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April 26, 1985

TMI Program Office
Attn: Dr. B. J. nyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. nyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Technical Evaluation Report for the Defueling Water Cleanup System

Attached for your review and approval is a revision to the Technical Evaluation Report for the Defueling Water Cleanup System (DWCS). This revision reflects the addition of a relief valve at the outlets of the defueling filter canisters, deletion of the Fuel Transfer Canal/Spent Fuel Pool Cleanup System boronometer, and correction of minor typographical errors. Revisions to the System Descriptions for the Defueling Water Cleanup (DWC) Reactor Vessel Cleanup System and the DWC Fuel Transfer Canal/ pent Fuel Pool Cleanup System as well as applicable drawings which reflect the above noted changes also are attached.

GPU Nuclear letter 4410-85-L-0005, dated January 14, 1985, previously submitted the application fee for the DWCS in accordance with 10 CFR part 170; therefore, no fee is included with this request.

Sincerely,

F. R. Standerfer
Vice President/Director, TMI-2

FRS/RDW/eml

Attachments

cc: Deputy Program Director - TMI Program Office, Dr. W. D. Travers

D009
1/1

Attachments

1. Division II System Design Description of the Defueling Water Cleanup Reactor Vessel Cleanup System, Doc. No. 15737-2-M72-DWC01, Rev. 3.

2. Division II System Design Description of the Defueling Water Cleanup Fuel Transfer Canal/Spent Fuel Pool Cleanup System, Doc. No. 15737-2-M72-DWC02, Rev. 3.

3. Reactor Vessel Cleanup System, Piping and Instrument Diagram, Bechtel Drawing 15737-2-M74-DWC01, Rev. 5.

4. Fuel Transfer Canal/Spent Fuel Pool Cleanup System, Piping and Instrument Diagram, Bechtel Drawing 15737-2-M74-DWC02, Rev. 5.

5. Auxiliary Systems, Piping and Instrument Diagram, Bechtel Drawing 15737-2-M74-DWC03, Rev. 5.

6. Fuel Handling Building, General Arrangement, Bechtel Drawing 15737-2-POA-6401, Rev. 2.

7. Reactor Building, General Arrangement, Bechtel Drawing 15737-2-POA-1303, Rev. 3.

TRANS. NO. DEDT- 1500

TRANSMITTAL

FROM: R. L. Rider
Manager, Design Engineering
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GPU Nuclear

TO: GPU NUCLEAR TMI-2 DIVISION
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MIDDLETOWN, PA 17057

ATTN: Supervisor, Records Services - A. Stowe

DATE: April 18, 1985

BECHTEL JOB 15737
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1	Other	Encl.	☒	Under Sep. Cover	☐

ACTION NUMBER

1. Approved. Manufacturing may proceed.
2. Approved. Submit final dwg. Mfg. may proceed.
3. Approved except as noted. Make changes and submit final dwg. Mfg. may proceed as approved.
4. Not Approved. Correct and resubmit.
5. Review not required. Mfg. may proceed.
6. Comment/Approval Due Date ______
7. Information Only
8. Other Issued For Use

DESCRIPTION

Action No.	Bechtel Number	Manufacturer's Number	Rev.	Title
8	15737-2-G03-106	-	4	Technical Evaluation Report for Defueling Water Cleanup System

SNL/tlj

(See Attached Sheet for Distribution List)

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Very truly yours,

for R. L. Rider
Manager, Design Engineering

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TER 15737-2-G03-106 REV. 4

ISSUE DATE April 18, 1985

☒ ITS
☐ NSR
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TMI-2
DIVISION
TECHNICAL EVALUATION REPORT
FOR

Defueling Water

Cleanup System

COG ENG Gerald K Boldt DATE 11/1/84

RTR Edward T. Smith DATE 11/1/84

COG ENG MGR. C.L. R.I. for RLR DATE 11/1/84

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
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1	12/7/84	Issued For Use	[illegible]	[illegible]	ETS	CLR
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Rev.	SUMMARY OF CHANGE
0	Initial issue November 1, 1984
1	Revised to incorporate system design changes and comments on Revision 0
2	Revised to incorporate comments on Revision 1
3	Revised to incorporate comments on Revision 2
4	Revised to reflect addition of relief valves at the outlets of the defueling filter canisters, deletion of fuel pool cleanup system boronometer and correction of minor typographical errors

SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.
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BECHTEL

REVISION STATUS SHEET

DOCUMENT TITLE: Technical Evaluation Report for Defueling Water Cleanup System

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1. Division II System Design Description of the Defueling Water Cleanup Reactor Vessel Cleanup System, Doc. No. 15737-2-M72-DWC01, Rev. 3. 4

2. Division II System Design Description of the Defueling Water Cleanup Fuel Transfer Canal/Spent Fuel Pool Cleanup System, Doc. No. 15737-2-M72-DWC02, Rev. 3. 4

3. Reactor Vessel Cleanup System, Piping and Instrument Diagram, Bechtel Drawing 15737-2-M74-DWC01, Rev. 5. 4

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5. Auxiliary Systems, Piping and Instrument Diagram, Bechtel Drawing 15737-2-M74-DWC03, Rev. 5. 4

6. Fuel Handling Building, General Arrangement, Bechtel Drawing 15737-2-POA-6401, Rev. 2. 4

7. Reactor Building, General Arrangement, Bechtel Drawing 15737-2-POA-1303, Rev. 3. 4

1.0 Introduction

1.1 General

The defueling water cleanup (DWC) system is designed to remove radioactive ions and particulate matter from the fuel transfer canal, spent fuel pool "A" and the reactor vessel. The majority of the particulate matter is removed by processing the water through nominal 0.5 micron rated sintered metal filters. The low micron rating of the filters will assure very low turbidity as well as reducing the particulate activity in the water.

Removal of the radioactive ions (i.e., soluble fission products) will be performed by processing a portion of the filter output through 4 x 4 liners (similar to those in use for EPICOR II) containing Zeolite, or the submerged demineralizer system (SDS).

1.2 Scope

The scope of this document includes the operation of the DWC system, the components of the DWC system and its interfaces to existing systems and components. This technical evaluation report (TER) is applicable only during the recovery mode as the DWC system is a temporary system required to support recovery operations and will be removed or reevaluated prior to plant restart. Evaluation of safety concerns related to the filter canisters is not within the scope of this TER and will be addressed in Reference 8. Licensing of the ion exchangers for offsite shipments is outside the scope of this TER. |3

2.0 System Description

2.1 General

The DWC system is designed to process water from the reactor vessel, spent fuel pool, and fuel transfer canal. The system's major functions are given below.

a) The DWC system filters the water contained in the reactor vessel, the spent fuel pool, and the fuel transfer canal to remove suspended solids above a nominal 0.5 micron rating. This is done to maintain the clarity of the water to a 1 NTU (nephelometric turbidity unit) rating.

b) The DWC system removes soluble fission products from the reactor vessel, the spent fuel pool, and the fuel transfer canal (FTC) by demineralization of the water. This is done to reduce the dose contribution from the water.

The DWC system is composed of two major subsystems which allow greater processing flexibility during post plenum removal operations. These two subsystems are, the reactor vessel cleanup system and the fuel transfer canal/spent fuel pool cleanup system. Online sampling of both subsystems for pH is provided by the system design. Online sampling for boron concentration and turbidity is provided for the reactor vessel cleanup system. Boron sampling for the fuel transfer canal/spent fuel pool cleanup system (FTC/SFP) will be done according to NRC approved procedures. The detailed system description for the DWC reactor vessel cleanup system is provided in Attachment 1. Attachment 2 provides the detailed system description for the DWC fuel transfer canal/spent fuel pool cleanup system. Also included as Attachments 3 through 7 are the following figures: 4

Attachment 3 Reactor Vessel Cleanup System, Piping and Instrument Diagram

Attachment 4 Fuel Transfer Canal/Spent Fuel Pool Cleanup System, Piping and Instrument Diagram

Attachment 5 Auxiliary Systems, Piping and Instrument Diagram

Attachment 6 Fuel Handling Building, General Arrangement

Attachment 7 Reactor Building, General Arrangement

2.2 Quality Classification

The quality classification of the DWC system with exception of the filter/canister units which are not within the scope of this TER is Important to Safety. Important to Safety as used here is defined in the TMI-2 Recovery Quality Classification List.

