Consumers Power
POWERING MICHIGAN'S PROGRESS

PALISADES NUCLEAR PLANT
ENGINEERING ANALYSIS COVER SHEET

EA-A-NL-91-169-01

Total Number of Sheets 14

Title Offsite Dose Calculations of Fuel Handling Accident

INITIATION AND REVIEW

Calculation Status	Preliminary	Pending	Final	Superseded
	☐	☐	X	☐

Rev #	Description	Initiated By	Initiated Date	Init Appd By	Review Method: Alt Calc	Review Method: Detail Review	Review Method: Qual Test	Technically Reviewed By	Technically Reviewed Date	Revr Appd By	CPCo Appd
0	Original Issue	SD McElroy [signature]	12/17/92	TCD 12/17/92	✓	✓		PAHarden [signature]	12/17/92	TCD 12/17/92	
1	Revise Input 4.4 to reflect appropriate Decay Heat Fraction	SD McElroy [signature]	4/13/93	TCD		✓		PAHarden [signature]	4/13/93	TCD	
2	Revised Input 4.4 to clarify	SD McElroy [signature]	4/15/93	[illegible]		✓		PAHarden [signature]	4/15/93	[illegible]	

PURPOSE:

The purpose of this analysis is to demonstrate that the offsite radiological doses of a Fuel Handling Accident (FHA), using the Regulatory Guide 1.25 source term, will be within the limits of 10 CFR 100 as defined in the Standard Review Plan 15.7.4. This analysis accounts for increased radial peaking for Cycle 11, as well as potential future increases in radial peaking factor up to a value of 1.8 for the peak assembly.

SUMMARY OF RESULTS:

The offsite doses from a fuel handling accident were calculated, bounding the maximum radial peaking factors in Cycle 11 and beyond. This was done to accommodate an increase in the peaking factor to 1.76 in Cycle 11. The analysis followed the guidelines and assumptions of Regulatory Guide 1.25 and the Standard Review Plan. The resultant doses were calculated to be 41.19 rem thyroid and 0.12 rem whole body for 0 to 2 hours at the site boundary, and 7.23 rem thyroid and 0.02 rem whole body for 30 days at the low population zone distance. These calculated doses are well within the limits of 10 CFR 100, as interpreted by the Standard Review Plan 15.7.4.

Table of Contents

1.0 OBJECTIVE

The purpose of this analysis is to demonstrate that the offsite radiological doses of a Fuel Handling Accident (FHA), using the Regulatory Guide 1.25 source term, will be within the limits of 10 CFR 100 as defined in the Standard Review Plan [Ref. 2.1]. This analysis accounts for increased radial peaking for Cycle 11, as well as potential future increases in radial peaking factors up to a value of 1.8.

2.0 REFERENCES

2.1 Regulatory Guide 1.25 Rev 2, "Assumptions Used For Evaluating The Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 1972.

2.2 NUREG-0800, USNRC Standard Review Plan, Section 15.7.4 Rev 1, "Radiological Consequences of Fuel Handling Accidents.

2.3 Letter from D.L. Ziemann (NRC) to D. Bixel (CPCo) dated June 21, 1979. Subject: Safety Evaluation by the Office of Nuclear Reactor Regulation Regarding the Fuel Handling Accident Inside Containment. Cart/Frame: 2600/0540.

2.4 Letter from D.L. Ziemann (NRC) to D.P. Hoffman (CPCo) dated February 5, 1980. Subject: XV-20 - Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment). Cart/Frame: 2614/0667.

2.5 NEDO-24782, "BWR Owners' Group NUREG-0578 Implementation: Analysis and Positions For Plant Unique Submittals," General Electric, August 1980.

2.6 NUREG/CR-1413 ORNL/NUREG-70, "A Radionuclide Decay Data Base - Index and Summary Table," Oak Ridge National Laboratory, May 1980.

2.7 NUREG/CR-5009 PN-6258, "Assessment of the Use of Extended Burnup Fuel in Lightwater Power Reactors," Pacific Northwest Laboratory.

