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June 27, 1988
4410-88-L-0100/0253P

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sirs:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Safety Evaluation Report for Completion of
Lower Core Support Assembly and Lower Head Defueling - Revision 1

GPU Nuclear letter 4410-88-L-0006 dated June 6, 1988, submitted the Safety Evaluation Report for Completion of Lower Core Support Assembly and Lower Head Defueling for NRC review and approval. Section 5.0, "Radiological Conditions" of that submittal committed to provide an update of the expected occupational exposure to complete Reactor Vessel (RV) defueling and the jobhours and person-rem expended to date for defueling activities. Attached, as Revision 1 to the subject document, is that information. Only the affected pages of the original document are provided (i.e., pages 3, 18 - 27).

GPU Nuclear currently estimates that approximately 1580 person-rem (i.e., an increase of 180 person-rem) will be required for completion of RV defueling. This increase is primarily due to an increase in defueling support activities such as operation and maintenance of the Automated Cutting Equipment System. A revised Table 5-1, updated to May 31, 1988, indicates that 1028 person-rem have been expended for RV defueling activities. A separate activity, entitled "Defueling Support," has been added to this table to more precisely reflect the activities associated with RV defueling.

Sincerely,

F. R. Standerfer
Director, TMI-2

RDW/emf

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Attachment

cc: Senior Resident Inspector, TMI - R. J. Conte
Regional Administrator, Region 1 - W. T. Russell
Director, Plant Directorate IV - J. F. Stolz
Systems Engineer, TMI Site - L. H. Thonus

SAFETY ANALYSIS

SA No. 4710-3221-88-01

Rev. No. 1

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TITLE

SAFETY EVALUATION REPORT FOR

COMPLETION OF LOWER CORE SUPPORT ASSEMBLY AND

LOWER HEAD DEFUELING

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CONCURRENCE

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Design Cognizant Engineer Robert J. Ryan Date 6/22/88 Rad Con ______________ Date 6/20/88

APPROVAL

Mgr Eng. Section Paul J Kozlis Date 6/22/88 Site Ops Director ______________ Date 6-23-88

FORM 4000-ENG-7310.09-1 (12/87)

Title Safety Evaluation REport for Completion of Lower Core Support Assembly and Lower Head Defueling

SUMMARY OF CHANGE

Rev.	SUMMARY OF CHANGE	Approval	Date
0	Initial submittal.		6/88
1	Updated Section 5.0 concerning the expected occupationa exposure to complete Reactor Vessel defueling and the jobhours and person-rem expended to date for defueling activities.		6/88

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accordance with approved procedures for such activities including 4000-PLN-3891.02, "TMI-2 Lifting and Handling Program." Each specific load handling activity is controlled by a Unit Work Instruction or procedure. Load handling activities will be performed by personnel who have been trained and qualified for these activities.

4.14 Reactor Building Basement

The potential for a criticality event in the Reactor Building basement was previously addressed in References 2 and 25.

The controls discussed in Section 4.13 of Reference 2 to ensure subcriticality of potential leakage into the cavity of the RV will continue to be maintained during LCSA/LH defueling. Therefore, criticality is precluded.

5.0 RADIOLOGICAL CONSIDERATIONS

Based on a comparison of activities associated with Reference 1 to those associated with LCSA/LH defueling, it is concluded that the radiological considerations associated with LCSA/LH defueling are bounded by Section 5 of Reference 1. However special precautions will be taken to prevent exposure of operating personnel during transport of radioactive and contaminated pieces of the LCSA from the RV to their storage location within the Reactor Building. Although these pieces of the LCSA will be inspected to ensure there is no visible fuel debris, all pieces are radioactive due to Co-60 activation and surface contamination by soluble fission products.

The sections of the LCSA to be removed under the scope of this SER are less radioactive than the lower grid rib assembly. The measured radiation level of a 5'x5' section of the lower grid rib assembly removed from the LCSA was 80 rem/hr within one (1) foot of the surface. At distance of 30 feet, the radiation level was less than 1 rem/hr following removal. This plate was rigged, moved, and unrigged remotely. Since the sections of the LCSA to be removed from the RV within the scope of this document will represent less of a radiation hazard, the adequacy of the personnel exposure control practices have been demonstrated by the lower grid rib assembly section removal.

An update of the jobhours and person-rem expended to date for all defueling activities is provided in Table 5.1. The overall estimated occupational exposure to complete Reactor Vessel defueling is approximately 1580 person-rem.