3.0 Technical Evaluations

3.1 General

The DWC system is totally contained within areas that have controlled ventilation and area isolation capability. This limits the environmental impact of the system during normal system operations, 4 shutdown or postulated accident conditions. The impact of postulated DWC system failures is provided below on a case-by-case basis.

The system failures evaluated are loss of power, loss of instrumentation/instrument air, filter media rupture, and line breaks. The design of the system is such that none of the events results in unacceptable consequences. Other safety concerns evaluated with respect to operation of the DWC system were decay heat removal, criticality, boron concentration control, heavy load drops, and radioactive releases. No unacceptable consequences were found to result from operation of the DWC system provided that proper administrative control is maintained.

3.2 Postulated System Failures

3.2.1 Reactor Vessel Cleanup System

3.2.1.1 Loss of Power

A loss of power to the entire system would simply shut the system down. A loss of power to the well pumps with an additional failure which results in simultaneous loss of level control in the ion exchangers would result in a flow mismatch. In this case, the system would be automatically shut down until power is restored. Loss of power to individual components would place that component in its safe mode for an air operated valve; for example, it would fail to a position that ensures no damage to other components. 1

Loss of power to the control panel would cause the loss of all information and fail all control and solenoid operated valves. The system would be shutdown until power is restored.

3.2.1.2 Loss of Instrumentation/Instrument Air

Loss of a single instrument channel will result in the loss of indication for that channel and, for those channels that have control features a flow mismatch. This flow mismatch will result in an automatic shutdown of the affected portion of the system.

Loss of the internals indexing fixture (IIF) level indication system (bubbler) will result in an erroneous level indication which will be noted when compared with a redundant level indication system. Since this system has no control features, no adverse system conditions will result.

Loss of instrument air will take the individual components to their fail safe position. Flow mismatches induced by loss of air will result in automatic trips. Loss of air to the IIF level monitoring system will initiate a low air supply pressure alarm.

3.2.1.3 Filter Media Rupture

A failure of the filter media in the canister could potentially release fuel fines to the ion exchange portion of the system. A post filter is located downstream of both filter trains in the line to the ion exchangers. This post-filter will trap any fuel fines which would be transported past the filter canisters in the event of filter failure. The post filter is sized to be critically safe and so that a gross rupture in a filter canister will increase the differential pressure to the alarm setpoint. Turbidity meters will aid in the detection of gross filter media rupture by detecting changes in water clarity.

Upon detection of a filter media rupture the filter trains will be isolated and the ruptured filter will be identified by observing the differential pressure
versus flow for each individual canister with flow 1
being recirculated to the reactor vessel. A lower differential pressure for a given flow will indicate
which filter is ruptured. The ruptured canister or 1
canisters and the post-filter cartridge would then be replaced as required and the system restarted.

3.2.1.4 Line Break

The principal consequence of any line, or hose break in the reactor vessel cleanup system is a loss of reactor vessel inventory. The system is designed to mitigate the consequences of such an incident to the extent possible.

To help prevent a hose rupture, all process water hoses are armoured. In case of a hose rupture or line rupture, downstream of the reactor vessel pumps, the system will trip these pumps on IIF low level and alarm at control panels in the control room and fuel handling building. This could deliver approximately 500 to 1000 gallons of reactor vessel water to the area of the break. The potential areas affected would be the Reactor Building and the Fuel Handling Building, each of which has sumps to contain the spill.

Siphoning of reactor vessel water could take place if
any of the lines connected to the well pump suction or 1
return hoses, or if the hoses themselves, are damaged or rupture. The two, 4 inch suction connections provided in the Westinghouse work platform will be provided with two, 3/4 inch holes drilled 18 inches below the water level which will act as a siphon breaker. The three 2 inch return lines will be equipped with spargers, which are holes drilled into the pipes. The first holes are drilled 18 inches below the water level which will act as a siphon breaker. The sample return line will terminate 18

inches below the water level. Also, isolation valves will be provided in the Westinghouse supplied piping which could be used to manually terminate the siphoning. Therefore, a maximum of approximately 3000 gallons of reactor vessel water would spill into the fuel transfer canal following a hose rupture. Approximately half of this water would be contained in the New Fuel Pit.