2.8 ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," July 1978.

2.9 Regulatory Guide 1.4 Rev 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," June 1974.

2.10 Palisades Plant Final Safety Analysis Report.

2.11 Palisades Plant Technical Specifications, Section 3.8.3, through Amendment 153.

2.12 EMF-92-178, "Palisades Cycle 11: Disposition and Analysis of Standard Review Plan Chapter 15 Events," Siemens Power Corporation, November 1992.

2.13 E-PAH-92-03, "Core Decay Heat Release Fractions Using NRC Branch Technical Position ASB 9-2"

2.14 Letter from H.G. Shaw (Siemens) to R.J. Gerling dated October 6, 1992. Subject: Fuel Internal Rod Pressure at End of Life.

3.0 BACKGROUND

The FHA (Fuel Handling Accident) analysis is the bounding radiological consequence analysis for the dropping of a fuel assembly. For an FHA, the source terms of Regulatory Guide 1.25 [Ref. 2.2] and Standard Review Plan (SRP) 15.7.4 [Ref. 2.1] are used. This source term amounts to 10% of the assembly iodine, krypton, and xenon found in the gap between the fuel pins and the cladding escaping to the outside atmosphere. The exception to this is Kr-85, where 30% escapes to the outside atmosphere. NUREG/CR-5009 updates the value of escaping I-131 to 12% due to higher fuel burnup.

The limits for offsite dose are 300 rem thyroid and 25 rem whole body, as found in 10 CFR 100. The Standard Review Plan 15.7.4 [Ref. 2.2] sets the acceptance criteria for a FHA as "appropriately within the guidelines" of 10 CFR 100 and gives the value of 25% of the 10 CFR 100 limits, or 75 rem thyroid and 6.25 rem whole body. In Reference 2.3, the dose of 91 rem thyroid is given and deemed acceptable. In Reference 2.4, the limit is given as 100 rem thyroid for Palisades Nuclear Plant. This value is the licensing basis thyroid dose limit for a FHA at Palisades.

4.0 ANALYSIS INPUT

4.1 The breathing rate for offsite doses is 3.47E-4 m^3/sec in accordance with Reference 2.9. This is the breathing rate given for a person offsite during the first eight hours of the accident. This is also the maximum breathing rate given, so it is deemed conservative.

4.2 The number of fuel assemblies in the core, 204, is found in Reference 2.10, Section 3.3.1.

4.2 The rated core thermal power, 2530 MW_{th} is from Reference 2.10.

4.3 The assembly radial peaking factor of 1.8 was used for this analysis. The value of 1.76 is the factor for cycle 11 and can be found in Reference 2.12, but 1.8 was used to bound radial peaking.

4.4 The decay heat fractional energy release 2 days after shutdown, based on Reference 2.13, is approximately 0.0049. This results in a Linear Heat Generation Rate of 0.026 kW/ft. This LHGR is less than the 0.05 kW/ft used to calculate the fuel rod pressure in Reference 2.14. Thus, the fuel rod pressure provided in Reference 2.14 is conservative. Rev2

4.5 The fuel rod gas pressure is less than 1200 psig based on Reference 2.14. This meets the requirements needed to make the Reference 2.1 assumptions valid. Reference 2.14 can be found as Attachment 8.1.

4.6 The Pool Decontamination Factors are 100 for iodine and 1 for all other nuclides. This is taken from Reference 2.1 based on Reference 2.14.

4.7 The radionuclide activities for one day after reactor shutdown were found in Reference 2.5 and are listed in Table 1. For the iodine isotopes, the values in Table 1 are doubled because Reference 2.5 gives 50% of the value needed.

4.8 The radionuclide half lives are found in Reference 2.6 and are listed in Table 1. The values found in Reference 2.6 were converted to minutes before being input into the table.

4.9 The dose conversion factors for the radionuclides are from Reference 2.8, and are listed in Table 1. The values found were converted to new units before being input into the table.

