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SUBJECT: Suppl info to 860912 ltr re application of amend to License NPF-22 concerning emergency Tech Spec change request. Safety evaluation encl.

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Harold W. Keiser
Vice President-Nuclear Operations
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September 12, 1986

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Project Director
BWR Project Directorate No. 3
Division of BWR Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
EMERGENCY TECHNICAL SPECIFICATION CHANGE REQUEST
PROPOSED AMENDMENT 41 TO LICENSE NO. NPF-22
SUPPLEMENTAL INFORMATION
PLA-2720 FILE R41-2 Docket No. 50-388

Dear Ms. Adensam:

As requested by the staff, this letter supplements information sent to you in our previous letter, PLA-2719, dated 9/12/86.

Attached is the safety evaluation (NL 86-005) of the alternate method for removal of decay heat when the unit is being defueled. Using the alternate method for removal of decay heat when refueling is bounded by the safety evaluation used when defueling since the decay heat is lower when refueling.

The following is a revised No Significant Hazards Evaluation of this proposed change.

NO SIGNIFICANT HAZARDS EVALUATION

I. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would allow SSES Unit 2 to follow a currently prescribed action statement upon entering OPERATIONAL CONDITION 5 instead of while already in OPERATIONAL CONDITION 5. Performing CORE ALTERATIONS is the basis of that transition and simply starting CORE ALTERATIONS instead of simply continuing CORE ALTERATIONS does not affect the probability or consequences of any accident previously analyzed for those conditions.

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Secondly, operation as described in this Emergency Technical Specification Change Request will involve movement of fuel; fuel handling accidents have been previously evaluated in FSAR Section 15.7.4. This proposed action will not involve any changes in fuel handling procedures, equipment, or coolant inventory. Thus, this change does not increase the probability or consequences of a fuel handling accident as previously evaluated in FSAR Section 15.7.4.

Additionally, the consequences of the proposed plant configuration for defueling were evaluated with respect to FSAR Appendix 9A, "Analysis for Non-Seismic Spent Fuel Pool Cooling Systems." This evaluation was performed in the attached Safety Evaluation NL 86-005. Part III of this analysis concluded that:

(1) The proposed plant configuration does not increase the probability of occurrence of a loss of fuel pool cooling. The events that could lead to a loss of fuel pool cooling, such as a seismic event, loss of service water, loss of power, loss of fuel pool pumps, etc., are all independent of the proposed plant configuration; the initiating events are independent of the fuel pool configuration.

(2) If a loss of fuel pool cooling were to occur during the proposed operations, the radiological consequences would be less severe than for the FSAR Appendix 9A event. The attached safety evaluation calculates a time to pool boiling for the proposed configuration of 46 hours; pool boiling occurs after 25 hours in the Appendix 9A event. This is significant since the activity release rate from the pool depends on the rate of evaporation (boiling rate). Also, the radiological consequences of the postulated event are proportional to the number of defective fuel pins (1% is assumed in the FSAR Appendix 9A analysis). Offgas radiation in Unit 2 indicates that less than 1 fuel rod had failed during Cycle 1; therefore, the radiological consequences of a loss of fuel pool cooling accident during the proposed configuration are bounded by the FSAR Appendix 9A event.

Thus, the proposed action does not involve an increase in the probability or consequences of a loss of fuel pool cooling accident.

II. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The only accidents of consequence for the proposed configuration are a fuel handling accident and any accident that would result in the inability to remove decay heat. Regardless of the initiating sequence of events, the consequences of any scenario resulting in the inability to remove decay heat are similar to and bounded by the FSAR Appendix 9A loss of fuel pool cooling event (as described in Part I, above). The fuel handling event analyzed in FSAR Section 15.7.4 is not different from that which could occur in this configuration. Therefore, there are no new accidents possible beyond those accidents previously analyzed in FSAR Section 15.7.4 and Appendix 9A.

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III. The proposed change does not involve a significant reduction in a margin of safety.

The answers expressed in I and II above indicate the insignificance of the role the mode change plays in terms of safety in this case. The margin of safety has not been significantly reduced by simply having an RHR loop available to support loading of the first bundle back into the core, after which OPERATIONAL CONDITION 5 would be entered, and the alternate decay heat removal method could be used. Also as described in the attached safety evaluation, the use of the alternate decay heat removal method does not reduce the margin of safety while going from OPERATIONAL CONDITION 5 to a defueled condition. Therefore, the margin of safety is not reduced while going from a defueled condition to OPERATIONAL CONDITION 5 since the attached safety evaluation demonstrates that this condition is bounded by existing FSAR analyses.

The following is additional information with respect to the schedule delay in restoring the RHR loop. When the original schedule was developed, it was anticipated that the work on the RHR 17A valve would take seven days to complete. The schedule was based on similar work that was completed on the Unit 1 RHR valves.

The commencement of work on the RHR valves was delayed two days due to problems with the refueling seal. After work on the valves commenced, there was a series of problems associated with the weld filler metal and the rework of the weld. These problems and retests added approximately twelve days to the original schedule.

As can be seen from the attached schedule, if refueling cannot commence as scheduled, there would be a day-for-day delay on the startup of the unit.