The recovery from these events would be accomplished by isolating the ruptured section and replacing the ruptured hose/pipe.

3.2.2 Fuel Transfer Canal/Spent Fuel Pool Cleanup System

3.2.2.1 Loss of Power

A loss of power to any portion of the system would shut that portion of the system down. Loss of power to individual components would place that component in its safe mode for an air operated valve, for example it would fail to a position that ensures no damage to other components.

3.2.2.2 Loss of Instrumentation/Instrument Air

Loss of a single instrument channel will result in the loss of indication for that channel and, for those channels that have control features a flow mismatch. This flow mismatch will result in an automatic shutdown of the affected portion of the system.

Loss of either the spent fuel pool or FTC level monitoring system will be noted when compared with the other. The readings should normally be the same since both water bodies are in communication via the fuel transfer tubes. Neither system has control features.

Loss of instrument air will take the individual components to their fail safe position. Flow mismatches induced by loss of air will result in automatic trips. Loss of air to the IIF level monitoring system (bubbler) will initiate a low air supply pressure alarm.

3.2.2.3 Filter Media Rupture

A failure of the filter media in the canister could potentially release fuel fines to the ion exchange portion of the system. Flow may be routed to DWC ion exchanger K-2 or to the SDS both of which have filters upstream to trap migrating fuel fines. Ion exchanger K-2 has a cartridge type filter in a critically safe 2 canister and SDS is equipped with two filters in series, both of which have borosilicate glass to control reactivity (see Ref 2). Differential pressure is measured across the filters to indicate ruptured filter media. The SDS filter bypass is administratively controlled to prevent inadvertent operation. 2

Upon detection of a filter media rupture the filter trains will be isolated and the ruptured filter will be identified by observing the differential pressure versus flow for each individual canister with flow being recirculated to the fuel pool. A lower differential pressure for a given flow will indicate which filter is ruptured. The affected canister or canisters and the SDS pre-filter vessel or filter canister post filter cartridge would then be replaced as required and the system restarted. 2

3.2.2.4 Line Break

If a rupture occurred in the FTC/spent fuel pool cleanup system, the DWC system spent fuel pool pumps could deliver fuel transfer canal and/or spent fuel pool water to the Fuel Handling Building or the Reactor Building. This action would lower the level in the canal and the pool. A drop of one inch in canal/pool level is approximately equivalent to 1250 gal. A level loss would be detected and alarmed (low level alarm 3" below normal liquid level) by at least one of the two redundant level indicating systems provided for the canal/pool. The operator would then shut the system down.

Process water hoses are employed in three services in this system; filter canister inlet/outlet, skimmers to well pumps, and downstream of penetration R-539.

If a filter canister inlet/outlet hose ruptures, that canister will be isolated and the hose replaced. Since these hoses are submerged in the SFP, this results in no net water loss.

If a hose connecting the skimmer to the well pumps breaks, then the ability to surface skim will be hampered or lost, but pump capacity will not be diminished as the hose is routed underwater to the pumps and a pump suction supply will continue to be available.

3.5 Boron Dilution

The only credible means of attaining criticality of the fuel contained in the vessel is through deboration of the RCS water. The approach described in References 6 and 7 for prevention of deboration will be followed for operation of the DWC system. Specific system evaluations with respect to deboration control will be performed prior to DWC system operation. Boron dilution during defueling will be addressed in a revision of the "Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System".

3.6 Heavy Load Drops

In-containment load handling will consist of the transfer of the DWC filter canisters from the deep end of the FTC to the Fuel Handling Building via the fuel transfer system. The handling of these canisters will be in accordance with procedures, or unit work instructions (UWI's) which will define load paths. These load paths will be administratively controlled to ensure that a postulated drop of a canister would not compromise plant safety or the integrity of the FTC floor.

