letter, Parr to Stein, dated November 21, 1975.

5. WPPSS to NRC letter, G02-76-294, "MSIV-CCS", dated July 14, 1976.

Q. 40.15

You state in 9.5.4.2.2 of the FSAR that the diesel generator fuel oil storage tank is provided with an outside fill and vent line. Indicate how these lines are protected from tornado and turbine missiles. Indicate the height at which these lines are terminated above plant grade and describe the measures to prevent entry of moisture into the storage tanks during adverse environmental conditions (e.g., high humidity and/or heaby precipitation).

RESPONSE:

Refer to 9.5.4.2 of the FSAR. The fuel oil storage tanks are provided with individual fill and vent lines which are protected against the entry of contaminants but are not missile protected. The fill lines are provided with screwed caps and the vent lines are provided with flame arrestors. The flame arrestors vent air from the underside such that the metal top of the arrestor prevents direct entry of moisture into the tank. The fill and vent lines terminate at 3.25 and 6.0 feet respectively above plant grade. In additon, the fuel oil is sampled periodically to detect water or contamination in the fuel oil before it could present a problem. Missile protection is not necessary since, in the unlikely event that a vent or fill line is ruptured due to a missile and in the extremely unlikely event sufficient contamination would enter the ruptured line to perturb diesel operation, fuel oil suction would be switched to the other tank. Either diesel may be supplied from either tank.

Q 40.26

Provide the results of an analysis demonstrating that the function of the diesel engine air intake system will not be degraded to an extent which prevents the diesel from developing its full rated power or cause shutdown of the diesel as a consequence of any meteorological or accident condition. Include in this discussion, the potential for the following materials being drawn into the diesel air intake: (a) gaseous (e.g., carbon dioxide) fire extinguishing materials; (b) recirculation of diesel combustion products; (c) accidental release of gases transported or stored in the vicinity of the diesel intakes, and (d) airborne dust. Discuss any potential restriction of the airflow to the diesel intakes. Discuss the effects of these phenonena.

RESPONSE:

a) The potential for oxygen depletion resulting from CO_2 generated by a fire and/or fire extinguisher was not considered to be significant in comparison to that resulting from diesel operation.

b) For the diesel generator operation and emission data, and the meteorological conditions assured, the maximum reduction of oxygen for incoming combustion air was calculated to be 2.3%. Under these circumstances, there would be little (0.5 percent) to no effect upon engine performance. Furthermore, the engine could be expected to operate indefinitely in this mode.

c) The only gas which could be released on site is chlorine. The accident considered was spillage of 2,000 pounds of liquid chlorine in the area of the circulating water pump house. The calculated peak chlorine gas concentration and average chlorine gas concentration at the diesel generator intakes are 17,649 ppm and 2,885 ppm. The chlorine gas plume would affect the diesel intake area for about 15 minutes. The performance of the diesel would be similar to that stated under (b) above, although ingestion of chlorine could be expected to result in accelerated wear on piston rings and cylinders.

d) For the postulated design worst case dust storm which lasts 18 hours and possesses an average airborn dust concentration of 8.9 mg/m^3, the dust mass which passes the diesel generator building intake louvers after particle impaction and re-entrainment is acounted for, is calculated to be 6.44 mg/m^3. The results of an analysis of the oil bath filters for the intake air system show the adequacy of these panel bath filters in handling a severe dust storm without affecting the diesel generators performance.

QUESTION 212.003 (6.3)

Your discussion of single failure does not adequately address ECCS passive failures during long-term cooling. Accordingly, provide a response to the attached Reactor Systems Branch Technical Position regarding the leak detection requirements for passive failures in the ECCS piping.

REACTOR SYSTEMS BRANCH TECHNICAL POSITION

Leak Detection Requirements for ECCS Passive Failures

The passive failures to be considered are limited to leaks from valve stem packing and pump seals. The sum of these leak rates may range from essentially no leakage up to the equivalent of the sudden failure of the seal of the largest ECCS pump (e.g., about 50 gpm). It is the staff's position that detection and alarms be provided to alert the operator of passive ECCS failures during long-term cooling. The timing of these alarms should be such that the reactor operator has sufficient time to identify and isolate the faulted ECCS line. Provide the following information regarding the ECCS leak detection system:

a. An identification and justification of the maximum leak rate;

b. The maximum allowable time for operator action, including a justification of the time interval;

c. A demonstration that the leak detection system will be sensitive enough to provide an alarm to the operator, subsequent identification by the operator of the faulted line, and, finally, permit the operator to isolate the faulted line prior to the leak creating any undesirable consequences such as flooding of redundant equipment. The minimum time to be considered for this sequence of events is 30 minutes.

d. A demonstration that the leak detection system can identify the faulted ECCS train and that the leak is isolable.

Additionally, the ECCS leak detection system must meet the following standards: (1) control room alarm; and (2) IEEE-279, except single failure requirements.

Response:

(a) The ECCS are capable of withstanding passive failures of valve stem packings and pump seals following a LOCA. The maximum leakage due to a failure of this nature could be as high as 23 gpm from an HPCS, LPCS or RHR pump seal failure. Valve stem leakage would be significantly less than this.

(b) The maximum allowable time for operator action is determined as the shorter of the time required to flood to the level of an ECCS pump motor in the secondary containment, or the time required to drain the suppression pool to a level below that required ECCS pump NPSH. The maximum NPSH required for any ECCS pump is 21 ft. (HPCS). With a minimum NPSH available of 36 ft., calculated in accordance with Regulatory Guide 1.1, and a leakage rate of 23 gpm, there is about 15 days of operator time available before NPSH becomes a problem. A Class IE level instrument will be installed in each ECCS pump room and it will be mounted just above floor level. After the operator receives an alarm in the control room, there is at least 44 hours of operator time available before the water level reaches the bottom of an ECCS pump, assuming a 23 gpm leak rate into the smallest ECCS pump room (RHR C).

(c) The sensitivity of the leak detection system is not vital to the identification and subsequent isolation of the faulted line prior to any undesirable consequences with at least 44 hours available.

(d) Any ECCS leak can be isolated, including any packing failure on any ECCS pump suction valve. This packing can be isolated by closing the valve since the valves are double-seat, wedge knife gates. With a Class IE level instrument in each ECCS pump room, there is no problem with identification of the faulted ECCS train.

The leak detection system will have a control room alarm and meet IEEE 279, except single failure requirements.