UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

April 10, 1992

NOTE TO: Jack Duncan, GE

FROM: Glenn Kelly, PRAB, DREP, NRR Glenn Kelly

SUBJECT: INITIAL CONCERNS RAISED BY ACRS ABOUT RWCU AND LOCAS OUTSIDE OF CONTAINMENT

As we discussed on the phone today, I am sending you a copy of early thoughts generated by the ACRS about some problems with how you took credit for the RWCU and FW systems in high pressure sequences in the ABWR PRA. There is also a concern about your evaluation of LOCAs outside of containment. We are just beginning to focus on these new issues and will be in discussion with you about them shortly.

Enclosure: as stated

RWCU PRA ISSUES FOR ABWR

① HEAT REMOVAL FOR HIGH PRES. SEQUENCES
② INITIATOR FOR LARGE LOCA OUTSIDE CONT.

① HEAT REMOVAL

- No fault tree in SSAR
- Credit taken as 0.1 for HP sequences (see Table 19.D.4-2) (all seq with W_1 heading).
  - only RHX bypass needed in write-up. this ignores need for resin bed isolation, temperature effects due to 500° to 95° ΔT on NRHX tubes, exit temp for RWCU and RBCCW system piping exceeding design limits, and isolation feature of TS between NHRX and pumps.
- Credit also taken in W_1 for NHR (normal heat removal via feedwater & condenser) at 0.01.
  - ignores fact that Q (at 0.05) already failed in sequences with W1 (Q includes FW).
  (also applies to seq. w/ W_2 and W_7)

Changes if credit not taken for RWCU or NHR

Figure	IE	OLD SEQ	NEW EQ
19D4-1	Rx SD	2.74 E-8 (W_1)	2.74 E-5
	"	1.72 E-11 (W_2)	1.72 E-9
19D4-2	Non Isol	1.76 E-8 (W_1)	1.76 E-5
	"	1.10 E-11 (W_2)	1.10 E-9
19D4-3	LOFW	4.45 E-8 (W_1)	4.45 E-5
	"	2.78 E-11 (W_2)	2.78 E-9
19D4-5	LOSP	1.27 E-8 (W_1)	1.27 E-7 *
		5.53 E-11 (W_1)	5.53 E-10 *
19D4-6	LOSP	8.11 E-9 (W_1)	8.11 E-8 *
		3.54 E-11 (W_1)	3.54 E-10 *
19D4-11	IORV	2.59 E-9 (W_7)	2.59 E-7
		3.10 E-12 (W_2)	3.10 E-10
19D4-12	Sm LOCA	6.22 E-10 (W_1)	6.22 E-9 *
		~~7.44 E-12 (W_2)~~	~~7.44 E-10~~
~~19D4-13~~	~~Med LOCA~~	~~3~~	
	TOTALS	1.1 E-7	8.9 E-5

a factor of 809

* LEFT IN CREDIT FOR NHR SINCE Q NOT ON THESE SEQUENCES

② INITIATOR FOR LARGE LOCA OUTSIDE CONT.

Based on ratio of hypothetical consequences in section 19E.2.3.3

There are many errors here, but the biggest is in the last ¶ of section 19E2.3.3.1. It fails to take into account that the cause of the cores melt scenario may have been a failure ~~___~~ in the bypass line (such as RWCU or FW or RCIC steam etc).

The flow fraction f_i ratios flow from a line to the aerosol generation rate. To properly use the ratio of consequences argument it should be ratioed with the leakage rate from other non-bypass paths. This factor artificially reduces the prob. numbers between 1 to 5 orders of magnitude

Formula in equation 7 is wrong (GE sent FAX with corrected version)

$$\sum_i P_{bp_i} \times f_i \ll \frac{C_{nbp}}{C_{bp}} \quad \left(\text{not } \frac{C_{nbp}}{C_{nbp}+C_{bp}}\right)$$

GE Nuclear Energy

ABWR

Date 4/3/92

To Chet Poslusny

Fax No.

This page plus 3 page(s)

From Jack Fox

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Subject Control Rod Criteria

Message "Internal" response to your letter dated January 28, 1992. Myself, Bob Huang and Paul vanDiemen will be participating in the 1PM EST conference call (my office) on Monday 4/6/92. If possible we might also cover Paul's recommended change.

Jack

GE NUCLEAR ENERGY
San Jose, California

March 12, 1992

cc: AJ James
JE Wood
PVD:92013

To: J.N. Fox
G.G. Jones

SUBJECT: SSAR Control Rod Design Criteria (ABWR)

Reference: C. Poslusny to P. Marriott, Control Rod Design Criteria, Docket No. 50-605, January 28, 1991

The Reference letter provides recommended changes to the control rod design criteria contained in Section 4C.2 of 23A6100AB Rev. C. The recommended wording changes in 4C.1 (Introduction) and the first sentence in 4C.2 are acceptable. While we accept the substance of the NRC recommended changes in the 4 acceptance criteria, we believe the recommended changes are conditions which may be more appropriate to include in 4C.3 (Basis For Acceptance Criteria).

Regardless of where the additional wording is located, any reference to sheath material should be avoided since this is a feature not common to all GE control rod designs. Instead, the wording should identify the phenomena of concern, i.e., crudding, crevices and stress corrosion, and not the piece parts which were affected in older designs.

A fifth criteria (Surveillance) has been added per the Reference letter along with the corresponding Acceptance Criteria, 4C.3.5.

The GE recommended wording changes for Section 4C is attached.

Please provide an appropriate response to the reference letter.

Paul van Diemen
Fuel Design and Development
M/C 148, ext. 56160

4C.2 GENERAL CRITERIA

(5) A surveillance program shall be implemented if a change in design features such as new absorber material or structural material not previously used in reactor cores could impact the function of the control rod.

4C.3.5 SURVEILLANCE PROGRAM

Visual inspection of the lead depletion control rod design possessing the new design feature and three additional control rods of such design that are within 15% of the estimated fast fluence of the lead control rod shall be performed. If fewer than three control rods are within 15% of the estimated fast fluence of the lead control rod, only those within 15% shall be inspected. Should evidence of a problem arise, arrangements will be made to inspect additional control rods to the extent necessary to identify the root cause of the problem.

4C.1 INTRODUCTION

A set of acceptance criteria has been established for evaluating new control rod designs. Control rod compliance with these criteria constitutes THE BASIS FOR USNRC acceptance and approval of the design~~, without specific USNRC review~~. The control rod licensing acceptance criteria and their bases are provided below. Control rod designs which have been approved by the USNRC or which meet the licensing acceptance criteria are documented in Reference 4C-4.

4C.2 GENERAL CRITERIA

Control rod designs MUST ~~meeting~~ the following acceptance criteria ~~are considered to be approved and do not require specific USNRC review~~.

(1) The control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material.

(2) The control rod shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.

(3) The material of the control rod shall be shown to be compatible with the reactor environment.

(4) The reactivity worth of the control rod shall be included in the plant core analyses.

(5) SEE ATTACHED

4C.3 BASIS FOR ACCEPTANCE CRITERIA

The following are the basis for the licensing acceptance criteria given in Section 4C.2.

4C.3.1 Stress, Strain and Fatigue

The control rod is evaluated to assure that it does not fail because of loads due to shipping, handling, and normal, abnormal, emergency, and faulted operating modes. To assure that the control rod does not fail, these loads must not exceed the ultimate stress and strain limit of the material. Fatigue must not exceed a fatigue usage factor of 1.0 INCLUDING IRRADIATION EFFECTS FOR ITS DESIGN LIFE.

The loads evaluated include those due to normal operational transients (scram and jogging), pressure differentials, thermal gradients, flow and system induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time as appropriate.

Conservatism is included in the analyses by including margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

4C.3.2 Control Rod Insertion

The control rod is evaluated to be sure that it can be inserted during normal, abnormal, emergency and faulted modes of operation within the limits assumed in the plant analyses AND THROUGHOUT ITS DESIGN LIFE. These evaluations include a combination of analyses of the geometrical clearance and actual testing. The analyses consider the effects of manufacturing tolerances, swelling and irradiating growth. Tests may be performed to demonstrate control rod insertion capability for conditions such as control rod or fuel channel deformation and vibrations due to safe shutdown earthquakes.

4C.3.3 Control Rod Material

The external control rod materials must be capable of withstanding the reactor coolant environment for the DESIGN life of the control rod. ~~Irradiation~~ Effects OF CRUDDING, CREVICES, STRESS CORROSION AND IRRADIATION upon the material must be included in the control rod and core evaluations. Irradiation effects to be considered include material hardening and absorber depletion and swelling.

4C.3.4 Reactivity

The reactivity worth of the control rod is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance and control rod drop times also effects must be included in the plant core analyses including nuclear, abnormal operational occurrences, infrequent events, and accidents.

4C.4 REFERENCES

1. *GE Control Rod Designs*, (to be issued).

4C.3.5 SEE ATTACHED

GE Nuclear Energy

ABWR

Date 3-16-92

To JAY Lee 504-1080
m/s 10D4

Fax No. ______

This page plus 9 page(s)

From H A CAREWAY

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Subject ABWR LOCA DOSE

Message Still owe you ECCS Leakage
Hopefully today

How Dose and X/Q is Evaluated

Three separate file print outs are provided:

1. Dose_sum.prn
2. Iodinsum.prn
3. Nblgssum.prn

1. "Dose_sum.prn" provides a picture of the individual results for offsite calculations on page one. Page two shows the control room calculation for the control room calculations and page three shows the summary for offsite doses. Abbreviations are:

Thy	Thyroid dose (rem)
WB	Whole body dose (rem)
NG	Noble gases
EL	Elemental and Particulate Iodines
OR	Organic Iodines
RESP	Organic iodines resuspended from MSIV leakage.

The results of the control room dose evaluation are passed onto the two other spreadsheets.

2. "Iodinsum.prn" performs the iodine control room calculation. On page one, the control room activity and integrated activities are shown for both pathways (all species are summed). Then the control room activity is calculated at the top of page 2 by using the following rules:

0.3 * Activity due to Reactor Building + 0.3 * Activity due to MSIV pathway

+

0.7 * Larger of either (Reactor Building or MSIV) based upon the time integrated Activity.

The same thing is done under Dose evaluation to get the Total column.

Then the doses are reduced by a factor of four (Reduction factor).

Then I note that up to this time the control room doses are based upon a 0-8hr chiqu of 0.01 (Reactor Building) and the total dose is 73.9. So the next table changes the chiqu to 0.004 which brings the dose down to 29.5.

Finally the last table on the page is the activity table redone with the reduction factor and the new chiqu.

3. "Nblgssum.prn" performs the same evaluation noble gases as "Iodinsum.prn" does for iodines.