Table 1

Nuclide	S_i (Ci/MW_t)	$t_{1/2}$ (min)	Dose Conversion Factors See Note
I-131	2.696E+4	11577.6	1073000
I-132	7.204E+2	138	6290
I-133	2.178E+4	1248	181300
I-134	3.040E-4	52.6	1073
I-135	4.778E+3	396.6	31450
Kr-83m	6.039E+0	109.8	.000003
Kr-85m	1.489E+2	268.8	.03031
Kr-85	2.999E+2	5638291	.000473
Kr-87	2.295E-2	76.3	.1447
Kr-88	4.452E+1	170.4	.369
Xe-131m	1.708E+2	17049.6	.001324
Xe-133m	1.547E+3	3153.6	.005375
Xe-133	5.171E+4	7552.8	.006259
Xe-135m	6.693E+2	15.36	.07647
Xe-135	7.388E+3	546.6	.04676
Xe-138	4.830E-27	14.13	.1969

NOTE: The Dose Conversion Factors are (Rem/Ci) for Iodine and (Rem/sec)/(Ci/m^3) for all other nuclides.

5.0 ASSUMPTIONS

5.1 The core is assumed to have run at 102% of rated power, or 2580.6 MW_t.

5.2 The FHA occurs inside containment, 48 hours after shutdown, as per Reference 2.12. This creates the most severe consequences for a fuel handling accident.

5.3 One fuel assembly is completely damaged as a result of the fuel handling accident. This is consistent with Reference 2.1.

5.4 The fuel assembly with the peak inventory is the one damaged, consistent with Reference 2.1

5.5 Consistent with References 2.1 and 2.7, 30% of the Kr-85 found in the gap between the cladding and the fuel is released, along with 12% of the I-131 and 10% of the remaining iodine, krypton, and xenon.

5.6 The release of radionuclides fails to trip high-radiation alarms and escapes from containment, unfiltered, into the environment. There is a direct flow path through one of the clean waste receiver tanks as long as a containment isolation signal is not generated

5.7 The release is over a two hour period as per Reference 2.1.

6.0 ANALYSIS

6.1 Fuel Activity

The first part of this analysis was to determine the activity of each radionuclide released to the reactor cavity water from the pellet-clad gap of the damaged assembly. First, the core inventory of each radionuclide per MW_t one day after shutdown was found in Reference 2.5. The value was divided by 204 (number of fuel assemblies in the core) to find the activity per MW_t for the one damaged assembly. This was multiplied by the 102% rated power of 2580.6 MW_t as well as the assembly peaking factor of 1.8 to find the activity in the peak assembly one day after shutdown. This value is then corrected for the extra day of decay using the radioactive decay equation. This value, unique to each radionuclide, is multiplied by the fraction of that nuclide in the pellet-clad gap that is released to yield the total nuclide release to the water from the damaged assembly. The equation for this can be written as:

$$A_i = F_g\left(\frac{S_i\ P\ F_r}{204}\right) e^{\frac{-LN(2)1440}{t_{1/2}}}$$

where

- A_i = activity released to water from pellet-clad gap, Ci
- S_i = Source term of individual isotope, Ci/MW_t
- F_g = fraction of inventory in the pellet-clad gap inventory released
- P = power of plant at 102% of rated power, MW_t
- F_r = peaking factor of peak assembly
- $t_{1/2}$ = half life of nuclide, in minutes

6.2 Activity Released Outside Containment

The next step in this analysis was to find the activity that is released from the water and into the outside air. This was done using the overall decontamination factor for the water. For all iodine isotopes, D_{eff} is 100, meaning that one hundredth, or 1% of all iodine released from the pellet clad gap makes it to the outside air. For all other isotopes, D_{eff} is set to 1, meaning that all of the activity released from the pellet clad gap makes it to the outside air. The denotation Q is used to represent the activity of each nuclide released to the outside air.