Load handling within the Fuel Handling Building will consist of the movement of SDS ion exchange liners, the reactor vessel cleanup system liners, the DWC filter canisters and transfer casks. The heavy load drop analysis for the SDS casks is given in reference 3. The reactor vessel cleanup system liners will be moved using the existing casks for the EPICOR II system. The total heavy load of the cask and liner is less than that of the shield plugs removed from spent fuel pool "A". The load path for movement of the liners,
although not over the spent fuel pool, is part of the same load path
used for the spent fuel pool "A" shield plug removal. The load path 2
and heavy load drop analysis provided in the spent fuel pool "A" refurbishment SER (Ref. 5), therefore, bounds the movement of the liners and casks. The radiological concerns associated with a load drop are bound by the analysis in Reference 4 which concludes the health and safety of public is not endangered as a result of this hypothetical accident. The handling of heavy loads over the spent
fuel pool is not within the scope of this document and will be 2
addressed in the Early Defueling SER.

3.7 Radioactive Releases

The operation and design of the DWC system was reviewed with respect to radioactive releases. No direct radioactive release paths to the environment exists for the system. Local spillage of contaminated water from the DWC system will result in a local contamination problem. Since the specific activity of the water is essentially that of the fuel transfer canal and spent fuel pool, no significant radioactive releases above those from the open pools can occur when processing pool water. Defueling activities have the potential of significantly increasing the specific activity of the reactor vessel water. To preclude any significant releases during these periods the operating procedures associated with processing reactor vessel water shall include requirements to ensure isolation of the system should a line break or massive system leakage occur.

During shutdown of the DWC system filter trains, radiolytic decomposition of the water in the filter canisters will cause the production of hydrogen and oxygen. In order to prevent the overpressurization of the filter canisters, an ASME Section VIII pressure relief valve is installed in the outlet pipe from each of the four DWCS filter canisters. The valves provide pressure relief in the event that the isolation valves in the DWCS filter inlet and outlet pipes are closed and pressure builds up within the filter canister as a result of radiolytic decomposition.

The filter canisters should not normally be isolated for extended periods; however, if they were, the maximum rate of hydrogen and oxygen generation within the canister based on conservative assumptions is estimated to be 0.029 acf/day. At this rate of gas generation, the pressure inside the canister would not reach the canister design pressure (150 psig) for at least 90 days. The relief 4
valve will release the pressure buildup before this pressure is exceeded with approximately 0.3 scf of hydrogen and oxygen released from each canister. The relief valves will continue to relieve pressure at about 15 day intervals, releasing a maximum of about 0.3 scf hydrogen and oxygen per canister per relief. The relief valves discharge to the open volume of the containment above the fuel transfer canal or to the operating level of the fuel handling building. Since both of these areas are continuously or regularly vented and since the maximum volume of hydrogen released is small, a buildup of hydrogen to a combustible concentration is not credible. Any particulate releases during the operation of the relief valves would be bounded by the line breaks discussed in sections 3.2.1.4 and 3.2.2.4.

4.0 Radiological and Environmental Assessment

4.1 Off-Site Dose Assessment

Operation of the DWC system could reduce the off-site doses which would result if the system were not available. Without operation of the DWC system specific activity of the water in the pools would slowly increase. This could lead to an increase in the local airborne concentration available for release via the plant ventilation system. However, operation of the DWC system will maintain the reactor and fuel pool water at very low specific activity, thereby minimizing this as a potential release source. Since the source available for release from the SDS greatly exceeds that available from the DWC system, the off-site dose analysis provided in the SDS TER (Ref. 4) bounds those of the DWC system.

4.2 On-Site Dose Assessment

4.2.1 Reactor Vessel Cleanup System

The potential exists that defueling may significantly increase the specific activity in the reactor vessel water. This could possibly occur during defueling through disturbance of the core debris. Material greater than nominal 0.5 microns would be captured in underwater filter canisters. The soluble fission products, particularly cesium-137, would be removed by processing through the associated ion exchange media. The filter canisters are located underwater at a depth greater than four feet in the reactor building and therefore do not represent a radiological problem. The water to be processed is piped through a reactor building penetration to the ion exchange media at 20 to 60 gpm (max. 30 gpm/train) depending on the specific activity of the reactor vessel water. These process lines and the liners for the ion exchange media represent potential radiological hazards.