Very truly yours,

C.T. Coddington for

H. W. Keiser
Vice President-Nuclear Operations

ctc/1ti202739a:mp

Attachment

cc: M. J. Campagnone - U.S. NRC

L. R. Plisco - U.S. NRC

T. M. Gerusky, Director
Bureau of Radiation Protection
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P.O. Box 2063
Harrisburg, PA 17120

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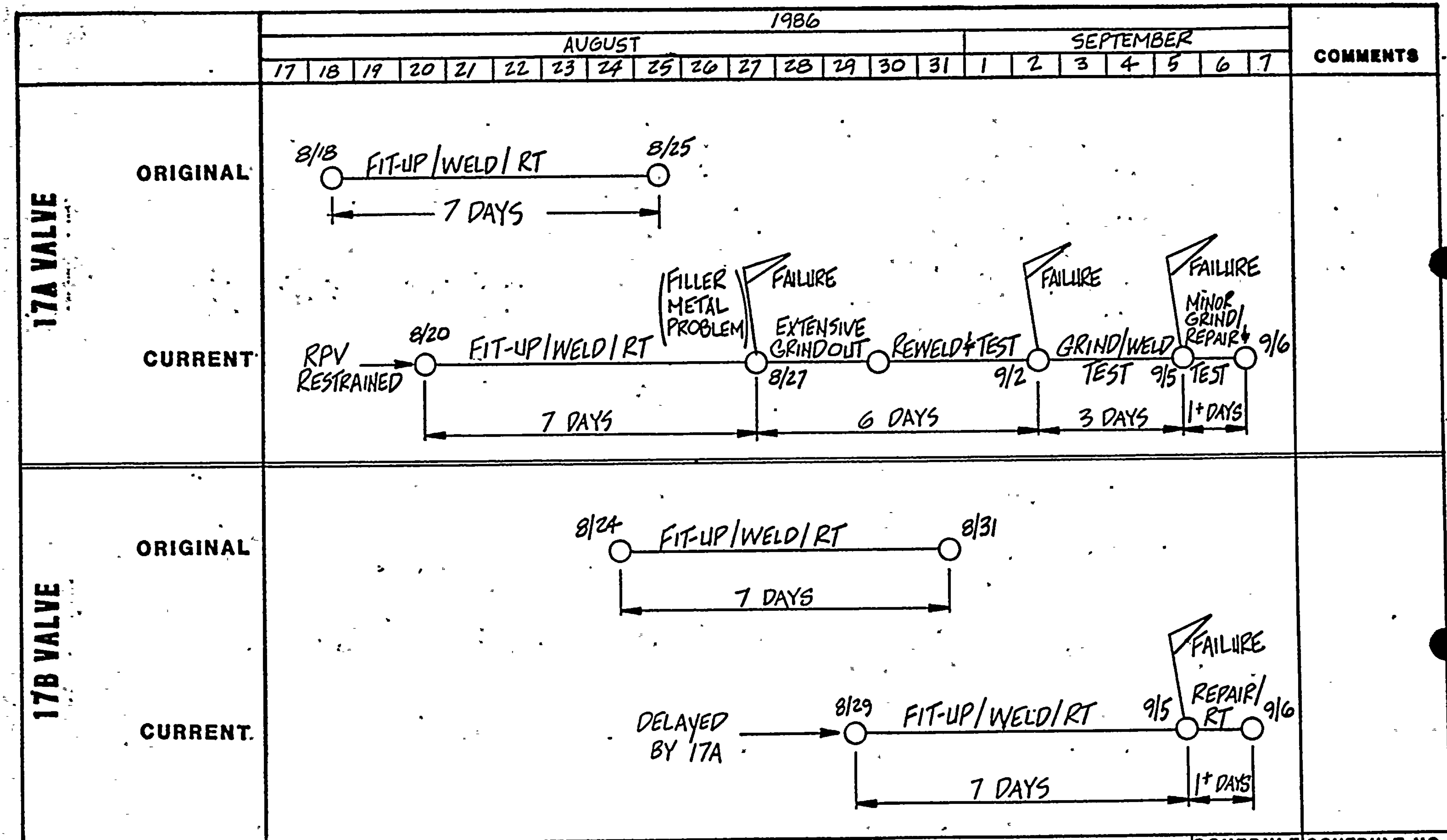
SSES UNIT 2 FIRST REFUEL & INSPECTION OUTAGE

MILESTONE STATUS

#	DESCRIPTION	SCHED DATE	F/C DATE
1	Open Breaker - Commence Outage	08/09/86	08/09/86A
2	Reach Condition 4 - Cold Shutdown	08/10/86	08/09/86A
3	Reach Condition 5 - Refuel	08/11/86	08/12/86A
4	Reactor Cavity Flooded	08/14/86	08/15/86A
5	Commence Core Offload	08/14/86	08/16/86A
6	Core Offload Complete	08/24/86	08/28/86A
7	Complete Division I Work	09/03/86	09/03/86A
8	Declare Division I RHR Operable	09/11/86	09/24/86F
*9	Commence Core Reload	09/14/86	09/24/86F
10	Declare Division II RHR Operable	09/14/86	09/27/86F
11	RWCU/FW Restored to Service	09/19/86	09/19/86F
12	Complete Division II Work	09/21/86	09/27/86F
13	Commence RPV Assembly	09/23/86	10/03/86F
14	Turbine Generator Work Complete	09/24/86	09/24/86F
15	Complete Diesel Generator Testing	09/26/86	09/26/86F
16	Restore to Condition 4	09/29/86	10/09/86F
17	Vessel Leak Test Complete	10/01/86	10/11/86F
18	Turbine Building Restoration Complete	10/02/86	10/12/86F
19	Condition 2 - Commence Startup	10/07/86	10/17/86F
20	Close Breaker - End Outage	10/11/86	10/22/86F



PENNSYLVANIA POWER & LIGHT	REV	DATE	DRWN	PLAN	APPROVAL	TITLE: UNIT 2 RHR EVOLUTION	SCHEDULE STATUS	SCHEDULE NO.
	0	9/5	JTM					SHEET 1 OF 1