Summary of Dose Results

Offsite - Reactor Building

2hr	Dist	Time	X/Q	Thy	WB
	300	2	1.18E-03	128	2.28
	800	2	2.19E-04	23.8	0.423

30 day	Dist	Time	X/Q	Thy	WB
	3219	0-8	5.61E-05	7.48	0.114
		8-24	2.22E-05	9.31	0.12
		1-4	7.87E-06	16.3	0.129
		4-30	1.7E-06	21.8	0.134
	4828	0-8	3.73E-05	4.97	0.0761
		8-24	1.21E-05	5.98	0.0794
		1-4	4.27E-06	9.74	0.0842
		4-30	9.09E-07	12.7	0.0867

++

Offsite - MSIV Leakage

2hr	Dist	Time	X/Q	Species	Thy	WB
	300	2	1.18E-03	EL	0.001562	6.02E-05
				NG	0	0.01221
				OR	0.1888	0.000728
				RESP	0.000424	1.64E-06
				Total	0.190786	0.013

30 day	Dist	Time	X/Q	Species	Thy	WB
	4828	0-8	3.73E-05	EL	0.03241	6.34E-05
				NG	0	0.01068
				OR	0.3927	0.000768
				RESP	0.000706	1.38E-06
		8-24	1.21E-05	EL	0.09507	0.000182
				NG	0	0.02334
				OR	1.162	0.002226
				RESP	0.00088	7.51E-07
		1-4	4.27E-06	EL	0.3896	0.000362
				NG	0	0.06165
				OR	5.816	0.004964
				RESP	0.052391	3.04E-05
		4-30	9.09E-07	EL	0.5132	0.00041
				NG	0	0.1041
				OR	13.71	0.007961
				RESP	3.168819	0.085678

Dist	Time	X/Q	Totals	Thy	WB
4828	0-8	3.73E-05		0.425816	0.011513
	8-24	1.21E-05		1.25795	0.025749
	1-4	4.27E-06		6.257991	0.067006
	4-30	9.09E-07		17.39202	0.198149

Monday March 16, 1992 09:57 d:\abwr\ssar\chp15\loca\dose_sum.prn Page: 1

Control Room - Reactor Building

Time	X/Q	Species	Thy	WB	Beta
0-8	0.01	Iodine	40.42	0.03183	0.04096
		NG	0	3.873	8.337
8-24	0.0059	Iodine	70.48	0.03746	0.0518
		NG	0	9.42	24.87
1-4	0.00375	Iodine	165	0.04505	0.06633
		NG	0	23.23	68.29
		Iodine	268.4	0.05031	0.07479
4-30	0.00165	NG	0	33.97	102.9

Time	X/Q	Totals	Thy	WB	Beta
0-8	1.00E-02		40.42	3.90483	8.37796
8-24	5.90E-03		70.48	9.45746	24.9218
1-4	3.75E-03		165	23.27505	68.35633
4-30	1.65E-03		268.4	34.02031	102.9748

++

Control Room MSIV

Time	X/Q	Species	Thy	WB	Beta
0-8	1.67E-03	EL	0.03462	1.23E-05	2.21E-05
		NG	0	0.05892	0.1277
		OR	0.4194	0.000149	0.000245
		RESP	0.000778	2.75E-07	4.55E-07
8-24	9.83E-04	EL	0.341	6.8E-05	0.000128
		NG	0	0.3174	0.9079
		OR	4.164	0.000831	0.001567
		RESP	0.013142	2.62E-06	4.95E-06
1-4	6.25E-04	EL	1.572	0.000167	0.000317
		NG	0	1.412	4.344
		OR	23.5	0.00233	0.004413
		RESP	0.773908	7.67E-05	0.000145
4-30	2.75E-04	EL	2.295	0.000204	0.000378
		NG	0	3.142	10.1
		OR	69.24	0.004639	0.008114
		RESP	18.76018	0.001257	0.002198

Time	X/Q	Totals	Thy	WB	Beta
0-8	0.00167		0.454798	0.059081	0.127968
8-24	0.000983		4.518142	0.318301	0.9096
1-4	0.000625		25.84591	1.414574	4.348876
4-30	0.000275		90.29518	3.14811	10.11069

Offsite Dose Summary

2hr Site Boundary

Dist	X/Q	Thy	WB
max	2.76E-03	300	5.37E+00
300	1.18E-03	1.28E+02	2.29E+00
800	2.19E-04	2.38E+01	4.26E-01

30 day LPZ

Dist	Time	X/Q	Thy	WB
[illegible]	0-8	3.73E-05	5.395816	0.087613
	8-24	1.21E-05	7.23795	0.105149
	1-4	4.27E-06	15.99799	0.151206
	4-30	9.09E-07	30.09202	0.284849

Worksheet to evaluate Control Room Dose and Inventory
Iodines

Workspace
D:\ABWR\SSAR\CHP15\LOCA\CNTRLRM\IODINSUM.WQ1 0 2 77 24
D:\ABWR\SSAR\CHP15\LOCA\DOSE_SUM.WQ1 0 2 77 24 Z
D:\ABWR\SSAR\CHP15\LOCA\CNTRLRM\ICNDSRCR.WQ1 0 2 77 24
D:\ABWR\SSAR\CHP15\LOCA\CNTRLRM\I_RB_CB.WQ1 0 2 77 24

Reactor Building Control Room Activity

Time	1	2	8	24	96	720
Isotope						
I-131	0.059939	0.020809	0.007151	0.011066	0.01198	0.000477
I-132	0.06491	0.016698	0.00095	1.21E-05	5.57E-15	0
I-133	0.121674	0.041004	0.011789	0.011336	0.001443	5.03E-13
I-134	0.062806	0.009926	3.04E-05	1.6E-10	0	0
I-135	0.10691	0.033528	0.00626	0.001904	1.36E-06	0

Reactor Building Integrated Control Room Activity

Time	1	2	8	24	96	720
Isotope						
I-131	266.1666	132.188	154.5776	465.283	2799.45	4884.36
I-132	330.7053	126.925	67.2149	10.1622	0.112376	3.73E-11
I-133	547.608	265.069	282.798	592.684	1098.67	74.3926
I-134	404.097	102.476	22.99742	0.118184	5.92E-07	0
I-135	497.14	226.172	195.7671	182.276	54.331	0.023257
	2045.717	852.83	723.355	1250.523	3952.563	4958.776
	2045.717	2898.547	3621.902	4872.425	8824.989	13783.76

MSIV Leakage Control Room Activity

Time	1	2	8	24	96	720
Isotope						
I-131	1.92E-06	2.6E-05	0.000672	0.001741	0.00344	0.000518
I-132	2.08E-06	2.09E-05	8.94E-05	1.93E-06	1.51E-15	0
I-133	3.89E-06	5.13E-05	0.001108	0.001784	0.000415	1.52E-12
I-134	2.01E-06	1.24E-05	2.87E-06	3.46E-11	0	0
I-135	3.42E-06	4.19E-05	0.000588	0.0003	4.13E-07	3.78E-40

MSIV Leakage Integrated Control Room Activity

Time	1	2	8	24	96	720
Isotope						
I-131	0.001087	0.040689	6.465974	63.10765	635.5321	3048.114
I-132	0.001223	0.036049	1.666858	1.157434	0.018572	1.03E-11
I-133	0.002215	0.080948	11.28539	79.01762	226.115	25.87513
I-134	0.00126	0.025488	0.247639	0.012076	9.52E-08	0
I-135	0.001964	0.067754	6.932217	23.23532	9.681106	0.007505
	0.007749	0.250929	26.59807	166.5301	871.3468	3073.997
	0.007749	0.258678	26.85675	193.3869	1064.734	4138.73

Total	Control Room Activity					
Time	1	2	8	24	96	720
Isotope						
I-131	5.99E-02	2.08E-02	7.35E-03	1.16E-02	1.30E-02	6.33E-04
I-132	6.49E-02	1.67E-02	9.77E-04	1.27E-05	6.03E-15	0
I-133	1.22E-01	4.10E-02	1.21E-02	1.19E-02	1.57E-03	9.59E-13
I-134	6.28E-02	9.93E-03	3.12E-05	1.70E-10	0	0
I-135	1.07E-01	3.35E-02	6.44E-03	1.99E-03	1.49E-06	1.13E-40
Total	4.16E-01	1.22E-01	2.69E-02	2.55E-02	1.46E-02	6.33E-04

Dose Evaluation

Time	RB	MSIV	Total
0-8	40.42	0.454798	4.06E+01
8-24	70.48	4.518142	7.18E+01
1-4	165	25.84591	1.73E+02
4-30	268.4	90.29518	2.95E+02

Reduction Factor 4

Time	
0-8	1.01E+01
8-24	1.80E+01
1-4	4.32E+01
4-30	7.39E+01

Chiqu Factor 0.004

Time	
0-8	4.06E+00
8-24	7.18E+00
1-4	1.73E+01
4-30	2.95E+01

Iodine Control Room Activity

Total	Control Room Activity					
Time	1	2	8	24	96	720
Isotope						
I-131	5.99E-03	2.08E-03	7.35E-04	1.16E-03	1.30E-03	6.33E-05
I-132	6.49E-03	1.67E-03	9.77E-05	1.27E-06	6.03E-16	0.00E+00
I-133	1.22E-02	4.10E-03	1.21E-03	1.19E-03	1.57E-04	9.59E-14
I-134	6.28E-03	9.93E-04	3.12E-06	1.70E-11	0.00E+00	0.00E+00
I-135	1.07E-02	3.35E-03	6.44E-04	1.99E-04	1.49E-07	1.13E-41
Total	4.16E-02	1.22E-02	2.69E-03	2.55E-03	1.46E-03	6.33E-05

This is a combine worksheet to evaluate noble gas inventory and dose in the control room.

Spreadsheets
Workspace
D:\ABWR\SSAR\CHP15\LOCA\CNTRLRM\NBLGSSUM.WQ1 0 2 77 24 Z
D:\ABWR\SSAR\CHP15\LOCA\DOSE_SUM.WQ1 0 2 77 24 Z
D:\ABWR\SSAR\CHP15\LOCA\CNTRLRM\NGCNDRCR.WQ1 0 2 77 24 Z
D:\ABWR\SSAR\CHP15\LOCA\CNTRLRM\NG_RB_CR.WQ1 0 2 77 24 Z

From Reactor Building
Control Room Integrated Activity

Time	1	2	8	24	96	720
Isotope						
KR83M	4747.121	3391.89	12222.68	3002.08	11.573	2.91E-11
KR85	564.0642	566.21	8001.45	44501.4	321496	1519420
KR85M	11504.95	9994.66	75552.8	87586.7	10293.8	0.112052
KR87	17756.22	10842.8	23512.7	2045.45	0.510595	0
KR88	29707.56	23776.2	129748.5	73997.7	2023.63	3.69E-05
KR89	513.3517	0.006977	1.4E-08	0	0	0
XE131M	295.5586	296.03	4141.727	22414.9	144898	338368
XE133	103094.2	102968	1422356	7435120	41037400	49397800
XE133M	4280.531	4246.7	56908.3	274030	1124200	465391
XE135	12803.27	11991	124523.2	308595	177419	618.355
XE135M	5059.13	491.452	40.99145	1.73E-05	0	0
XE137	1754.086	0.160728	2.93E-06	0	0	0
XE138	19800.05	1515.3	92.59022	7.38E-06	0	0
Total	211880.1	170080.4	1857101	8251293	43617743	51921797

From Reactor Building
Control Room Inventory

Time	1	2	8	24	96	720
Isotope						
KR83M	1.20278	0.786897	0.331054	0.001468	4.55E-15	0
KR85	0.167853	0.159453	0.628698	1.08843	1.5502	0.579431
KR85M	3.20109	2.59648	3.96727	0.548249	8.96E-06	0
KR87	4.16311	2.28894	0.339255	9.31E-05	1.06E-21	0
KR88	7.9504	5.89372	5.2478	0.171822	4.31E-09	0
KR89	2.59E-05	5.03E-11	0	0	0	0
XE131M	0.087863	0.083266	0.323604	0.539112	0.645864	0.053911
XE133	30.6079	28.9175	110.335	175.004	168.087	2.06669
XE133M	1.26694	1.18824	4.33884	6.1211	3.47016	0.000442
XE135	3.58986	3.24966	8.13559	4.19495	0.025657	0
XE135M	0.409873	0.02751	1.35E-08	9.01E-27	0	0
XE137	0.000498	8.73E-09	1.35E-36	0	0	0
XE138	1.3699	0.069461	6.33E-09	0	0	0