From there, the dose received due to each isotope was computed. This was done by multiplying the Q value for each isotope by the dose conversion factor (DCF_{eff}). Then it was multiplied by the Atmospheric Dispersion Factor. This number is based on the distance away from the location, so two values were used: one for the site boundary and one for the low population zone. For the iodine isotopes, the value produced was multiplied by the breathing rate, since the calculated dose from iodine is thyroid dose due to inhalation. This is represented by the following equations:

$$H_{wb} = Q\ DCF\ \frac{\chi}{Q}$$

for iodine isotopes and:

$$H_{thy} = \left(Q\ DCF\ \frac{\chi}{Q}\right)\ BR$$

for all other radionuclides where

H_{thy}	=	dose to thyroid from inhalation of iodine
H_{wb}	=	dose to whole body from xenon and krypton exposure
χ/Q	=	atmospheric dispersion factor for location
DCF	=	dose conversion factor for isotope
Q	=	activity released outside containment
BR	=	breathing rate of an individual subject to radionuclide exposure

The values of H_{thy} were then summed separately for the site boundary (sb) and the low population zone (lpz). The same was done for H.

The values found as maximum doses are:

H_{thy}(sb)	=	41.19 rem
H_{thy}(lpz)	=	7.23 rem
H_{wb}(sb)	=	0.12 rem
H_{wb}(lpz)	=	0.02 rem

These values are well within the values given as limits by References 2.2 and 2.4.

7.0 SUMMARY

This analysis was performed to demonstrate that the radiological consequences of a fuel handling accident are within the limits of References 2.2 and 2.4. The activity in the pellet-clad gap of the peak fuel assembly at the time of the accident was found. Then, the fraction of activity that is would be released is accounted for. The decontaminating ability of the water was then factored in to provide the activity released outside containment of each radionuclide involved. Finally, the dose related to this release was found at the site boundary as well as the low population zone, and the thyroid dose was corrected for the breathing rate of the person subject to the exposure. The exposures calculated for each isotope involved were them summed to find the overall exposures encountered.

The doses calculated at the site boundary were 41.19 rem thyroid and 0.12 rem whole body dose. The doses for the low population zone were calculated to be 7.23 rem thyroid and 0.02 rem whole body. These are well inside the limits established and are considered acceptable.

List of Attachments

8.1 Reference 2.14: Letter from H.G. Show (Siemens) to R.J. Gerling dated October 6, 1992. Subject: Fuel Internal Rod Pressure at End of Life.

8.2 Technical Review Checklist

8.3 Engineering Analysis Checklist

8.4 Document Review Sheet

SIEMENS

October 6, 1992
HGS:327:92

Mr. R. J. Gerling
Consumers Power Company
Palisades Nuclear Power Plant
27780 Blue Star Memorial Hwy.
Covert, MI 49043-9530

Dear Bud:

In response to information requested by CPCo. a RODEX2 gas pressure analysis for a fuel rod stored in the Palisades spent fuel storage pool has been performed.

Key input variables for the analysis are the FSAR design basis of 150°F for the spent fuel storage pool and an LHGR = 0.05 kW/ft. ANSI/ANS-5.1-1979 indicates that the ratio of the decay heat rate to the heat rate for infinite operation is 0.0028 two weeks following shutdown. Multiplying 0.0028 by the core average LHGR (5.23 kW/ft) gives 0.014 kW/ft; thus, the LHGR of 0.05 kW/ft is conservative. The hot operating end of life internal rod pressure was 2698 psia.

The analysis indicates that gas pressure in the fuel rod stored in the pool will be less than 1170 psia. For reference purposes, the SPC calculation number is E-5059-337-6X.

If you have any questions, please contact Mr. Jim Hulsman.

Very truly yours,

Bud

H. G. Shaw
Contract Administrator

tim

c: T. C. Bordine, CPCo Jackson
[redacted] Palisades
J. W. Hulsman, SPC
S. F. Pierce, CPCo Palisades

Siemens Power Corporation
Nuclear Division - Headquarters
155 108th Avenue NE, PO Box 90777 Bellevue, WA 98009-0777 Tel: (206) 453-4300 Fax: (206) 453-4446

TECHNICAL REVIEW CHECKLIST

EA - S-NL-92-169-01 REV. 0

This checklist provides guidance for the review of engineering analyses. Answer questions Yes or No, or N/A if they do not apply. Document all comments on a 3110 Form. Satisfactory resolution of comments and completion of this checklist is noted by the Technically Reviewed signature on the Initiation and Review record block of Form 3519.

