file date 12/29/9

Appendix 10 Spent Fuel Heatup Time Assuming Adiabatic Conditions

In the preliminary study, the technical working group used a conservative calculation to demonstrate a minimum amount of time available to take unplanned emergency off-site evacuation actions. We calculated the time to heat a single fuel rod from 30 °C to 900 °C at various times following permanent shutdown without considering any heat losses. This is a very conservative, non-realistic calculation because some cooling through conduction, convention or radiation would occur and increase the time to heatup. However, it provides a measure of time that is a bounding value for all heatup scenarios, including scenarios such as having a flat material fall on top of an assembly. This information was not used in determining interim criteria.

The time available prior to a release is useful because the likelihood that personnel actions or offsite actions could be taken could affect the consequences of the event. Figures A11-1 and A11-2 show the increase in time that is required to heat a fuel pin from 30 °C to 900 °C as the time following shutdown increases given that all the heat is maintained in the fuel pin. For the calculations, the staff used a decay heat per assembly and divided it equally among the pins. It assumed a 9X9 assembly for the PWRs and a 17x17 assembly for the BWRs. The design assumptions values are provided in Table A11-1. The values for the progression of the calculation are provided in Tables A11-2 to A11-4. Decay heats were computed using an extrapolation of the decay power tables in NUREG/CR-5625 [Ref. 1]. The decay heat in NUREG/CR-5625 is based on ORIGEN calculations. The tables for the decay heat extend to burnups of 50 GWD/MTU for PWRs and 45 GWD/MTU for BWRs. The staff recognizes that the decay heat is only valid for values up to the maximum values in the tables, but the functional dependence of the decay power with respect to burnup for values in the table indicate that extrapolation may provide a reasonable estimate of the decay heat for burnup values beyond the limits of the tables. The BWR decay heat was calculated using a specific power of 26.178 MW/MTU. The PWR decay heat was calculated using a specific power of 37.482 MW/MTU. Both the PWR and BWR decay heats were calculated for a burnup of 60 GWD/MTU and include an uncertainty factor of 6 percent.

Reference:

1. Hermann, et.al., "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data", NUREG/CR-5625, September 1994.

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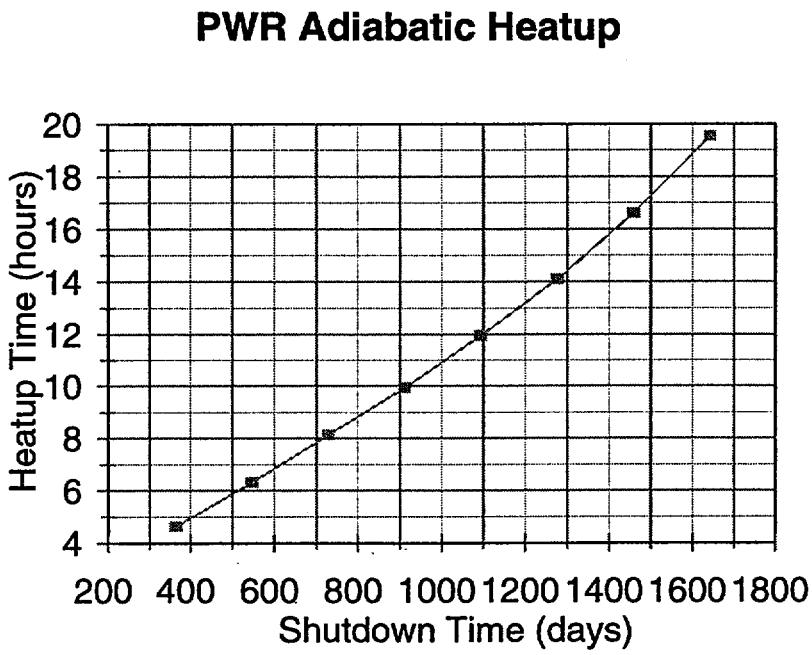