From MSIV Leakage

Control Room Integrated Activity

Time	1	2	8	24	96	720
Isotope						
KR83M	0.181357	5.37911	227.4536	116.3359	0.675431	2.84E-12
KR85	0.024154	0.963905	188.5193	2142.967	25961.06	301406.8
KR85M	0.469763	16.5179	1628.542	3903.502	628.6702	0.011498
KR87	0.641859	16.6191	383.4696	75.43388	0.029546	0
KR88	1.17997	38.6261	2638.087	3039.946	120.036	3.76E-06
KR89	6.3E-05	2.65E-06	6.22E-11	0	0	0
XE131M	0.012647	0.503731	97.46362	1077.877	11619.59	61120.06
XE133	4.40746	175.113	33418.75	356892	3324755	7981393
XE133M	0.182601	7.21233	1332.084	13096.38	87379.09	62658.63
XE135	0.53597	20.127	2817.812	14166.31	11732.17	65.3903
XE135M	0.084993	0.489633	0.271828	5.86E-07	0	0
XE137	0.000619	6.59E-05	1.33E-08	0	0	0
XE138	0.296727	1.42842	0.587421	2.49E-07	0	0
Total	8.018183	282.9803	42733.04	394410.7	3462196	8406644

From MSIV Leakage

Control Room Inventory

Time	1	2	8	24	96	720
Isotope						
KR83M	0.000309	0.003121	0.010854	8.29E-05	4.6E-16	0
KR85	4.31E-05	0.000632	0.020579	0.061121	0.15333	0.144784
KR85M	0.000821	0.010298	0.129949	0.030864	8.94E-07	0
KR87	0.001068	0.009078	0.011131	5.28E-06	9.43E-23	0
KR88	0.00204	0.023375	0.171961	0.009687	4.32E-10	0
KR89	6.65E-09	2E-13	0	0	0	0
XE131M	2.25E-05	0.00033	0.010593	0.030275	0.063891	0.013471
XE133	0.007855	0.114688	3.61167	9.8282	16.6308	0.51642
XE133M	0.000325	0.004713	0.142031	0.3438	0.343493	0.000111
XE135	0.000947	0.012888	0.26639	0.23585	0.002549	7.33E-24
XE135M	0.000145	0.000109	4.47E-10	0	0	0
XE137	1.28E-07	3.46E-11	0	0	0	0
XE138	0.000352	0.000275	2.1E-10	0	0	0

Summary

Control Room Inventory

Time	1	2	8	24	96	720
Isotope						
KR83M	1.20E+00	7.88E-01	3.34E-01	1.49E-03	4.69E-15	0
KR85	1.68E-01	1.60E-01	6.35E-01	1.11E+00	1.60E+00	6.23E-01
KR85M	3.20E+00	2.60E+00	4.01E+00	5.58E-01	9.22E-06	0
KR87	4.16E+00	2.29E+00	3.43E-01	9.47E-05	1.08E-21	0
KR88	7.95E+00	5.90E+00	5.30E+00	1.75E-01	4.43E-09	0
KR89	2.59E-05	5.04E-11	0	0	0	0
XE131M	8.79E-02	8.34E-02	3.27E-01	5.48E-01	6.65E-01	5.80E-02
XE133	3.06E+01	2.90E+01	1.11E+02	1.78E+02	1.73E+02	2.22E+00
XE133M	1.27E+00	1.19E+00	4.38E+00	6.22E+00	3.57E+00	4.75E-04
XE135	3.69E+00	3.25E+00	8.22E+00	4.27E+00	2.64E-02	2.20E-24
XE135M	4.10E-01	2.75E-02	1.36E-08	9.01E-27	0	0
XE137	4.98E-04	8.74E-09	1.35E-36	0	0	0
XE138	1.37E+00	6.95E-02	6.40E-09	0	0	0
Total	5.41E+01	4.53E+01	1.35E+02	1.91E+02	1.79E+02	2.90E+00

Dose Evaluation

Time	Reactor Building WB	Reactor Building Beta	MSIV Leakage WB	MSIV Leakage Beta	Total WB	Total Beta
0-8	3.90483	8.37796	0.059081	0.127968	3.927554	8.41635
8-24	9.45746	24.9218	0.318301	0.9096	9.55295	25.19468
1-4	23.27505	68.35633	1.414574	4.348876	23.69942	69.66099
4-30	34.02031	102.9748	3.1481	10.11069	34.96474	106.008

Credit for Intake Vents

4	WB	Beta
0-8	0.980639	2.104088
8-24	2.388238	6.29867
1-4	5.924856	17.41525
4-30	8.741185	26.502

Chiqu Variation

0.004	WB	Beta
0-8	3.92E-01	8.42E-01
8-24	9.55E-01	2.52E+00
1-4	2.37E+00	6.97E+00
4-30	3.50E+00	1.06E+01

Summary

Control Room Inventory

Time	1	2	8	24	96	720
Isotope						
KR83M	1.20E-01	7.88E-02	3.34E-02	1.49E-04	4.69E-16	0.00E+00
KR85	1.68E-02	1.60E-02	6.35E-02	1.11E-01	1.60E-01	6.23E-02
KR85M	3.20E-01	2.60E-01	4.01E-01	5.53E-02	9.22E-07	0.00E+00
KR87	4.16E-01	2.29E-01	3.43E-02	9.47E-06	1.08E-22	0.00E+00
KR88	7.95E-01	5.90E-01	5.30E-01	1.75E-02	4.43E-10	0.00E+00
KR89	2.59E-06	5.04E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE131M	8.79E-03	8.34E-03	3.27E-02	5.48E-02	6.65E-02	5.80E-03
XE133	3.06E+00	2.90E+00	1.11E+01	1.78E+01	1.73E+01	2.22E-01
XE133M	1.27E-01	1.19E-01	4.38E-01	6.22E-01	3.57E-01	4.75E-05
XE135	3.69E-01	3.25E-01	8.22E-01	4.27E-01	2.64E-03	2.20E-25
XE135M	4.10E-02	2.75E-03	1.36E-09	9.01E-28	0.00E+00	0.00E+00
XE137	4.98E-05	8.74E-10	1.35E-37	0.00E+00	0.00E+00	0.00E+00
XE138	1.37E-01	6.95E-03	6.40E-10	0.00E+00	0.00E+00	0.00E+00
Total	5.41E+00	4.53E+00	1.35E+01	1.91E+01	1.79E+01	2.90E-01

GE Nuclear Energy

ABWR

Date 3-12-92

Fax No.

To

JAY LEE, phone 504-1080
M/S 10 D4

This page plus 4 page(s)

From H A CAREWAY

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Subject RAMSTELL X/Q EVALUATION

Message STILL GETTING STUFF together for other Two items. WILL FAX LATTER.

Evaluation of Ramsdell Chi/Q

1. To determine the Ramsdell χ/Q for a given wind speed and stability, recourse was made to a spreadsheet (see following page) which calculated χ/Q from Ramsdell equation (9) from the 21st DOE/NRC Nuclear Air Cleaning Confrence Paper.

$$\chi/Q = 1 / (F_o + \pi U \Sigma_{wy} \Sigma_{wz})$$

From the spread sheet it is shown that $F_o = 0$ for these calculations. Therefore only Σ_{wy} and Σ_{wz} needed to be calculated for each combination of speed and stability.

2. To calculate these sigma values, standard σ_y and σ_z need to be calculated. Recourse was made to the Pavan code subroutine POLYN (see sheet 2) from which the coding was pulled and a table created in the spreadsheet so that for a given wind speed, the standard σ_y and σ_z were calculated for stabilities 1 through 7. Then the spreadsheet software picked the proper σ_y and σ_z from the table coresponding to the stabilty given in the "Basic Input Parameters".

3. Following the calcuation of σ_y and σ_z, σ_{yw} and σ_{zw} need to be calculate from Ramsdell equations (7) and (8) which are done under the headings of "ssgma_yw" and "ssgma_zw". Factor 1 is the evaluation of the exponential term in the two equations. We note that the equations in the report are in error and the factor 0.0869 in the exponentials should be negative (telcon Ramsdell). Factor 2 is the evaluation for the terms in brackets in the equations. Finally factor 3 is the solution of the equations number (7) and (8). We also note the the first square root of area term found in each equation is also in error and should be the area to the first power (telcon Ramsdell).

4. Σ_{wy} and Σ_{wz} are then evaluated from equations (5) and (6) and shown in the spreadsheet as terms "SSigma_wy" and "SSigma_wy" at the bottom of the page.

5. Finally, χ/Q is evaluated as Chiqu = at the top of the spreadsheet for the basic input parameters.

6. Since spreadsheets have the capability of evaluating to tables, tables of wind speed versus stability were made for distances of 41 and 108 meters as is show on the third page.

This Spread sheet calculates the X/Q value for
a given distance and stability using the
technique given by Ramsdell in equation 9 of
the 21st DOE/NRC Nuclear Air Cleaning Conference

Basic input parameters

Distance	108	meters
Stability	1	
Wind Spd	0.5	meter/second
Bldg Area	2000	sq meter
Release Flow	0	cubic meter/second

Chiqu = 6.05E-05

Calculation of Standard Sigma Y and Sigma Z from PAVAN
subroutine 'POLYN'

Stability		AY	Sigma Y		
	1	0.3658	25.10		
	2	0.2751	18.87		
	3	0.2089	14.33		
	4	0.1471	10.09		
	5	0.1046	7.18		
	6	0.0722	4.95		
	7	n/a	0.53		
		AZ	BZ	CZ	Sigma Z
	1	0.00066	1.941	9.27	15.11
	2	0.0382	1.149	3.3	11.59
	3	0.113	0.911	0	8.05
	4	0.222	0.725	-1.7	4.92
	5	0.211	0.678	-1.3	3.75
	6	0.086	0.74	-0.35	2.40
	7	n/a	n/a	n/a	0.19

Calculation for Super Sigmas (SSigma) based upon Ramsdell's
formulations in equations 5, 6, 7 and 8

Compensation factor, k= 0.5 only characterized by Ramsdell
Stability Factor, S= 1.00 Sqrt of Stability, ref
telcon with Ramsdell

ssgma_yw			
	1	2	3
	0.810698	0.019169	10089.02
ssgma_zw			
	1	2	3
	0.810698	0.019169	10089.02

SSigma_wy	103.53
SSigma_wy	101.57

```
000000       SUBROUTINE POLYN(IC,AVAL,RESULT,LDSRT)
000013       DIMENSION AY(6),AZ(6,3),BZ(6,3),CZ(6,3),DIS(2)
000013       DIMENSION SZ(7,3),AK(11),EX(11),X(7)
000013       DATA SZ/1.114,1.322,1.633,2.0,2.431,2.889,3.398,0.982,1.230,1.519,
000013      1        1.845,2.255,2.69,3.114,0.903,1.130,1.398,1.708,2.079,
000013      2        2.462,2.903/
000013       DATA AK,EX,X/.144,.443,.78,1.38,.519,.348,.297,.449,.279,.294,.303
000013      1            ,.826,.517,.314,.11,.921,.901,.891,.731,.865,.916,.975
000013      2            ,2.0,2.301,2.602,2.903,3.204,3.505,3.806/
000013       DATA AY/.3658,.2751,.2089,.1471,.1046,.0722/
000013       DATA AZ/.192,.156,.116,.079,.063,.053,    .00066,.0382,.113,.222
000013      X,.211,.086,    .00024,.055,.113,1.26,6.73,18.05/
000013       DATA BZ/.936,.922,.905,.881,.871,.814,    1.941,1.149,.911,.725
000013      X,.678,.74,    2.094,1.098,.911,.516,.305,.18/
000013       DATA CZ/6*0.,9.27,3.3,0.,-1.7,-1.3,-.35,    -9.6,2.,..,-13.,-34.
000013      X,-48.6/
000013       DATA DIS/100.,1000./,BY/.9031/
000013       IF(LDSRT.EQ.1)   GO TO 100
000017       IF(IC.LE.7) GO TO 20
000023       IF(IC.EQ.14) GO TO 25
000027       IX=IC-7
000032       RESULT=AY(IX)*AVAL**BY
000042       GO TO 999
000043    25 F=AY(6)*AVAL**BY
000052       E=AY(5)*AVAL**BY
000061       RESULT=2.*ALOG10(F)-ALOG10(E)
000076       GO TO 500
000077    20 DO 2 L=1,2
000101       IF(AVAL.LT.DIS(L)) GO TO 3
000106     2 CONTINUE
000112       L=3
000114     3 IF(IC.EQ.7) GO TO 30
000120       RESULT=AZ(IC,L)*AVAL**BZ(IC,L)+CZ(IC,L)
000134       GO TO 999
000135    30 F=AZ(6,L)*AVAL**BZ(6,L)+CZ(6,L)
000150       E=AZ(5,L)*AVAL**BZ(5,L)+CZ(5,L)
000163       RESULT=2.*ALOG10(F)-ALOG10(E)
000201       GO TO 500
000201 C**** DESERT TYPE SIGMA'S
000202   100 IF(IC.LE.3)   GO TO 10
000206       II=IC-3
000211       RESULT=AK(II)*AVAL**EX(II)
000221       GO TO 999
000222    10 RESULT=0.
000224       AVLG=ALOG10(AVAL)
000231       DO 1 J=1,7
000233       XN=1.
000235       XD=1.
000236       DO 4 K=1,7
000241       IF(K.EQ.J) GO TO 4
000244       XN=XN*(AVLG-X(K))
000251       XD=XD*(X(J)-X(K))
000257     4 CONTINUE
000263     1 RESULT=RESULT+SZ(J,IC)*XN/XD
000301   500 RESULT=10.**RESULT
000307   999 IF(RESULT.GT.1000.) RESULT=1000.
000314       RETURN
000317       END
```