TABLE 5.1

JOBHOURS AND PERSON-REM EXPENDED THROUGH MAY 31, 1988

Activity	Jobhours	Person-Rem
Preparation and Installation	5,120	120
Operation	43,534	423
Defueling Support	28,793	440
Maintenance	970	45
Decon and Removal*	0	0
Totals	78,417	1028

* No activity associated with final decontamination and removal of defueling equipment has currently been performed, thus, no jobhours and person-rem are given. Note, decontamination maintenance in the Reactor Building is not considered part of this activity.

6.0 IMPACT ON PLANT ACTIVITIES

The major potential impact of LCSA/LH defueling on plant activities is the effect of fuel movement in Unit 2 on operations in Unit 1. Based on the evaluation provided in Reference 1 and the similarity of the activities considered in Reference 1 to those activities within the scope of this SER, it is concluded that the LCSA/LH defueling operations in Unit 2 will not affect personnel in Unit 1.

7.0 10 CFR 50.59 EVALUATION

10 CFR 50, Paragraph 50.59, permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided the change, test, or experiment is determined not to be an unreviewed safety question and does not involve a modification of the plant technical specifications.

10 CFR 50, Paragraph 50.59, states a proposed change involves an unreviewed safety question if:

a. The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c. The margin of safety, as defined in the basis for any technical specification, is reduced.

Although there are notable differences between the proposed defueling activities for TMI-2 and routine activities described in the FSAR, the consequences of postulated accidents are not different and as demonstrated in Reference 1, are sufficiently similar to be compared. Reference 1 compared two (2) potential events during defueling, a canister drop accident and a Krypton 85 release, with two (2) events described in the FSAR, a fuel handling accident and a waste gas decay tank failure. The comparison demonstrated that, on a worst case basis, the consequences of the FSAR events bound the consequences of any defueling-related event.

A variety of postulated events were analyzed in this SER for LCSA/LH defueling. The analysis of these events provided in Section 4 results in the conclusion that the postulated events are bounded by previous evaluations and/or do not result in an unanalyzed condition.

To determine if LCSA/LH defueling activities involve an unreviewed safety question, the following questions must be evaluated.

Has the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report been increased?

A variety of events were analyzed in Reference 1. It was demonstrated that these events were bounded by comparable events analyzed in the FSAR. It was shown that the potential consequences from these events were substantially less than the potential consequences of comparable events analyzed in the FSAR. Reference 2 evaluates the consequences of potential events during LCSA/LH disassembly and defueling and demonstrates that LCSA/LH defueling can be performed safely.

This SER demonstrates that there is a high probability that the incore nozzles have maintained their original integrity; thus, the potential for a leak due to a load drop is not increased. Additionally, because a RV leak is not likely, the potential for fuel fines from the RV to migrate to the cavity beneath the RV in the Reactor Building basement due to an incore nozzle failure is remote. Further, Reference 2 demonstrates that a basement criticality event external to the vessel due to the presence of this fuel is prevented because of the boron concentration that will be present in the cavity.

By considering postulated events and reviewing various safety mechanisms (i.e., fire protection and decay heat removal), it has been demonstrated that LCSA defueling activities will not adversely effect equipment classified as important to safety (ITS). Consequently, it is concluded that the probability of a malfunction of ITS equipment or the consequences of a malfunction of ITS equipment has not been increased.

Therefore, it is concluded that the proposed activities associated with LCSA defueling do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

Has the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report been created?

The variety of postulated events analyzed in References 1 and 2 considered a spectrum of event types which potentially could occur as a result of the defueling process. A comparison of those events with comparable events in the FSAR demonstrated that the event types postulated for the defueling process are similar and bounded by the FSAR. In addition, no new event type was identified which was different than those previously analyzed in the FSAR or other SERs previously approved by the NRC. Section 4 of this SER evaluates events postulated for LCSA/LH defueling. These type of events have been previously evaluated and, therefore, do not represent a different type of accident or malfunction.

Has the margin of safety, as defined in the basis for any technical specification, been reduced?

Technical Specification safety margins at TMI-2 are concerned with criticality control and prevention of further core damage due to overheating. Technical Specification safety margins will be maintained throughout the LCSA/LH defueling process. Subcriticality is ensured by establishing the RCS boron concentration at greater than 4350 ppm or equivalent and ensuring that this concentration is maintained by monitoring the boron concentration and inventory levels and by isolating potential deboration pathways. Systems will remain in place to add borated cooling water to the core in the event of an unisolable leak from the RV to prevent overheating and potential criticality. Additional borated water has been added to the cavity beneath the RV to bring the boron concentration above 3500 ppm as specified in Reference 2. This action ensures that a criticality event external to the vessel is not credible. The introduction of unborated water from the torch cooling system will not create the potential for a criticality because no more than three (3) gallons of unborated water can be inadvertently drained into the RV (Reference 6).