Figure A11-1

Adiabatic Heatup Time vs. Shutdown Time for a PWR

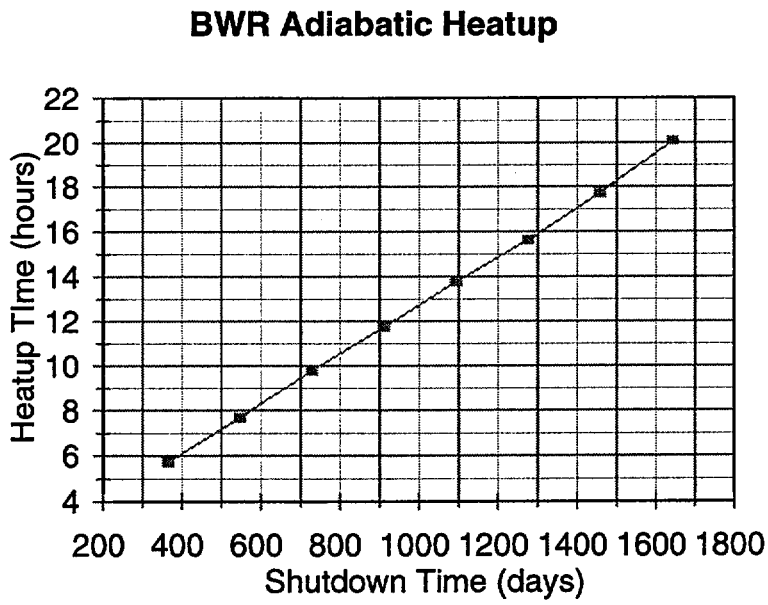

Figure A11-2

Adiabatic Heatup Time vs. Shutdown Time for a BWR

Table A11-1 Assumption Values for Calculation

	PWR		BWR	
burnup/assembly (MWD/MTU)	60000	60000		
MTU/assembly	0.47		0.17	
MWt/core	3400		3400	
assemblies/core	193		764	
MW/assembly		17.6		4.5
rods/assembly	(17x17) 264		(9x9) 72	
pellet diameter (m)	0.0081916		0.00906	
inside clad diameter (m)	0.0083276		0.00925	
outside clad diameter (m)	0.0094996		0.01077	
fuel length (m)	3.65		3.8	
rod peaking factor	1.166		1.166	
axial peaking factor	1.2		1.2	
fraction theoretical UO2	0.95		0.95	

Table A11-2 Decay Power in Kw/MTU

Year(s)	PWR Decay Power	BWR Decay Power
1	15.5561572	12.544759
1.5	11.416337106	9.3360165
2	8.85047	7.3451549
2.5	7.252431291	6.1212766
3	6.0268945746	5.2179396
3.5	5.1149641086	4.601295
4	4.341018	4.0575242
4.5	3.6841778116	3.578015

Table A11-3 Decay Power in Watts for One Assembly

(kW/MTU)*MTU*1000
Decay Power*0.47*E3 Decay Power*0.17*E3

Shutdown days	PWR Decay Power(W)	BWR Decay Power(W)
365	7311.393884	2132.609
547.5	5365.6784396	587.1228
730	4159.7209	1248.6763
912.5	3408.6427068	1040.617
1095	2832.6404501	887.04973
1277.5	2404.0331311	782.22016
1460	2040.27846	689.77911
1642.5	1731.5635715	608.26254

Table A11-4 Calculated Adiabatic Heatup Time from 30 C to 900 C in Hours for One Fuel Rod

Shutdown Days	PWR Heatup time (hrs)	BWR Heatup time (hrs)
365	4.64	5.73
547.5	6.32	7.7
730	8.15	9.79
912.5	9.94	11.75
1095	11.96	13.78
1277.5	14.1	15.64
1460	16.62	17.72
1642.5	19.57	20.11

Appendix 11 Stakeholder Interactions

The technical staff reviewed and evaluated available technical information and methods to use as the risk-informed, technical basis for reviewing decommissioning exemption requests and rulemaking related to EP, safeguards, indemnification, and other areas. When the draft report was released for public comment in June 1999, stakeholder's identified concerns which we addressed for inclusion in the final report. The early stakeholder input has improved the overall quality of the draft report. Provide below is a list of the meetings that were held with the stakeholders and comments could have been provided. To ensure that stakeholders are clear on how their input in the draft report was considered, we have provided a discussion regarding our consideration of new information or views that were presented by stakeholders in various technical areas.

Public meetings regarding the technical working group study:

March 17, 1999	Commission meeting in Rockville, MD
May 5, 1999	Stakeholder meeting with NRC staff in Rockville, MD
April 13, 1999	Stakeholder meeting with NRC staff in Rockville, MD
June 7, 1999	Stakeholder meeting with NRC staff in Rockville, MD
June 8, 1999	Stakeholder meeting with Sam Collins in Rockville, MD
July 15-16, 1999	Workshop on decommissioning plant spent fuel pool accident risk in Gaithersburg, MD
November 3, 1999	Stakeholder meeting with Sam Collins in Rockville, MD
November 5, 1999	ACRS meeting in Rockville, MD
November 8, 1999	Commission meeting in Rockville, MD
November 19, 1999	Stakeholder meeting with NRC staff in Rockville, MD

Probability Risk Assessment

An industry stakeholder raised the concern that the PRA was too conservative and that some of the assumptions made were unrealistic. We refined the PRA analysis and the results have been subjected to an independent technical review. The results of our review are found in Chapter 4 of this report.