σ_y

σ_z

IC = >7 σ_z
IC = 8-14 σ_y
AVAL = dist, m
LDSRT = 0 - standard
1 - desert

Comparison Of Ramsdell

Chiqu's evaluated at 41 meters

	1	2	3	4	5	6	7
0.5	3.84E-04	3.96E-04	4.04E-04	4.10E-04	4.14E-04	4.17E-04	4.20E-04
1	6.79E-04	7.34E-04	7.73E-04	8.03E-04	8.18E-04	8.29E-04	8.39E-04
2	9.40E-04	1.15E-03	1.33E-03	1.49E-03	1.57E-03	1.62E-03	1.68E-03
4	8.63E-04	1.24E-03	1.73E-03	2.32E-03	2.70E-03	3.01E-03	3.35E-03
6	6.87E-04	1.07E-03	1.66E-03	2.58E-03	3.31E-03	4.02E-03	5.03E-03
8	5.53E-04	8.96E-04	1.49E-03	2.55E-03	3.54E-03	4.68E-03	6.69E-03
10	4.58E-04	7.57E-04	1.31E-03	2.40E-03	3.54E-03	5.04E-03	8.35E-03

Chiqu's evaluated at 108 meters

	1	2	3	4	5	6	7
0.5	6.05E-05	6.34E-05	6.54E-05	6.71E-05	6.84E-05	6.95E-05	7.05E-05
1	1.08E-04	1.18E-04	1.26E-04	1.32E-04	1.35E-04	1.38E-04	1.41E-04
2	1.53E-04	1.88E-04	2.18E-04	2.45E-04	2.61E-04	2.72E-04	2.82E-04
4	1.44E-04	2.09E-04	2.90E-04	3.88E-04	4.53E-04	5.07E-04	5.64E-04
6	1.16E-04	1.81E-04	2.82E-04	4.38E-04	5.62E-04	6.95E-04	8.44E-04
8	9.40E-05	1.53E-04	2.54E-04	4.37E-04	6.06E-04	8.06E-04	1.12E-03
10	7.81E-05	1.30E-04	2.24E-04	4.16E-04	6.11E-04	8.77E-04	1.40E-03

Ratio 41/108

	1	2	3	4	5	6	7
0.5	6.34E+00	6.24E+00	6.17E+00	6.10E+00	6.05E+00	6.00E+00	5.95E+00
1	6.28E+00	6.21E+00	6.15E+00	6.09E+00	6.04E+00	5.99E+00	5.95E+00
2	6.14E+00	6.11E+00	6.09E+00	6.06E+00	6.02E+00	5.98E+00	5.95E+00
4	5.98E+00	5.97E+00	5.97E+00	5.96E+00	5.96E+00	5.93E+00	5.95E+00
6	5.92E+00	5.90E+00	5.89E+00	5.88E+00	5.89E+00	5.87E+00	5.95E+00
8	5.89E+00	5.86E+00	5.85E+00	5.82E+00	5.83E+00	5.81E+00	5.95E+00
10	5.87E+00	5.84E+00	5.83E+00	5.78E+00	5.79E+00	5.74E+00	5.96E+00

5.974614

GE Nuclear Energy

ABWR

Date 3-12-92

Fax No. ______

To

Roger Pedersen phone 504-3162
m/s 1004

This page plus 6 page(s)

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Subject Update to SSAR 12.4

Message DAC to follow LATTER TODAY.
Warning 12.4 Not yet proofread.

CHAPTER 12.4

TABLE OF CONTENTS

TABLES

12.4 DOSE ASSESSMENT

Dose assessment is an important part of determining and projecting that the plant design and proposed methods of operation assures that occupational radiation exposure will be as low as reasonably achievable. Dose assessment depends upon estimates of occupancy, dose rates in various occupied areas, number of personnel involved in reactor operations and surveillance, routine maintenance, waste processing, refueling, in-service inspection, and special maintenance.

The goal is to reduce the exposure associated with each phase of plant operation and maintenance to the minimum level consistent with practical considerations for accomplishing each task. To achieve this goal, the ABWR design includes numerous significant design improvements to reduce occupational exposures from past experience. The design improvements include the elimination of recirculation piping and valves, improved water chemistry and low cobalt alloys at the cooling water boundary, reduced equipment maintenance and improved access, RHR discharge to the feedwater piping, overhaul handling and refueling devices, multiple main steam line plugs, automatic MSIV seat lapping system and reactor vessel stud tensioner. In assessing the collective occupational dose, each potentially significant dose-causing activity was evaluated. Values referred to as typical BWR operations are taken from references 1 through 4 which are a compendium of maintenance and work tasks for BWR-6, GESSAR.

12.4.1 Drywell Dose

The following provides the basis by which the drywell dose estimates for occupational exposure were made.

(1) The main steam isolation valves are located in the upper drywell area (4 valves) and in the reactor building outboard of the primary containment isolation wall (4 valves). These valves require periodic testing and maintenance to insure proper action and leak tightness. Typical values for BWR's for maintenance of these valves is 4,000 hours of drywell and 5,000 hours of reactor building work in effective radiation fields of 13.5 mRem/hr and 3.6 mRem/hr respectively. The ABWR design incorporates three specific features to reduce occupational exposure in the MSIV maintenance area: (1) improved water chemistry with lower overall contamination rates, (2) improved maintenance procedures with some procedures automated, and (3) reduced radiation fields, primarily due to the absence of the recirculation piping. Each area is discussed below.

Beginning in the early 1980's the BWR Owner's Group began an extensive study of the causes for failure of MSIV's to meet the technical leakage specification limits and extensive person-hours required to maintain these valves. As a result of these studies, the ABWR will use the latest technology for valve maintenance including mechanical aids for valve disassembly and assembly, automated lapping devices, and slightly relaxed leakage specifications to delete unnecessary maintenance. As a result of these aids, it is estimated that overall maintenance hours will be reduced by 50-60 percent.

Early studies on dose rates during MSIV maintenance showed increases in dose rate directly proportional to recirculation line activity. The ABWR has deleted the recirculation lines entirely thereby removing the singly most significant source of radiation in the drywell. The second most significant dose for MSIV operations will be the deposited and suspended activity in the feedwater lines. The deposited activity in the feedwater lines is expected to be lower than typical BWRs owing to an enhanced condensate system with full clean up of all condensate water, a 2% reactor water clean up system, and titanium condenser tubes. Additionally, the ABWR is designed to limit the use of cobalt bearing materials on moving components which have historically been identified as major sources of in water contamination. Overall, the feedwater line radiation is expected to be a factor of three lower than current BWRs. Because of these factors, it is expected that the effective dose rate in the drywell will be 1.8mRem/hr and 1.3mRem/hr in the steam tunnel outboard of the primary containment.

(2) Drywell valve and pump maintenance other than the MSIVs consists primarily of maintaining the safety relief valves (SRVs) which for the most part consist of minor maintenance or removal of valves to a maintenance facility. Overall typical values for a BWR for these tasks are 1,450 person-hours per year in an effective radiation field of

17mRem/hr. In the ABWR, the primary source of radiation exposure, the recirculation lines and pumps, have been removed. Overall the reduction in drywell dose level is for these types of maintenance is expected to be a factor of two or 9mRem/hr. Overhead tracks and in place removal equipment is provided in the ABWR for an estimated person-power hours reduction to 1,150 person-hour per year broken down into 200 person-hours for 18 SRV maintenance at 6mRem/hr, 200 person-hours per year to pull and replace 3 RIPs with one heat exchanger at 20mRem/hr, and the remainder on miscellaneous valves at 4.5mRem/hr.

(3) Control rod drive maintenance is significantly reduced in the ABWR with the introduction of fine motion control rod drives (FMCRD). Based upon European experience, two FMCRDs will be replaced and repaired per outage along with twenty motors. Estimated work will consist of 64 person-hours under vessel preparation, 40 person-hours FMCRD removal and reinstallation, 200 person hours motor removal and installation, and 64 person-hours cleanup. Typical under vessel effective dose rates are 17mRem/hr but because of the removal of the recirculation pumps and lines has dose rates have been reduced to 6.5mRem/hr.

(4) The LPRM/TIP system assumes the servicing of two sensors per year and is based upon a total of 200 person-hours per year at an effective dose rate of 50mRem/hr which is typical for BWR operations.

(5) Inservice inspection consists of primarily NDE examination of vessel and piping systems and welds. Typical BWR values are 2400 person-hours per year at 12mRem/hr effective exposure rate. ABWR inservice inspection is estimated based upon the following:

Elimination of recirculation lines and pumps with the following savings:

Elimination of 14 nozzle inspections at 2 per year, saving 360 person hours

Elimination of shield penetration and shield plug removal saving 240 person hours per year

Elimination of ~~Reduction on~~ weld inspection on recirculation lines estimated at 240 person hour per year

Reduction in drywell dose by 50% with the provision that the feedwater line dose is more than half the recirculation line dose and general drywell dose level and therefore removal of recirculation line inspection is estimated to be weighted at twice the general drywell dose rate. Reword

Overall it is estimated that by use of automated turtles for inspection person-hour expended in ISI will be reduced by a factor of two.

The ABWR uses a forged ring pressure vessel in comparison to older plate welded vessels reducing the total vessel weld length inspection by 30% and the total weld inspection by 10% in the drywell.

The ABWR design incorporates specific access into inspection areas past insulation areas with an estimated savings of 120 person-hours. Reword

Overall person-hours reduction is 1,200 person-hours at approximately half the typical effective dose rate or 5.5mRem/hr.

(6) Other drywell work includes items such as minor valve maintenance, instrumentation work, and all other drywell work. Typical BWR work in this area estimates 5,500 person-hours per year at 17 mRem/hr. Overall reduction in this effort due to ABWR design improvements are:

Significant savings in total hours are estimated due to removal of the recirculation lines with miscellaneous recirculation line work such as line snubbers, fewer drywell cooling units, and less assembly/disassembly work on insulation due to the use of automated units. Overall it is estimated that 2,000 person hours savings can be made.

Overall reduction in the drywell radiation due to removal of the recirculation system results in the reduction of the overall upper drywell dose rate to 1.8mRem/hr and the lower drywell dose rate to 5.6mRem/hr since the components involved such as drywell coolers typically do not carry radioactive inventory. Assuming that of the remaining 3,500 person-hour, 2,000 is upper drywell work and 1,500 is lower drywell work at their respective effective dose rates.