		(Y, N, N/A)
1.	Have the proper input codes, standards and design principles been specified?	Y
2.	Have the input codes, standards and design principles been properly applied?	Y
3.	Are all inputs and assumptions valid and the basis for their use documented?	Y
4.	Is Vendor information used as input addressed correctly in the analysis?	Y
5.	If the analysis argument departs from Vendor Information/Recommendations, is the departure justification documented?	N/A
6.	Are assumptions accurately described and reasonable?	Y
7.	Has the use of engineering judgament been documented and justified?	Y
8.	Are all constants, variables and formulas correct and properly applied?	Y
9.	Have any minor (insignificant) errors been identified? If yes; Identify on the 3110 Form and justify their insignificance.	See attached 3110 Form
10.	Does analysis involve welding? If Yes; verify the following information is accurately represented on the analysis drawing (Output document). • Type of Weld • Size of Weld • Material Being Joined • Thickness of Material Being Joined • Location of Weld(s) • Appropriate Weld Symbology	N/A
11.	Has the objective of the analysis been met?	Y
12.	Have administrative requirements such as numbering and format been satisfied?	Y

12/17/92

Form 3698 9-89

PALISADES NUCLEAR PLANT

ENGINEERING ANALYSIS CHECKLIST

Items Affected By This EA	Affected Yes	Affected No	Revision Required	Identify*	Closeout
1. Other EAs	☐	☒			
2. Design Documents Elec E-38 through E-49	☐	☒			
3. Design Documents Mech M259, M664, M665	☐	☒			
4.0 LICENSING DOCUMENTS					
4.1 Final Safety Analysis Report (FSAR)	☒	☐	Yes	FSAR 14.19	FC-934
4.2 Technical Specifications	☐	☒			
4.3 Standing Order 54	☐	☒			
5.0 PROCEDURES					
5.1 Administrative Procedures	☐	☒			
5.2 Working Procedures	☐	☒			
5.3 Tech Spec Surveillance Procedures	☐	☒			
6.0 OTHER DOCUMENTS					
6.1 Q-List	☐	☒			
6.2 Plant Drawings	☐	☒			
6.3 Equipment Data Base	☐	☒			
6.4 Spare Parts (Stock/MMS)	☐	☒			
6.5 Fire Protection Program Report (FPPR)	☐	☒			
6.6 Design Basis Documents	☐	☒			
6.7 Operating Checklists	☐	☒			
6.8 SPCC/PIPP Oil and Hazardous Material Spill Prevention Plan	☐	☒			
6.9 EEQ Documents	☐	☒			

Do any of the following documents need to be generated as a result of this EA:

	Yes	No	
1. Corrective Action Document?	☐	☒	Reference ______
2. Safety Evaluation?	☒	☐	Reference ______
3. EEQ Evaluation Sheet?	☐	☒	Reference ______
Is PRC Review of this EA Required?	☐	☒	

Completed By Scott McElfresh

Date 12/17/92

*Identify Section, No, Drawing, Document, etc.

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NUCLEAR OPERATIONS DEPARTMENT
Document Review Sheet

Document Title	Document Number	Revision Number	Page
OFFSITE DOSE CALCULATIONS OF FUEL HANDLING ACCIDENT	EA-A-NL-91-169-01	0	Page 1 of 2

Item Number	Page and/or Section Number	Comments	Response or Resolution
1	General	Various typographical & grammatical errors shown on the marked up copy	Corrected
2	Pg 4/§3.0	List the dose limits stated in each of the given references for clarity.	Done
3	Pg 5/Table 2	There are two typo's in the table. The DCF for I-134 is 1073, and the DCF for Xe-133 is 0.006259.	Corrected
4	Pg 5/Table 2	The source term values for xenon in the table are incorrect. You have the values at shutdown for xenon, but the values one day after shutdown for everything else. The values at one day are Xe-131m = 1.768E+2, Xe-133m = 1.547E+3, Xe-133 = 5.171E+4, Xe-135m = 6.693E-2, Xe-135 = 7.388E+3, and Xe-138 = 4.830E-27. This could explain the slight difference in calculated whole body doses. Alternate calcs are attached.	Corrected