Human Reliability Analysis

Industry stakeholders raised a concern that the June 1999 draft report did not give sufficient credit for operator actions in the area of human reliability analysis (HRA). Specifically, industry stated that the NRC draft report did not reflect the potential for actions such as self-checking, longer reaction times available, management oversight, design simplicity, second crew member check, additional shift attention in recovery, or additional cues causing increased attention.

We in turn, enlisted the support of HRA experts to refine the analysis contained in the June 1999 draft report. Furthermore, the HRA results were subjected to an independent technical review. A more detailed discussion of HRA, which also addresses the industry stakeholder concerns, is included in Chapter 4 of this report.

Heavy Loads

Industry stakeholders raised a concern that the heavy load risk assessment in the draft report did not give sufficient credit for NUREG-0612 actions and used the conservative upper bound values.

To address these concerns, we employed the use of more recent Navy data to requantify the fault tree, included the mean value estimate for compatibility with Regulatory Guide 1.174, and addressed industries voluntary commitment to Phase II of NUREG-0612. The final results were subjected to an independent technical review and are included in Chapter 4 of this report.

Seismic

To take credit for the seismic design margins existent in spent fuel pools, we sought an appropriate method to identify potential structural vulnerabilities without having to perform a detailed fragility review. At a public workshop on July 15-16, 1999, industry proposed development of a simple spent fuel pool seismic checklist as a way of assessing seismic vulnerabilities without performing quantifying analyses.

In a letter dated August 18, 1999, NEI proposed a "seismic checklist" for screening which we consider to be an excellent approach to plant-specific seismic assessments; however, some deficiencies have been identified in the checklist proposed by NEI. A discussion of the "seismic checklist" and our concerns are discussed in Chapter 4 of this report.

Criticality

A public stakeholder concluded that the June 1999 draft report did not address the potential for a criticality accident to occur in a SFP for a decommissioned plant.

Therefore, we examined the mechanisms by which a criticality accident could occur to assess the potential for criticality, evaluate its consequences, and assess the likelihood of an criticality event. Those results were subjected to an independent technical review where additional mechanisms were proposed and examined. The results of the analysis are included in Chapter 5 of this report.

Thermal Hydraulic Assessment

Industry stakeholders raised a concern that the thermal hydraulic assessment presented in the June 1999 draft report used overly conservative adiabatic heatup calculations and a maximum clad temperature that was too conservative compared to the zirconium ignition temperature.

We refined the thermal hydraulic analysis presented in the draft report and subjected those results to an independent technical review. The results of the analysis are included in Chapter 3 of this report.

Concrete Aging of the SFP

A public stakeholder raised a concern that the June 1999 draft report did not address the potential for concrete aging in SFP's at decommissioned plants.

At the July 16, 1999 workshop, we committed to discussing the issue of concrete strength over

time and the potential aging/degradation issue for SFP's. A discussion of this topic is found in Appendix 2 of this report.

Seismic Effects on SFP Integrity

A public stakeholder raised a concern that the June 1999 draft report did not examine vulnerabilities related to the transfer tube and other SFP design vulnerabilities resulting from a seismic event for decommissioned plants. A discussion of this topic is found in Appendix 2 of this report.

Partial Draindown and Exothermic Reaction of SFP

An industry stakeholder reported that we did not address or consider the implications of a partial draindown as being as serious or worse then a complete draindown. They also raised a concern that the draft report did not address the potential for a hydrogen explosion resulting from an exothermic reaction between steam and zirconium. A discussion of this topic is found in Chapter 3 of this report.

Safeguards

A public stakeholder stated that the draft report did not address the potential or threat for vehicle-borne bombs. We revised the draft report to include a discussion on this matter. A discussion of this topic is found in Chapter 6 of this report.

Impact of Decommissioning on Operating Units

A public stakeholder raised a concern that we did not consider the impacts on operating units of removing the water from a SFP at a decommissioning site, such as Millstone and SONGS. We revised the draft report to include a discussion on this matter. The results of our review are found in **[Chapter]** of this report.

Industry Information

An industry stakeholder suggested that we should include industry data on zirconium fires from Teledyne Wah Chang, a zirconium manufacturer. At the request of industry representatives, on Thursday, May 13, 1999, we contacted two employees of Teledyne Wah Chang by telephone to gain more information on zirconium fires. As a result of this call, references were sent to us that discussed zirconium cladding.

The results of both the call and the references enabled the working group to determine that the zirconium information presented by Teledyne Wah Chang was consistent with the working group findings.