12.4.2 Reactor Building Dose

The following provides the basis by which the reactor building dose estimates for occupational exposure were made

(1) Vessel access and reassembly typically requires 4500 person-hours of work at an effective dose rate of 3 mr/hr. The ABWR work will involve the use of a stud tensioner for a 96 bolt top head. The projected time to remove 96 bolts with this equipment is between 600 to 1200 person-hours. Due to the larger ABWR vessel and expected reduced water contamination with the improved clean up system, the estimated projected effective dose rate is 1.5 mRem/hr.

(2) ABWR refuelling is accomplished via an automated refuelling bridge. All operations for refuelling are accomplished from an enclosed automation center off the refuelling floor. Time for refuelling is reduced from a typical 4,400 person-hours down to 2,000 person and from an effective dose rate of 2.5 mRem/hr to less than 0.2mRem/hr.

(3) RHR/CUW maintenance work consists of inspections for two pumps per year in each system. In the RHR system this consumes 150 person-hours per year at an effective dose rate of 40mRem/hr. In the CUW system this typically uses 1400 person-hours per year at an effective dose rate of 14mRem/hr. ABWR will use canned pumps for both system with an estimated reduction in maintenance to 100 person-hours per pump. With improved water chemistry and overall reductions in reactor water concentrations due to the two percent cleanup system the effective dose rate is estimated at twenty percent of the typical value for these system.

(4) FMCRD rebuilding estimates are taken from similar work done in Europe since no significant U.S. data exists to date. Two drives will be rebuilt at an effective dose rate of 4.5 mRem/hr and 30-60 hours per drive.

(5) Instrumentation work typically requires 1,000 person-hours of work per year at an effective dose rate of 5.0 mRem/hr. ABWR should take about the same effort in instrumentation, however because of the increased emphasis and improved water chemistry systems, should reduce the effective dose rate to two-thirds the typical value or 3.0mRem/hr.

(6) All other work in the reactor building typically takes 7,400 person-hours per year at an effective dose rate of 2.8mRem/hr. This work includes all valve work, RIP rebuild work, minor maintenance, and CRD hydraulic line work. The major task in this area is the hydraulic control units which require 5,000 person-hours per year at an effective dose rate of 3.3mRem/hr. With the use of the FMCRD units, an additional savings of 2,000 person-hours is anticipated. In addition, the ABWR reactor building has been designed to provide for ease of maintenance with overhead lifts, coordinated hatch ways and ample space to maintain in place equipment. In addition, with the exception of one tank and the pressure vessel, all the equipment in the reactor building is removable with those pieces which can be expected to be moved being palatalized. Because of these factors, an overall reduction in work of 1,000 person-hours is estimated. Because of the improved water chemistry the overall effective dose rate is anticipated at one-half the typical BWR dose rate.

12.4.3 Radwaste Building Dose

Radwaste building work consists of pump and valve maintenance, shipment handling, radwaste management, and general clean up activity. Typically, 6,700 hours are expended per year at an effective dose rate of 5.5mRem/hr. The ABWR radwaste building is designed along the same lines as newer radwaste facilities overseas. The building incorporates enhanced remote control and shielding for handling of resin materials which is expected to reduce overall maintenance by 1500 to 2000 hour per year at significantly reduced dose levels. In addition, radwaste pumps for ABWR are expected to utilize air driven, rack mounted pumps. Such pumps which are designed to handle slurries have been proven to show much longer life times between maintenance and being basically a very small portable pump, can be readily replaced. Replaced pumps are then subject to intense chemical decontamination prior to maintenance and repair. Overseas utilities have reported occupational exposures typically less than 1 person-rem per year using this design. For ABWR assuming 2,000 hours reduction in maintenance due to remote handling and an additional 500 hours reduction for pump replacement, 4,200 hours per year are estimated with reduced effective dose rates of 2.5mRem/hr owing primarily to remotting those jobs involving high radiation exposure.

12.4.4 Turbine Building Dose

(1) Typical BWR valve maintenance in the turbine building uses 1,150 hours per year at an effective dose rate of 9.5mRem/hr. The valve maintenance requirements for ABWR do not vary significantly over current plants, therefore the total hours for this type of work is assumed as approximately the same excepting minor adjustments for improved valves, maintenance jigs, and automated devices will lower the estimated maintenance time to 1,000 hours. The effective dose rate of 9.5 mRem/hr is estimated at more than one half this value due to basically improvements in BWR fuel over the generation of fuel from which this data was taken bringing the effective dose rate down to 3.9mRem/hr. In addition, beta shielding is recommended for work on valving where possible which it is estimated will reduce the overall effective dose rate by an additional 10% to 3.5mRem/hr.

(2) In a similar fashion the turbine maintenance work typically requires 18,500 hours of work at an effective dose rate of 0.3mRem/hr. With additional operational improvements in automating turbine maintenance, overall work is estimated to be reduced to 15,500 hours. The effective dose rate for the turbine is not expected to be as sensitive to fuel performance as will the turbines but is estimated to reflect a decrease in dose to 0.2mRem/hr for turbine overhaul work.

(3) Work on the turbine hall condensate system typically requires 2,000 hours per year at an effective dose rate of 7.5mrem/hr. The condensate system in ABWR uses hollow-fiber filled filters which require half the maintenance of a typical system. In addition, with the plant incorporating Fe control in the feedwater system and a significant reduction in cobalt bearing materials, the overall effective dose rate is estimated at half the above value.

(4) Other work in the turbine building typically takes 13,140 hours per year at an effective dose rate of 0.1mRem/hr. Only minor changes can be assumed with ABWR with some remote operations and slight reductions in operating exposures. For the ABWR it is estimated that a 10% reduction can be realized with improving technology with no significant change in dose rate.

12.4.5 Work at Power

Work at power typically requires 5,000 hours per year at an effective dose rate of 6.6mRem/hr for the BWR. This category covers literally all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment, and minor equipment repair. Overall the ABWR has been designed with more automated and remotted equipment. It is expected that items of routine monitoring will be performed by camera or additional instrumentation. Most equipment in ABWR is palatalized which permits quick and easy replacement and removal for decontamination and repair. Therefore a reduction in actual hours need at power is estimate at 1,000 hours less than the typical value. In the area of effective dose rate, the ABWR is expected to have significantly lower general radiation levels over current plants owing to more stringent water chemistry controls, a full flow condensate flow system, a 2% clean up water program, titanium condenser tubes, Fe feedwater control, and low cobalt usage. In addition, the ABWR is the most compartmentalized BWR design which (1) permits better shielding in specific work areas, and (2) lowers collateral radiation contamination. Overall then it is estimated that the effective dose rate for work at power will be slightly over two thirds the typical rate or 4.0mRem/hr.

12.4.6 References

1. Knecht, P.D., *BWR/6 Drywell and Containment Maintenance and Testing Access Time Estimates*, GE Report NEDE-23819, May 1978.

2. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Radwaste Building*, GE Report NEDE-23996-2, May 1979.

3. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Auxiliary and Fuel Buildings*, GE Report NEDE-23996-1, May 1979.

4. *Study of Advanced BWR Features, Plant Definition/Feasibility Results*, Volume III, Appendix Part G, GE NEDE-24679, Oct 1979.

Table 12.4-1

PROJECTED ANNUAL RADIATION EXPOSURE

Operation Task	SSAR Section	Hours per year	mRem/hr	Person-Rem/yr
Drywell				
MSIV	12.4.1(1)	4,200	1.5	6.3
SRV,RIP,etc	12.4.1(2)	1,150	7.5	8.6
FMCRD	12.4.1(3)	370	6.5	2.4
LPRM/TIP	12.4.1(4)	200	50.0	10.0
ISI	12.4.1(5)	1,200	5.5	6.6
Other	12.4.1(6)	3,500	3.5	12.3
Total		10,620		46.2
Reactor Building				
Vessel	12.4.2(1)	1,200	1.5	1.8
Refueling	12.4.2(2)	2,000	0.2	0.4
RHR/CUW	12.4.2(3)	400	8.0	3.2
FMCRD	12.4.2(4)	120	4.5	0.5
Instrument	12.4.2(5)	1,000	3.0	3.0
Other	12.4.2(6)	4,400	1.5	6.6
Total		9,120		15.5
Radwaste Building	12.4.3	4,200	2.5	10.5
Turbine Building				
Valve Maint	12.4.4(1)	1,000	3.5	3.5
Turbine Ovrhl	12.4.4(2)	15,500	0.2	3.1
Condensate	12.4.4(3)	1,000	3.5	3.5
Other	12.4.4(4)	11,800	0.1	1.2
Total		29,300		11.3
Work at Power	12.4.5	4,000	4.0	16.0
Totals		**48,120**		**99.5**

GE Nuclear Energy

ABWR

Date 3-16-92

To Roger Pedersen 504-3162 M/S 10D4

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Subject Revised DAC

Message Revised DAC per Telcon

Table 3.2a: PLANT SHIELDING DESIGN

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The plant design shall provide radiation shielding for rooms, corridors and operating areas commensurate with their occupancy requirements to maintain radiation exposures to plant personnel as low as reasonably achievable.	1. An analysis of the expected radiation levels in each plant area will be performed to verify the adequacy of the shielding design. This analysis shall consider the following: a. Confirmatory calculations shall consider all significant radiation sources (greater than 5% contribution) for an area. Radiation source strength in plant systems and components will be determined based upon an assumed source term of 100,000 μCurie/second offgas release rate (after 30 minutes decay), a 200 μCurie/gram steam N-16 source term at the vessel exit nozzle, and a core inventory commensurate with a 4005 MWT equilibrium core at 51.6 kwatt/liter. All source terms shall be adjusted for radiological decay and buildup of activated corrosion and wear products. b. Commonly accepted shielding codes, using nuclear properties derived from well known references (such as Vitamin C and ANSI/ANS-6.4) shall be used to model and evaluate plant radiation environments. 1) For non-complex geometries, point kernel shielding codes (such as QAD or GGG) shall be used. 2) For complex geometries, more sophisticated two or three dimensional transport codes (such as DORT or TORT) shall be used. c. In any calculation, a safety factor shall be applied based upon benchmark comparisons of the code and data collected from known and measured environments.	1. Maximum expected radiation levels are well within (25% or less) of the radiation zone designation, for each plant area, as indicated in Figures __________

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The plant design shall provide shielded cubicles, labyrinth access, and space for temporary shielding to allow for maintenance of plant components without significant radiation exposure from adjacent plant systems or equipment.	2. Using the methods identified in (1) above, radiation levels present in areas where maintenance is performed shall be evaluated for the contribution from adjacent high radiation areas and equipment.	2. Shielding design is such that radiation from adjacent areas shall contribute no more than a small fraction (10% or less) of the radiation field intensity or less than 0.06mrem/hr whichever is larger, in plant areas where maintenance is performed.
3. The plant radiation shielding design shall permit operators to perform required safety functions in vital areas of the plant (including access and egress of these areas) under accident conditions.	3. An analysis of the expected high radiation levels in each area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident (vital area) shall be performed to verify the adequacy of the plant shielding design. This analysis shall use calculational methods consistent with (1.b) above and a radiation source term (adjusted for radioactive decay) based on the following: a. Liquid containing systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and recirculation liquids recirculated by the residual heat removal system (RHR), the high pressure core flooder (HPCF), and the reactor core isolation cooling (RCIC) systems. b. Gas containing systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor containing systems (such as the main steam lines) these core inventory fractions are assumed to be contained in the reactor coolant vapor space.	3. Under accident conditions, radiation shielding design allows access, occupancy and egress of vital areas such that personnel radiation exposures do not exceed 5 rem to the whole body, or its equivalent, for the duration of the accident (based on the required frequency of access to each vital area). For areas requiring continuous occupancy (such as the control room), local radiation hot spots shall not exceed 15 mrem/hr (averaged over 30 days).