No Technical Specification changes are required to conduct the activities bounded by this SER.

In conclusion, the LCSA defueling activities do not:

- o Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report, or
- o Create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or
- o Reduce the margin of safety as defined in the basis for any Technical Specification.

Therefore, the LCSA defueling activities do not constitute an unreviewed safety question.

8.0 ENVIRONMENTAL ASSESSMENT

Based on Section 8.0 of Reference 1 and noting the similarities between the activities considered in Reference 1 to those activities within the scope of this SER, it can be concluded that the proposed LCSA/LH defueling activities can be performed with no significant environmental impact.

9.0 CONCLUSIONS

Activities associated with LCSA/LH defueling have been described and evaluated. The evaluations have shown that the radioactivity releases to the environment that will result from the planned activities will not exceed allowable limits. (Reference 1 provides the specific offsite dose analysis.) It has been demonstrated that the consequences of postulated accidents with respect to potential core disturbances will not compromise plant safety. The evaluations have also shown that the tasks and tooling employed follow the continued commitment to maintain radiation exposure levels ALARA. Therefore, it is concluded that LCSA/LH defueling activities can be performed without presenting undue risk to the health and safety of the public.

10.0 REFERENCES

1. Safety Evaluation Report for Defueling the TMI-2 Reactor Vessel, Revision 10, 15737-G07-108, May 1986.

2. Safety Evaluation Report for Lower Core Support Assembly Defueling, Revision 2, 4710-3221-86-011, January 1988.

3. NRC Letter dated April 1, 1988, Lower Core Support Assembly Defueling.

4. Criticality Report for the Reactor Coolant System, Revision 0, 15737-2-N09-001, October 1984.

5. Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities, Revision 1, 15737-2-N09-002, September 1985.

6. Criticality Safety Assessment for Using the Plasma Arc Torch to Cut the LCSA, 15737-2-N09-004, November 1987.

7. GPU Nuclear letter 4410-88-L-0067 dated April 29, 1988, "Plasma Arc Torch Coolant System."

8. GPU Nuclear letter 4410-88-L-0026, dated February 26, 1988, "Response to NRC Comments on the Criticality Safety Assessment for Using the Plasma Arc Torch to Cut the LCSA."

9. Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System, Revision 2.

10. Safety Evaluation Report for Core Stratification Sample Acquisition, Revision 4, 15737-2-G07-109, July 3, 1986.

11. GPU TPO/TMI-127, Revision 0, "Technical Plan for Pyrophoricity," December 1984.

12. EG&G Plasma Arc Test Report, LCSA-4, April 30, 1986.

13. Evaluation of the Structural Integrity of the TMI-2 Reactor Vessel Lower Head - Final Report, June 1985, B&W 77-1154826-00.

14. EG&G-TMI-7784, August 1987, "TMI-2 Reactor Vessel Lower Head Heatup Calculations."

15. GPU TPO/TMI-175, Revision 6, "Analysis of Gamma Scanning of Incore Detector No. L-11 in Lower Reactor Vessel Head," June 1985.

16. EG&G-TMI-7811, September 1987, "Thermal Interaction of Core Melt Debris With The TMI-2 Baffle, Core Former and Lower Head Structures."

17. GPU Nuclear letter 4410-87-L-0189, dated December 28, 1987, "Use of Core Bore Machine for Dismantling the Lower Core Support Assembly," to Document Control Desk from F. R. Standerfer.

18. NRC Letter NRC/TMI-88-003, dated January 8, 1988, to F. R. Standerfer from W. D. Travers.

19. GPU Nuclear letter 4410-86-L-0162, dated September 19, 1986, "Core Bore Operations," to W. D. Travers from F. R. Standerfer.

20. NRC Letter NRC/TMI-86-101, dated October 16, 1986, "Core Bore Operations," to F. R. Standerfer from W. D. Travers.

21. GPU Nuclear Technical Bulletin 88-02, Revision 0, dated April , 1988

22. GEND-INF-031 Volume II, April 1984, "TMI-2 Incore Instrument Damage - An Update."

23. GPU Nuclear Technical Bulletin 88-05, Revision 0, dated April , 1988.

24. Safety Evaluation Report for Core Support Assembly and Lower Head Defueling, Revision 0, 4710-3221-86-011, February 1987.