To assess the radiological hazards, the dose rates from DWCS piping and components during operation were evaluated. Sources in the water were assumed to be fuel particles and dissolved radioactive materials. The design basis concentrations of these sources are 1 ppm suspended solids and a concentration of soluble materials equivalent in dose rate to 0.02 μCi/ml of cesium-137. During operation at the design basis concentrations, the dose rate from a long 3" diameter unshielded hose is 0.2 milirem/hour at a distance of 2 feet.

During defueling operations both the solubles and suspended solids concentrations in the water may increase. To assess increases in dose rates during upset water conditions, a combination of a 20 curie cesium-137 spike and an instantaneous release of approximately 35 lb of suspendabl fine debris to the reactor vessel volume is postulated. A long 3" diameter hose carrying water at the resulting concentrations would result in a dose rate of 9 millirem/hc 2 feet from the hose. Process lines which are downstream of the filters do not contain the suspended solids concentrations postulated for the upset water conditions. A 3" diameter hose downstream of the filters would produce a dose rate of 2 millirem/hour at a distance of 2 feet, due to the soluble radioactive materials remaining in the water.

Shielding of lines upstream of the filters may be used to reduce dose rates in areas of personnel occupancy.

Dose rates from solubles are based on the specific activity of cesium-137. Other isotopes which may contribute significantly to gamma dose rates are cesium-134 and antimony-125. The cesium-134 concentration is normally an order of magnitude less than that of cesium-137. Antimony-125 is not removed by the DWCS ion exchangers with a reliable decontamination factor. However, the dose rate for antimony-125 is less than that of cesium-137 for a given concentration. If antimony-125 in the DWCS becomes a significant dose contributor to workers, the reactor coolant may be processed through the EPICOR II system in a batch processing mode. Batch processing will be used because chemical adjustment of the coolant is required. EPICOR II will remove the antimony-125 with a satisfactory decontamination factor.

Three zeolite ion exchangers are needed to handle the flow from DWC system. Two are needed for the reactor vessel cleanup system to provide a 60 gpm flowrate through the ion exchangers. One is used for FTC/spent fuel pool cleanup. SDS is also to be used for FTC/spent fuel pool cleanup.

The shielding requirements for these liners will be based on a homogenized 500 Ci source in a 4 x 4 liner, similar in construction to those used for EPICOR II. Since changeout of liners will be based on radiation level, and since the 500 Ci loading is conservatively high (actual loading should be approximately 100 Ci, see Section 4.3), the calculated shielding requirement is considered acceptable.

The contact dose rate on the side of the liner for a homogenized 500 Ci source is approximately 185 R/hr. The liners will be shielded to limit the shield contact dose rate at the side and on top of the liner to a maximum of 5 millirem/hr. The concrete floor will reduce the dose rates on lower elevations to less than 5 millirem/hr.

Both dose rates represent an upper bound, and as indicated, the dose rates would not pose any undue operational constraints if actually attained.

If hoses or piping in the DWC system break, water will be released in the Reactor Building or the FHB. This water may contain suspended fuel particles and dissolved radioactive materials. The specific activity of the DWC system water will be maintained low enough that personnel access to the spill area will not be precluded. After the removal of the spilled water, the area may require decontamination to reduce loose surface contamination to acceptable levels. Thus there are no safety concerns associated with the breakage of DWC system hoses or pipes.

4.2.2 Fuel Transfer Canal/Spent Fuel Pool Cleanup System

The fuel transfer canal/spent fuel pool cleanup system processes water through the DWC ion exchanger K-2 or SDS. The water in the pools will be maintained by this system at .01 to .02 μCi/ml of cesium-137. This is significantly lower concentrations than water processed by SDS. The analysis provided in the SDS TER Reference 4 therefore bounds the doses possible from this system.

4.3 Occupational Exposures

Operation of the DWC system will reduce the occupational exposure during defueling operations by maintaining low specific activities in the fuel transfer canal, spent fuel pool and reactor vessel. The DWC system is designed to maintain the maximum Cesium-137 concentration in the water to between .01 and .02 μCi/ml. This will result in a contribution to general area dose rates of 10 to 20 millirem/hr from the water.