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The plant design shall provide radiation shielding to maintain radiation exposure to the general public as low as is reasonably achievable.	4. Using the methods identified in (1) above, the radiation dose to the maximally exposed member of the general public from direct and scattered shall be determined.	4. The radiation dose to the maximally exposed member of the public when combined with dose commitments from all other radiation pathways from the nuclear fuel cycle (including liquid and gaseous pathways) shall be within 25 r[illegible] whole body dose or 75mrem to any o[illegible] organ

Table 3.2b: VENTILATION AND AIRBORNE MONITORING

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Plant design shall provide adequate containment of airborne radioactive materials and the ventilation system will ensure that concentrations of airborne radionuclides are maintained at levels consistent with personnel access requirements.	1. Expected concentrations of airborne radioactive material shall be calculated by nuclide for normal plant operations, anticipated operational occurrences for each equipment cubicle, corridor, and operating area requiring personnel access. Calculations shall consider: a. Design ventilation flow rates for each area, b. Typical leakage characteristics for equipment located in each area, and b. A radiation source term in each fluid system shall be determined based upon an assumed offgas rate of 100,000 Curie/second (30 minute decay) appropriately adjusted for radiological decay and buildup of activated corrosion and wear products.	1. Calculation of radioactive airborne concentration shall demonstrate that: a. For normally occupied rooms and areas of the plant (i.e. those areas requiring routine access to operate and maintain the plant) equilibrium concentrations of airborne nuclides will be a small fraction (10% or less) of the occupational concentration limits listed in 10 CFR 20 Appendix B. b. For rooms that require infrequent access (such as for non-routine equipment maintenance), the ventilation system shall be capable of reducing radioactive airborne concentrations to (and maintaining them at) the occupational concentration limits listed in 10CFR20 Appendix B during the periods that occupancy is required. c. For rooms that seldom require access (such as tank rooms), plant design shall provide sufficient containment and ventilation to ensure airborne contamination does not spread to other areas.

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Airborne radioactivity monitoring shall be provided for those normally occupied areas of the plant in which there exists as significant potential for airborne contamination (greater than 0.1 per year)	2. An analysis shall be performed to identify the plant areas that require airborne radioactivity monitoring.	2. Airborne radioactivity monitoring system shall: a. Have the capability of detecting concentrations equivalent to the occupational concentration limits of 10CFR20 Appendix B, for the most restrictive particulate and/or iodine radionuclide in the area, within 10 hours. b. Provide a calibrated response, representatvie of the concentrations within the area (i.E. air sampling monitors in ventilation exhaust streams shall collect and isokinetic sample). c. Provide local audible alarms (visual alarms in high noise areas) with variable alarm set points, and readout/annunciation capability in the control room.

GE Nuclear Energy

ABWR

Date 3-19-92

To Roger Pedersen
504-3162

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Subject Revised DAC & Sec 12.4

Message PER TELCON

Table 3.2a: PLANT SHIELDING DESIGN

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The plant design shall provide radiation shielding for rooms, corridors and operating areas commensurate with their occupancy requirements to maintain radiation exposures to plant personnel as low as reasonably achievable.	1. An analysis of the expected radiation levels in each plant area will be performed to verify the adequacy of the shielding design. This analysis shall consider the following: a. Confirmatory calculations shall consider all significant radiation sources (greater than 1% contribution) for an area. Radiation source strength in plant systems and components will be determined based upon an assumed source term of 100,000 μCurie/second offgas release rate (after 30 minutes decay), a 200 μCurie/gram-steam N-16 source term at the vessel exit nozzle, and a core inventory commensurate with a 4005 MWT equilibrium core at 51.6 kwatt/liter. All source terms shall be adjusted for radiological decay and buildup of activated corrosion and wear products. b. Commonly accepted shielding codes, using nuclear properties derived from well known references (such as Vitamin C and ANSI/ANS-6.4) shall be used to model and evaluate plant radiation environments. 1) For non-complex geometries, point kernel shielding codes (such as QAD or GGG) shall be used. 2) For complex geometries, more sophisticated two or three dimensional transport codes (such as DORT or TORT) shall be used. c. In any calculation, a safety factor shall be applied based upon benchmark comparisons of the code and data collected from known and measured environments.	1. Maximum expected radiation levels are well within (25% or less) of the radiation zone designation, for each plant area, as indicated in Figures __________.

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The plant design shall provide shielded cubicles, labyrinth access, and space for temporary shielding to allow for maintenance of plant components without significant radiation exposure from adjacent plant systems or equipment.	2. Using the methods identified in (1) above, radiation levels present in areas where maintenance is performed shall be evaluated for the contribution from adjacent high radiation areas and equipment.	2. Shielding design (with temporary shielding installed, where appropriate) is such that radiation from adjacent areas shall contribute no more than a small fraction (10% or less) of the radiation field intensity or less than 0.06mrem/hr whichever is larger, in plant areas where maintenance is performed.
3. The plant radiation shielding design shall permit operators to perform required safety functions in vital areas of the plant (including access and egress of these areas) under accident conditions.	3. An analysis of the expected high radiation levels in each area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident (vital area) shall be performed to verify the adequacy of the plant shielding design. This analysis shall use calculational methods consistent with (1.b) above and a radiation source term (adjusted for radioactive decay) based on the following: a. Liquid containing systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and recirculation liquids recirculated by the residual heat removal system (RHR), the high pressure core flooder (HPCF), and the reactor core isolation cooling (RCIC) systems. b. Gas containing systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor containing systems (such as the main steam lines) these core inventory fractions are assumed to be contained in the reactor coolant vapor space.	3. Under accident conditions, radiation shielding design allows access, occupancy and egress of vital areas such that personnel radiation exposures do not exceed 5 rem to the whole body, or its equivalent, for the duration of the accident (based on the required frequency of access to each vital area). For areas requiring continuous occupancy (such as the control room), local radiation hot spots shall not exceed 15 mrem/hr (averaged over 30 days).

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The plant design shall provide radiation shielding to maintain radiation exposure to the general public as low as is reasonably achievable.	4. Using the methods identified in (1) above, the radiation dose to the maximally exposed member of the general public from direct and scattered shall be determined.	4. The radiation dose to the maximally exposed member of the public is a small fraction (10% or less) of the dose limit to a member of the public listed in 40CFR190.

Table 3.2b: VENTILATION AND AIRBORNE MONITORING

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Plant design shall provide adequate containment of airborne radioactive materials and the ventilation system will ensure that concentrations of airborne radionuclides are maintained at levels consistent with personnel access requirements.	1. Expected concentrations of airborne radioactive material shall be calculated by nuclide for normal plant operations, anticipated operational occurrences for each equipment cubicle, corridor, and operating area requiring personnel access. Calculations shall consider: a. Design ventilation flow rates for each area, b. Typical leakage characteristics for equipment located in each area, and b. A radiation source term in each fluid system shall be determined based upon an assumed offgas rate of 100,000 Curie/second (30 minute decay) appropriately adjusted for radiological decay and buildup of activated corrosion and wear products.	1. Calculation of radioactive airborne concentration shall demonstrate that: a. For normally occupied rooms and areas of the plant (i.e. those areas requiring routine access to operate and maintain the plant) equilibrium concentrations of airborne nuclides will be a small fraction (10% or less) of the occupational concentration limits listed in 10 CFR 20 Appendix B. b. For rooms that require infrequent access (such as for non-routine equipment maintenance), the ventilation system shall be capable of reducing radioactive airborne concentrations to (and maintaining them at) the occupational concentration limits listed in 10CFR20 Appendix B during the periods that occupancy is required. c. For rooms that seldom require access (such as tank rooms), plant design shall provide sufficient containment and ventilation to ensure airborne contamination does not spread to other areas.

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Airborne radioactivity monitoring shall be provided for those normally occupied areas of the plant in which there exists as significant potential for airborne contamination (greater than 0.1 per year)	2. An analysis shall be performed to identify the plant areas that require airborne radioactivity monitoring.	2. Airborne radioactivity monitoring system shall: a. Have the capability of detecting the time integrated change in concentrations of the most limiting particulate and iodine radionuclides in each area equivalent to the occupational concentration limits in 10CFR20, Appendix B for 10hours. b. Provide a calibrated response, representatvie of the concentrations within the area (i.E. air sampling monitors in ventilation exhaust streams shall collect and isokinetic sample). c. Provide local audible alarms (visual alarms in high noise areas) with variable alarm set points, and readout/annunciation capability in the control room.

17mRem/hr. In the ABWR, the primary source of radiation exposure, the recirculation lines and pumps, have been removed. Overall the reduction in drywell dose level is for these types of maintenance is expected to be a factor of two or 9mRem/hr. Overhead tracks and in place removal equipment is provided in the ABWR for an estimated person-power reduction to 1,150 person-hour per year broken down into 200 person-hours for 18 SRV maintenance at 6mRem/hr, 200 person-hours per year to pull and replace 3 RIPs with one heat exchanger at 20mRem/hr, and the remainder on miscellaneous valves at 4.5mRem/hr.

(3) Control rod drive maintenance is significantly reduced in the ABWR with the introduction of fine motion control rod drives (FMCRD). Based upon European experience, two FMCRDs will be replaced and repaired per outage along with twenty motors. Estimated work will consist of 64 person-hours under vessel preparation, 40 person-hours FMCRD removal and reinstallation, 200 person hours motor removal and installation, and 64 person-hours cleanup. Typical under vessel effective dose rates are 17mRem/hr but because of the removal of the recirculation pumps and lines has been reduced to 6.5mRem/hr.

(4) The LPRM/TIP system assumes the servicing of two sensors per year and is based upon a total of 200 person-hours per year at an effective dose rate of 50mRem/hr which is typical for BWR operations.

(5) Inservice inspection consists of primarily NDE examination of vessel and piping systems and welds. Typical BWR values are 2400 person-hours per year at 12mRem/hr effective exposure rate. ABWR inservice inspection is estimated based upon the following:

Elimination of recirculation lines and pumps with the following savings:

Elimination of 14 nozzle inspections at 2 per year, saving 360 person hours

Elimination of shield penetration and shield plug removal saving 240 person hours per year

Elimination of ~~Reduction on~~ weld inspection on recirculation lines estimated at 240 person hour per year

Reduction in drywell dose by 50% with the provision that the feedwater line dose is more than half the recirculation line dose and general drywell dose level and therefore removal of recirculation line inspection is estimated to be weighted at twice the general drywell dose rate.

Overall it is estimated that by use of automated turtles for inspection person-hour expended in ISI will be reduced by a factor of two.

The ABWR uses a forged ring pressure vessel in comparison to older plate welded vessels reducing the total vessel weld length inspection by 30% and the total weld inspection in the drywell by 10%.

The ABWR design incorporates specific access into inspection areas past insulation areas with an estimated savings of 120 person-hours.

Overall person-hours reduction is 1,200 person-hours at approximately half the typical effective dose rate or 5.5mRem/hr.

(6) Other drywell work includes items such as minor valve maintenance, instrumentation work, and all other drywell work. Typical BWR work in this area estimates 5,500 person-hours per year at 17 mRem/hr. Overall reduction in this effort due to ABWR design improvements are:

Significant savings in total hours are estimated due to removal of the recirculation lines with miscellaneous recirculation line work such as line snubbers, fewer drywell cooling units, and less assembly/disassembly work on insulation due to the use of automated units. Overall it is estimated that 2,000 person hours savings can be made.

Overall reduction in the drywell radiation due to removal of the recirculation system results in the reduction of the overall upper drywell dose rate to 1.8mRem/hr and the lower drywell dose rate to 5.6mRem/hr since the components involved such as drywell coolers typically do not carry radioactive inventory. Assuming that of the remaining 3,500 person-hour, 2,000 is upper drywell work and 1,500 is lower drywell work at their respective effective dose rates.