All comments resolved.
[signature] 12/17/92

Reviewer	Organization	Date	Review Coordinator	Date	Document Sponsor	Date
[signature]	RASA	12/16/92			Scott McGraw	12/17/92

Form 3110 10 91

	t=24 hrs Ci/MW	t=24 hrs Aa (Ci)	(1/min) Lambda	t=48 Aa (Ci)	Q (Ci)	Site Boundary Thyroid (Rem)	Site Boundary Whole Body (Rem)	Low Population Zone Thyroid (Rem)	Low Population Zone Whole Body (Rem)
Kr-83m	6.039E+00	137.5	6.313E-03	0.015	0.001549		8.8E-13		1.5E-13
Kr-85m	1.489E+02	3390.5	2.579E-03	82.678	8.267843		0.000038		0.000006
Kr-85	2.999E+02	6828.7	1.230E-07	6827.514	2048.254		0.000150		0.000026
Kr-87	2.295E-02	0.5	9.084E-03	0.000	0.000000		2.4E-12		4.3E-13
Kr-88	4.452E+01	1013.7	4.068E-03	2.896	0.289638		0.000016		0.000002
Xe-131m	1.708E+02	3889.1	4.065E-05	3667.998	366.7997		0.000075		0.000013
Xe-133m	1.547E+03	35225.2	2.198E-04	25668.112	2566.811		0.002138		0.000375
Xe-133	5.171E+04	1177436.7	9.177E-05	1031682.524	103168.2		0.100088		0.017563
Xe-135m	6.693E+02	15240.0	4.513E-02	0.000	9.1E-26		1.1E-30		1.9E-31
Xe-135	7.388E+03	168224.8	1.268E-03	27095.860	2709.585		0.019638		0.003446
Xe-138	4.830E-27	0.0	4.906E-02	0.000	2.3E-57		7.0E-62		1.2E-62
I-131	2.696E+04	613879.2	5.987E-05	563172.157	675.8065	39.00168		6.844165	
I-132	7.204E+02	16403.5	5.023E-03	11.848	0.011847	0.000004		0.000000	
I-133	2.178E+04	495930.6	5.554E-04	222885.904	222.8859	2.173409		0.381398	
I-134	3.046E-04	0.0	1.318E-02	0.000	4.0E-14	2.3E-18		4.0E-19	
I-135	4.778E+03	108795.1	1.748E-03	8778.854	8.778854	0.014849		0.002605	
						41.18994	0.122146	7.228170	0.021434

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NUCLEAR OPERATIONS DEPARTMENT
Document Review Sheet

Document Title	Document Number	Revision Number	Page
Offsite Dose Calculations of Fuel Handling Accident	EA-A-NL-91-169-01	1	Page 1 of 1

Item Number	Page and/or Section Number	Comments	Response or Resolution
		Based on a decay heat fraction of .0049 at 2 days after shutdown, LHR would be .0049 (5.23 kW/ft) = 0.026 kW/ft. This is approximately half of the LHR used to develop fuel rod pressure. No resolution required.	

Reviewer	Organization	Date	Review Coordinator	Date	Document Sponsor	Date
[signature]	RA SA	4/13/93	[signature]	4/13/93		

Form 3110 10-91

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NUCLEAR OPERATIONS DEPARTMENT
Document Review Sheet

Document Title	Document Number	Revision Number	Page
Offsite Dose Calculations of Fuel Handling Accident	[illegible]	2	1 of 1

Item Number	Page and/or Section Number	Comments	Response or Resolution
		No Comments	

Reviewer	Organization	Date	Review Coordinator	Date	Document Sponsor	Date
[illegible]	[illegible]	[illegible]				

Form 3110 10-91