It is estimated that approximately 42, 4x4 liners each loaded with 52 curies of Cesium-137 will be required for the reactor vessel cleanup system. The occupational dose to workers during each change-out is estimated to be less 0.1 man-rem. Therefore the total accumulated dose for change out of the estimated 42, 4x4 liners is 4.2 man-rem.

The following table provides an estimate of the man-hours and man-rem associated with the installation, operation, maintenance and removal of the in-containment and fuel handling building portions of the DWCS. These estimates are based upon current man-hour projections.

IN-CONTAINMENT

Activity	Man-Hours	Dose Rate (mR/hr)	Man-Rem
Installation	505	60	30.3
Operation	40	60	2.4
Maintenance	85	60	5.1
Removal	250	60	15.0

FUEL HANDLING BUILDING

Activity	Man-Hours	Dose Rate (mR/hr)	Man-Rem
Installation	34,400	0.3	10.3
Operation	26,280	0.3	7.9
Maintenance	8,600	0.3	2.6
Removal	17,200	0.3	5.2

The total man rem attributable to the operation and maintenance of the DWC system, as a whole, is expected to be between 65 and 125 man-rem. This estimate is based upon a total of 80 man-rem from above increased by 20% for Health Physics coverage and allowing $\pm$ 30% due to uncertainties.

5.0 Safety Evaluation

5.1 Technical Specifications/Recovery Operations Plan

No additional Technical Specifications/Recovery Operations Plan changes, beyond those required for head removal, are required to install and operate the DWC system.

5.2 Safety Questions (10CFR50.59)

10CFR50, 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.

A proposed change involves an unreviewed safety question if:

4

a) The possibility of occurrence or the consequences 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.

The DWC system does not increase the probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in a safety analysis report. The system failures evaluated are presented in section 3.2 of this report. No failures of the DWC system were found which would increase the probability of occurence or the consequenses of an accident or malfunction of equipment important to safety. In addition, operation of the DWC system will be performed under strict administrative procedural control to further ensure safe operation. The procedures used for operation of the DWC system will be reviewed and approved prior to use in accordance with Technical Specification 6.8.1.

The possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis report is not created by the existence of the DWC system. The DWC system is essentially a liquid radwaste system utilized to maintain clarity and low specific activity in the reactor vessel, fuel transfer canal, and spent fuel pool water. As such, the possibility of an accident or malfunction is of the same type as previously evaluated for other liquid radwaste systems.

Operation of the DWC system does not result in a reduction in the margin of safety as defined in the bases for the Technical Specifications. Liquid effluents will not be released to the environment directly from DWC system operations. The effluents from operation of the DWC will be returned to the sources in order to maintain proper water levels. Any gaseous effluents resulting from DWC system operations will traverse existing gaseous effluent flow paths. The gaseous effluents will be less than those generated during processing of the water from the reactor building basement by SDS. The results of the radioactive release analysis presented in the SDS Technical Evaluation Report therefore bound the releases from the DWC. Since no change in the maximum permissible concentrations or the instrument configuration or setpoints specified in Appendix B of the Technical Specifications was required for SDS operation, and since the DWC system operation is bounded by the SDS operation, no changes are required for DWC system operation.

Based on the above, the installation and operation of the DWC system does not present an unreviewed safety question as defined in 10 CFR 50.59.

6.0 References

1. Recovery Program System Description, Auxiliary Building Emergency Liquid Clean-up System (EPICOR II), GPUNC Letter 4410-84-L-0023, Feb. 24, 1984.

2. Technical Evaluation Report (TER) for the Submerged Demineralizer System, GPUNC Letter 4410-84-L-109 dated July 19, 1984.

3. Letter from G. K. Hovey, GPU, to B. J. Snyder, NRC, dated September 30, 1981, "Control of Heavy Loads". GPU letter No. LL2-81-0227. 4

4. Same as Reference 2.

5. Safety Evaluation Report (SER) for the Refurbishment of Fuel Pool "A", Revision 1, June 1983, GPUNC Letter 4410-83-L-0156, July 29. 1983.

6. SER for Removal of the TMI-2 Reactor Vessel Head, Revision 5, February 1984.

7. SER for the Operation of the IIF Processing System, Revision 1, May 1984.

8. TER for Defueling Canisters.