12.4.2 Reactor Building Dose

The following provides the basis by which the reactor building dose estimates for occupational exposure were made.

(1) Vessel access and reassembly typically requires 4500 person-hours of work at an effective dose rate of 3 mr/hr. The ABWR work will involve the use of a stud tensioner for a 96 bolt top head. The projected time to remove 96 bolts with this equipment is between 600 to 1200 person-hours. Due to the larger ABWR vessel and expected reduced water contamination with the improved clean up system, the estimated projected effective dose rate is 1.5 mRem/hr.

(2) ABWR refuelling is accomplished via an automated refuelling bridge. All operations for refuelling are accomplished from an enclosed automation center off the refuelling floor. Time for refuelling is reduced from a typical 4,400 person hours down to 2,000 person and from an effective dose rate of 2.5 mRem/hr to less than 0.2mRem/hr.

(3) RHR/CUW maintenance work consists of inspections for two pumps per year in each system. In the RHR system this consumes 150 person-hours per year at an effective dose rate of 40mRem/hr. In the CUW system this typically uses 1400 person-hours per year at an effective dose rate of 14mRem/hr. ABWR will use canned pumps for both system with an estimated reduction in maintenance to 100 person-hours per pump. With improved water chemistry and overall reductions in reactor water concentrations due to the two percent cleanup system the effective dose rate is estimated at twenty percent of the typical value for these system.

(4) FMCRD rebuilding estimates are taken from similar work done in Europe since no significant U.S. data exists to date. Two drives will be rebuilt at an effective dose rate of 4.5 mRem/hr and 30-60 hours per drive.

(5) Instrumentation work typically requires 1,000 person-hours of work per year at an effective dose rate of 5.0 mRem/hr. ABWR should take about the same effort in instrumentation, however because of the increased emphasis and improved water chemistry systems, should reduce the effective dose rate to two-thirds the typical value or 3.0mRem/hr.

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12.4.3 Radwaste Building Dose

Radwaste building work consists of pump and valve maintenance, shipment handling, radwaste management, and general clean up activity. Typically, 6,700 hours are expended per year at an effective dose rate of 5.5mRem/hr. The ABWR radwaste building is designed along the same lines as newer radwaste facilities overseas. The building incorporates enhanced remote control and shielding for handling of resin materials which is expected to reduce overall maintenance by 1500 to 2000 hour per year at significantly reduced dose levels. In addition, radwaste pumps for ABWR are expected to utilize air driven, rack mounted pumps. Such pumps which are designed to handle slurries have been proven to show much longer life times between maintenance and being basically a very small portable pump, can be readily replaced. Replaced pumps are then subject to intense chemical decontamination prior to maintenance and repair. Overseas utilities have reported occupational exposures typically less than 1 person-rem per year using this design. For ABWR assuming 2,000 hours reduction in maintenance due to remote handling and an additional 500 hours reduction for pump replacement, 4,200 hours per year are estimated with reduced effective dose rates of 2.5mRem/hr owing primarily to remotting those jobs involving high radiation exposure.

12.4.4 Turbine Building Dose

(1) Typical BWR valve maintenance in the turbine building uses 1,150 hours per year at an effective dose rate of 9.5mRem/hr. The valve maintenance requirements for ABWR do not vary significantly over current plants, therefore the total hours for this type of work is assumed as approximately the same excepting minor adjustments for improved valves, maintenance jigs, and automated devices will lower the estimated maintenance time to 1,000 hours. The effective dose rate of 9.5 mRem/hr is estimated at more than one half this value due to basically improvements in BWR fuel over the generation of fuel from which this data was taken bringing the effective dose rate down to 3.9mRem/hr. In addition, beta shielding is recommended for work on valving where possible which it is estimated will reduce the overall effective dose rate by an additional 10% to 3.5mRem/hr.

Replace with Rev 5

(2) In a similar fashion the turbine maintenance work typically requires 18,500 hours of work at an effective dose rate of 0.3mRem/hr. With additional operational improvements in automating turbine maintenance, overall work is estimated to be reduced to 15,500 hours. The effective dose rate for the turbine is not expected to be as sensitive to fuel performance as will the turbines but is estimated to reflect a decrease in dose to 0.2mRem/hr for turbine overhaul work.

(3) Work on the turbine hall condensate system typically requires 2,000 hours per year at an effective dose rate of 7.5mrem/hr. The condensate system in ABWR uses hollow-fiber filled filters which require half the maintenance of a typical system. In addition, with the plant incorporating Fe control in the feedwater system and a significant reduction in cobalt bearing materials, the overall effective dose rate is estimated at half the above value.

(4) Other work in the turbine building typically takes 13,140 hours per year at an effective dose rate of 0.1mRem/hr. Only minor changes can be assumed with ABWR with some remote operations and slight reductions in operating exposures. For the ABWR it is estimated that a 10% reduction can be realized with improving technology with no significant change in dose rate.

12.4.5 Work at Power

Work at power typically requires 5,000 hours per year at an effective dose rate of 6.6mRem/hr for the BWR. This category covers literally all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment, and minor equipment repair. Overall the ABWR has been designed with more automated and remotted equipment. It is expected that items of routine monitoring will be performed by camera or additional instrumentation. Most equipment in ABWR is palatalized which permits quick and easy replacement and removal for decontamination and repair. Therefore a reduction in actual hours need at power is estimate at 1,000 hours less than the typical value. In the area of effective dose rate, the ABWR is expected to have significantly lower general radiation levels over current plants owing to more stringent water chemistry controls, a full flow condensate flow system, a 2% clean up water program, titanium condenser tubes, Fe feedwater control, and low cobalt usage. In addition, the ABWR is the most compartmentalized BWR design which (1) permits better shielding in specific work areas, and (2) lowers collateral radiation contamination. Overall then it is estimated that the effective dose rate for work at power will be slightly over two thirds the typical rate or 4.0mRem/hr.

Replace with Rev 6

12.4.6 References

1. Knecht, P.D., *BWR/6 Drywell and Containment Maintenance and Testing Access Time Estimates*, GE Report NEDE-23819, May 1978.

2. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Radwaste Building*, GE Report NEDE-23996-2, May 1979.

3. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Auxiliary and Fuel Buildings*, GE Report NEDE-23996-1, May 1979.

4. *Study of Advanced BWR Features, Plant Definition/Feasibility Results*, Volume III, Appendix Part G, GE NEDE-24679, Oct 1979.

Table 12.4-1

PROJECTED ANNUAL RADIATION EXPOSURE

Operation Task	SSAR Section	Hours per year	mRem/hr	Person-Rem/yr
Drywell				
MSIV	12.4.1(1)	4,200	1.5	6.3
SRV,RIP,etc	12.4.1(2)	1,150	7.5	8.6
FMCRD	12.4.1(3)	370	6.5	2.4
LPRM/TIP	12.4.1(4)	200	50.0	10.0
ISI	12.4.1(5)	1,200	5.5	6.6
Other	12.4.1(6)	3,500	3.5	12.3
Total		10,620		46.2
Reactor Building				
Vessel	12.4.2(1)	1,200	1.5	1.8
Refueling	12.4.2(2)	2,000	0.2	0.4
RHR/CUW	12.4.2(3)	400	8.0	3.2
FMCRD	12.4.2(4)	120	4.5	0.5
Instrument	12.4.2(5)	1,000	3.0	3.0
Other	12.4.2(6)	4,400	1.5	6.6
Total		9,120		15.5
Radwaste Building	12.4.3	4,200	2.5	10.5
Turbine Building				
Valve Maint	12.4.4(1)	1,000	~~3.5~~ 3.9	~~3.5~~ 3.9
Turbine Ovrhl	12.4.4(2)	15,500	0.2	3.1
Condensate	12.4.4(3)	1,000	3.5	3.5
Other	12.4.4(4)	11,800	0.1	1.2
Total		29,300		~~11.3~~ 11.7
Work at Power	12.4.5	4,000	4.0	16.0
Totals		48,120		99.~~5~~9

Revision 1

Replace paragraph beginning with "Reduction in drywell dose by 50%...." with:

Reduction in drywell dose by 50% based upon the assumption that the contact dose rate on the feedwater line is less than half the contact dose rate on the typical BWR recirculation line. Hence at equal distances from the line, the total general drywell dose rate which is dominated by the recirculation and feedwater lines will be less than half what is typically seen with recirculation lines.

Revision 2

Replace paragraph beginning with "The ABWR design incorporates...." with:

The ABWR design incorporates specific access panels and shield doors into required inspection areas permitting easy bypass of insulation areas resulting in an estimated person hour savings of 120 person hours.

Revision 3

Replace sentence beginning with "Typical BWR work in this......" with:

These miscellaneous tasks in the drywell consume on the average 5,500 person-hours per year in a radiation field of 17mRem/hr. However, this average is a combination of some specific higher radiation tasks such as work on recirculation lines (involving snubbers, weld inspection, etc) and many lower radiation tasks such as work on drywell coolers.

Revision 4

Replace sentences beginning with " For ABWR assuming 2,000 ..." to end of paragraph with:

For ABWR, it is then assumed that the maintenance effort expended per year is reduced by 2,000 person hours from 6,700 to 4,700 person hours due to the introduction of automated equipment. An additional reduction of 500 person hours down to 4,200 person hours is assumed based upon the use of air pumps as specified above. The overall radiation field to which the worker is exposed on the average is then expected to be reduced from 5.5mRem/hr to 2.5mRem/hr since most of the high radiation tasks are eliminated by automation or remoting the tasks or in the case of the air pumps reduced by decontamination at separate facilities prior to pump maintenance.

Revision 5

Replace sentences beginning with "The effective dose rate of 9.5....." to end of paragraph with:

In the ABWR, the estimated effective radiation field of 3.9 mRem/hr for turbine building work is expected to be less than half the typical dose rate of 9.5mRem/hr due to the use of newer fuels which are less resistant to pin size leaks. The radiation fields in the turbine hall during maintenance are a combination of contamination from fission products from the fuel and corrosion products from the vessel and piping. Offgas measurements of the performance of the newer fuels when operated under proper water chemistry standards (required for ABWR) have shown fission product releases and orders of magnitude less than older fuels. Likewise the ABWR has placed stringent controls over material usage especially in the vessel and other high temperature components to minimize corrosion product releases.

<u>Revision 6</u>

Replace sentences beginning with "In addition, the ABWR is....." to end of paragraph with:

In addition, the ABWR has in the basic design, compartmentalized all major pieces of equipment so that any piece of equipment can be maintained or removed for maintenance without affecting the normal plant operations. This design concept thereby reduces radiation exposure to personnel maintaining or testing one piece of equipment from both shine and airborne contamination from other equipment. Finally, the ABWR has incorporated in the basic design the use of Hydrogen Water Chemistry (HWC) and the additional shielding necessary to protect from the factor of four increase in N-16 shine produced through the steam lines into the turbine building. For normally occupied areas, sufficient shielding is provided to protect from N-16 shine. In areas which may be occupied temporarily for specific maintenance or surveillance tasks and where additional shielding is not appropriate (for the surveillance function) or deemed reasonable, the HWC injection can be stopped causing the N-16 shine to decrease to within normal operating BWR limits within 90 seconds and thus permitting those actions needed. Overall, it is estimated that the effective dose rate for work at power will be slightly over two thirds the typical rate or 4.0 mRem/hr.

GE Nuclear Energy

ABWR

Date 3-26-92

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Subject Amendment 21 Revision to Section 12.4

Message

17mRem/hr. In the ABWR, the primary source of radiation exposure, the recirculation lines and pumps, have been removed. Overall the reduction in drywell dose level is for these types of maintenance is expected to be a factor of two or 9mRem/hr. Overhead tracks and in place removal equipment is provided in the ABWR for an estimated person-hour reduction to 1,150 person-hour per year broken down into 200 person-hours for 18 SRV maintenance at 6mRem/hr, 200 person-hours per year to pull and replace 3 RIPs with one heat exchanger at 20mRem/hr, and the remainder on miscellaneous valves at 4.5mRem/hr.