25. Safety Evaluation Report for Reactor Building Sump Criticality, Revision 2, 4550-3254-85-02, January 1986.

TABLE 1

THERMOCOUPLE LENGTHS

Assembly Number	Grid Location	Original Length in Reactor (ft)	Calculated Reduction in Length (ft)	Length From Reactor Base (ft) *
1	H8	21.00	17.04	3.96
2	H9	20.97	6.91	14.06
3	G9	20.93	18.93	1.99
4	F8	20.86	19.70	1.16
5	E9	20.64	17.31	3.33
6	F7	20.82	19.58	1.24
7	E7	20.64	19.75	0.88
8	G6	20.82	20.04	0.78
9	G5	20.64	20.27	0.37
10	H5	20.68	16.65	4.02
11	K5	20.64	6.93	13.71
12	L6	20.71	8.95	11.76
13	M7	20.64	10.77	9.87
14	N8	20.41	9.84	10.57
15	N9	20.37	16.93	3.44
16	M9	20.64	19.97	0.66
17	M10	20.53	18.13	2.40
18	L11	20.53	7.96	12.56
19	K11	20.64	19.92	0.72
20	K12	20.37	--	--
21	H13	20.06	--	--
22	G13	20.02	13.49	6.53
23	F13	19.89	--	--
24	F12	20.26	6.63	13.63
25	G11	20.64	--	--
26	E11	20.33	20.33	0.00
27	D10	20.26	19.20	1.06
28	C10	19.89	10.77	9.12
29	C9	20.02	9.46	10.56
30	B8	19.59	9.52	10.08
31	B7	19.55	8.07	11.48
32	C6	19.89	8.42	11.48
33	D5	20.06	8.44	11.62
34	E4	20.06	--	--
35	F3	19.89	--	--
36	G2	19.55	8.92	10.63
37	H1	19.00	7.30	11.70
38	L2	19.42	10.41	9.01
39	L3	19.89	10.48	9.42
40	M3	19.68	--	--
41	N4	19.77	7.81	11.96
42	O5	19.68	10.15	9.53
43	O6	19.89	--	--
44	P6	19.42	--	--

*These measurements have an uncertainty of ± 1.25 feet.
--Indicates open circuits.

TABLE 1

THERMOCOUPLE LENGTHS

Assembly Number	Grid Location	Original Length in Reactor (ft)	Calculated Reduction in Length (ft)	Length From Reactor Base (ft) *
45	R7	18.95	16.93	2.01
46	R10	18.80	6.58	12.22
47	O10	19.89	--	--
48	O12	19.37	8.41	10.96
49	M14	19.19	13.72	5.47
50	L13	19.89	--	--
51	O14	18.85	4.49	14.35
52	C13	18.95	8.37	10.58

*These measurements have an uncertainty of ± 1.25 feet.
--Indicates open circuits.

TABLE 2

CATEGORIZATION OF INCORE DETECTOR OBSERVATIONS

CATEGORY	NUMBER OF INCORES	CALCULATED LOCATION OF THERMOCOUPLE JUNCTION RELATIVE TO LOWER GRID RIB SECTION (a)	NUMBER & LOCATION OF DETECTOR SEPARATION BASED ON VIDEO DATA (ABOVE/BELOW LOWER GRID RIB SECTION)	AGREEMENT BETWEEN THERMOCOUPLE REDUCTION DATA AND VIDEO DATA %
A	23	ABOVE LOWER GRID	20-ABOVE LOWER GRID 2-BELOW LOWER GRID 1-UNKNOWN	87%
B	16	BELOW LOWER GRID	11-BELOW LOWER GRID 5-ABOVE LOWER GRID	69%
C	2	AT LOWER GRID(b)	2-ABOVE LOWER GRID	100%
-	*41*	*SUBTOTAL*	*33 OUT OF 41 AGREE*	80%
D	11	OPEN JUNCTION	11-ABOVE LOWER GRID	N/A
ALL	52	OVERALL	38-ABOVE LOWER GRID 13-BELOW LOWER GRID 1-UNKNOWN	80% (33 out of 41)

(a) BASED ON GEND-INF-031 Vol.II, April-84.
(b) MEASUREMENTS INCORPORATE AN UNCERTAINTY OF +/- 1.25 ft.