(3) Control rod drive maintenance is significantly reduced in the ABWR with the introduction of fine motion control rod drives (FMCRD). Based upon European experience, two FMCRDs will be replaced and repaired per outage along with twenty motors. Estimated work will consist of 64 person-hours under vessel preparation, 40 person-hours FMCRD removal and reinstallation, 200 person hours motor removal and installation, and 64 person-hours cleanup. Typical under vessel effective dose rates are 17mRem/hr but because of the removal of the recirculation pumps and lines dose rate have been reduced to 6.5mRem/hr.

(4) The LPRM/TIP system assumes the servicing of two sensors per year and is based upon a total of 200 person-hours per year at an effective dose rate of 50mRem/hr which is typical for BWR operations.

(5) Inservice inspection consists of primarily NDE examination of vessel and piping systems and welds. Typical BWR values are 2400 person-hours per year at 12mRem/hr effective exposure rate. ABWR inservice inspection is estimated based upon the following:

Elimination of recirculation lines and pumps with the following savings:

Elimination of 14 nozzle inspections at 2 per year, saving 360 person hours

Elimination of shield penetration and shield plug removal saving 240 person hours per year

Elimination of weld inspection on recirculation lines estimated at 240 person hour per year

Reduction in drywell dose by 50% based upon the assumption that the contact dose rate on the feedwater line is less than half the contact dose rate on the typical BWR recirculation line. Hence at equal distances from the line, the total general drywell dose rate which is dominated by the recirculation and feedwater lines will be less than half what is typically seen with recirculation lines.

Overall it is estimated that by use of automated turtles for inspection person-hour expended in ISI will be reduced by a factor of two.

The ABWR uses a forged ring pressure vessel in comparison to older plate welded vessels reducing the total vessel weld length inspection by 30% and the total weld inspection in the drywell by 10%.

The ABWR design incorporates specific access panels and shield doors into required inspection areas permitting easy bypass of insulation areas resulting in an estimated person hour savings of 120 person hours.

Overall person-hours reduction is 1,200 person-hours at approximately half the typical effective dose rate or 5.5mRem/hr.

(6) Other drywell work includes items such as minor valve maintenance, instrumentation work, and all other drywell work. These miscellaneous tasks in the drywell consume on the average 5,500 person-hours per year in a radiation field of 17mRem/hr. However, this average is a combination of some specific higher radiation tasks such as work on recirculation lines (involving snubbers, weld inspection, etc) and many lower radiation tasks such as work on drywell coolers. Overall reduction in this effort due to ABWR design improvements are:

Significant savings in total hours are estimated due to removal of the recirculation lines with miscellaneous recirculation line work such as line snubbers, fewer drywell cooling units, and less assembly/disassembly work on insulation due to the use of automated units. Overall it is estimated that 2,000 person hours savings can be made.

Overall reduction in the drywell radiation due to removal of the recirculation system results in the reduction of the overall upper drywell dose rate to 1.8mRem/hr and the lower drywell dose rate to 5.6mRem/hr since the components involved such as drywell coolers typically do not carry radioactive inventory. Assuming that of the remaining 3,500 person-hour, 2,000 is upper drywell work and 1,500 is lower drywell work at their respective effective dose rates.

12.4.2 Reactor Building Dose

The following provides the basis by which the reactor building dose estimates for occupational exposure were made.

(1) Vessel access and reassembly typically requires 4500 person-hours of work at an effective dose rate of 3 mr/hr. The ABWR work will involve the use of a stud tensioner for a 96 bolt top head. The projected time to remove 96 bolts with this equipment is between 600 to 1200 person-hours. Due to the larger ABWR vessel and expected reduced water contamination with the improved clean up system, the estimated projected effective dose rate is 1.5 mRem/hr.

(2) ABWR refuelling is accomplished via an automated refuelling bridge. All operations for refuelling are accomplished from an enclosed automation center off the refuelling floor. Time for refuelling is reduced from a typical 4,400 person-hours down to 2,000 person and from an effective dose rate of 2.5 mRem/hr to less than 0.2mRem/hr.

(3) RHR/CUW maintenance work consists of inspections for two pumps per year in each system. In the RHR system this consumes 150 person-hours per year at an effective dose rate of 40mRem/hr. In the CUW system this typically uses 1400 person-hours per year at an effective dose rate of 14mRem/hr. ABWR will use canned pumps for both system with an estimated reduction in maintenance to 100 person-hours per pump. With improved water chemistry and overall reductions in reactor water concentrations due to the two percent cleanup system the effective dose rate is estimated at twenty percent of the typical value for these system.

(4) FMCRD rebuilding estimates are taken from similar work done in Europe since no significant U.S. data exists to date. Two drives will be rebuilt at an effective dose rate of 4.5 mRem/hr and 30-60 hours per drive.

(5) Instrumentation work typically requires 1,000 person-hours of work per year at an effective dose rate of 5.0 mRem/hr. ABWR should take about the same effort in instrumentation, however because of the increased emphasis and improved water chemistry systems, should reduce the effective dose rate to two-thirds the typical value or 3.0mRem/hr.

(6) All other work in the reactor building typically takes 7,400 person-hours per year at an effective dose rate of 2.8mRem/hr. This work includes all valve work, RIP rebuild work, minor maintenance, and CRD hydraulic line work. The major task in this area is the hydraulic control units which require 5,000 person-hours per year at an effective dose rate of 3.3mRem/hr. With the use of the FMCRD units, an additional savings of 2,000 person-hours is anticipated. In addition, the ABWR reactor building has been designed to provide for ease of maintenance with overhead lifts, coordinated hatch ways and ample space to maintain in place equipment. In addition, with the exception of one tank and the pressure vessel, all the equipment in the reactor building is removable with those pieces which can be expected to be moved being palatalized. Because of these factors, an overall reduction in work of 1,000 person-hours is estimated. Because of the improved water chemistry the overall effective dose rate is anticipated at one-half the typical BWR dose rate.

12.4.3 Radwaste Building Dose

Radwaste building work consists of pump and valve maintenance, shipment handling, radwaste management, and general clean up activity. Typically, 6,700 hours are expended per year at an effective dose rate of 5.5mRem/hr. The ABWR radwaste building is designed along the same lines as newer radwaste facilities overseas. The building incorporates enhanced remote control and shielding for handling of resin materials which is expected to reduce overall maintenance by 1500 to 2000 hour per year at significantly reduced dose levels. In addition, radwaste pumps for ABWR are expected to utilize air driven, rack mounted pumps. Such pumps which are designed to handle slurries have been proven to show much longer life times between maintenance and being basically a very small portable pump, can be readily replaced. Replaced pumps are then subject to intense chemical decontamination prior to maintenance and repair. Overseas utilities have reported occupational exposures typically less than 1 person-rem per year using this design. For ABWR, it is then assumed that the maintenance effort expended per year is reduced by 2,000 person hours from 6,700 to 4,700 person hours due to the introduction of automated equipment. An additional reduction of 500 person hours down to 4,200 person hours is assumed based upon the use of air pumps as specified above. The overall radiation field to which the worker is exposed on the average is then expected to be reduced from 5.5mRem/hr to 2.5mRem/hr since most of the high radiation tasks are eliminated by automation or remoting the tasks or in the case of the air pumps reduced by decontamination at separate facilities prior to pump maintenance.

12.4.4 Turbine Building Dose

(1) Typical BWR valve maintenance in the turbine building uses 1,150 hours per year at an effective dose rate of 9.5mRem/hr. The valve maintenance requirements for ABWR do not vary significantly over current plants, therefore the total hours for this type of work is assumed as approximately the same excepting minor adjustments for improved valves, maintenance jigs, and automated devices will lower the estimated maintenance time to 1,000 hours. In the ABWR, the estimated effective radiation field of 3.9 mRem/hr for turbine building work is expected to be less than half the typical dose rate of 9.5mRem/hr due to the use of newer fuels which are ~~less~~ more resistant to pin size leaks. The radiation fields in the turbine hall during maintenance are a combination of contamination from fission products from the fuel and corrosion products from the vessel and piping. Offgas measurements of the performance of the newer fuels when operated under proper water chemistry standards (required for ABWR) have shown fission product releases and order of magnitude less than older fuels. Likewise the ABWR has placed stringent controls over material usage especially in the vessel and other high temperature components to minimize corrosion product releases.

(2) In a similar fashion the turbine maintenance work typically requires 18,500 hours of work at an effective dose rate of 0.3mRem/hr. With additional operational improvements in automating turbine maintenance, overall work is estimated to be reduced to 15,500 hours. The effective dose rate for the turbine is not expected to be as sensitive to fuel performance as will the turbines but is estimated to reflect a decrease in dose to 0.2mRem/hr for turbine overhaul work.

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12.4.5 Work at Power

Work at power typically requires 5,000 hours per year at an effective dose rate of 6.6mRem/hr for the BWR. This category covers literally all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment, and minor equipment repair. Overall the ABWR has been designed with more automated and remotted equipment. It is expected that items of routine monitoring will be performed by camera or additional instrumentation. Most equipment in ABWR is palatalized which permits quick and easy replacement and removal for decontamination and repair. Therefore a reduction in actual hours need at power is estimate at 1,000 hours less than the typical value. In the area of effective dose rate, the ABWR is expected to have significantly lower general radiation levels over current plants owing to more stringent water chemistry controls, a full flow condensate flow system, a 2% clean up water program, titanium condenser tubes, Fe feedwater control, and low cobalt usage. In addition the ABWR has in the basic design, compartmentalized all major pieces of equipment so that any piece of equipment can be maintained or removed for maintenance without affecting the normal plant operations. This design concept thereby reduces radiation exposure to personnel maintaining or testing one piece of equipment from both shine and airborne contamination from other equipment. Finally, the ABWR has incorporated in the basic design the use of Hydrogen Water Chemistry (HWC) and the additional shielding necessary to protect from the factor of four increase in N-16 shine produced through the steam lines into the turbine building. For normally occupied areas, sufficient shielding is provided to protect from N-16 shine. In areas which may be occupied temporarily for specific maintenance or surveillance tasks and where additional shielding is not appropriate (for the surveillance function) or deemed reasonable, the HWC injection can be stopped causing the N-16 shine to decrease to within normal operating BWR limits within 90 seconds and thus permitting those actions needed. Overall, it is estimated that the effective dose rate for work at power will be slightly over two thirds the typical rate or 4.0 mRem/hr.

12.4.6 References

1. Knecht, P.D., *BWR/6 Drywell and Containment Maintenance and Testing Access Time Estimates*, GE Report NEDE-23819, May 1978.

2. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Radwaste Building*, GE Report NEDE-23996-2, May 1979.

3. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Auxiliary and Fuel Buildings*, GE Report NEDE-23996-1, May 1979.

4. *Study of Advanced BWR Features, Plant Definition/Feasibility Results*, Volume III, Appendix Part G, GE NEDE-24679, Oct 1979.

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Other	12.4.1(6)	3,500	3.5	12.3
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Refueling	12.4.2(2)	2,000	0.2	0.4
RHR/CUW	12.4.2(3)	400	8.0	3.2
FMCRD	12.4.2(4)	120	4.5	0.5
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Other	12.4.2(6)	4,400	1.5	6.6
Total		9,120		15.5
Radwaste Building	12.4.3	4,200	2.5	10.5
Turbine Building				
Valve Maint	12.4.4(1)	1,000	3.9	3.9
Turbine Ovrhl	12.4.4(2)	15,500	0.2	3.1
Condensate	12.4.4(3)	1,000	3.5	3.5
Other	12.4.4(4)	11,800	0.1	1.2
Total		29,300		11.7
Work at Power	12.4.5	4,000	4.0	16.0
Totals		**48,120**		**99.9**