THIRD REPORT OF TECHNICAL ASSISTANCE AND ADVISORY GROUP (TAAG)

AUGUST 31, 1982

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THIRD TAAG REPORT

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THIRD TAAG REPORT

Introduction

This report is the third one prepared by the Technical Assistance and Advisory Group (TAAG). The first report covered the period of time from December 22, 1981 to March 1, 1982. The second report covered the TAAG effort from March 1, 1982 to May 15, 1982. Both of the prior reports were transmitted to TAAG, U.S. Department of Energy, EG&G Idaho, G.P.U. Nuclear, Bechtel Northern, U.S. Nuclear Regulatory Commission and Electric Power Research Institute distribution lists. The third report covers the period of time from May 15, 1982, until August 31, 1982.

In the first period TAAG suggested that three "prompt steps" be taken to determine the condition of the reactor core and structural members. These steps were:

1. the insertion of a TV camera through a control drive mechanism lead-screw cavity - to view the internals of the reactor vessel.

2. a test of moving control rods to determine condition of those rods and the surrounding fuel elements.

3. a survey of the in-core instruments for evidence of core damage.

During the third period for which this report is prepared, the "prompt steps" were taken. Conclusions from these steps and suggested use of these conclusions in preparing for the future is a major element of this report (See Section IV A).

The report is organized in accordance with the letter of instruction regarding TAAG activities during the report period. That letter from Mr. R. C. Arnold, President, GPU Nuclear, requested the following:

For the period from May 15 to August 31, 1982, it is requested that TAAG undertake the following:

1. Continue to examine the prerequisites for early lifting of the reactor vessel head with emphasis on:

   a. The environmental conditions which must exist in the region of the head prior to and during head removal;

   b. The preparations which should be made prior to head lift in the event of incidents or equipment difficulties during or following the head lift.

2. Continue to examine methods for locating and identifying fuel in the coolant systems.

3. Continue to follow the RCS water clean-up and requirements for the chemistry of the refueling canal water.

4. Further examine methods of reducing airborne activity, including the investigation of instruments that could be used for in-plant monitoring of airborne activity.

5. Continue follow-up of:

   a. Quick Look, Rod Motion Test, and In-Core Instrument Survey;

   b. Early Head Removal;

   c. Control of radiation exposure.

I. PREREQUISITES FOR HEAD LIFT

A. Introduction

As stated in the introduction to this report, TAAG was requested to continue its review of the prerequisites for early lifting of the reactor vessel head. Emphasis was to be placed on the environmental conditions which must exist in the head region and preparations which should be made in advance for incidents or equipment difficulties.

Consistent with this request, TAAG has, during the period covered by this report, continued its review of the prerequisites for head removal. This secton of this report will discuss these prerequisites. Additional discussions of items relevant to this subject will also be found in Section IV.

This Secton is divided into the following subsections:

- General Prerequisites
- Environment in the Head Region
- Contingency Planning

Each recommended prerequisite will be discussed individually in the appropriate subsection. The order of the discussions of recommended prerequisites does not relate to their relative importance.

It is assumed that the head lift procedures will, where possible, utilize the techniques used in the past, modified as necessary to account for the effects of the accident. Hence, the prerequisites discussed herein address those aspects of the head lift that may change due to the unique Unit 2 conditions and do not address the prerequisites for a conventional head removal, many of which it is assumed will be used for this lift.

B. General Prerequisites

1. A safety evaluation of the head lift and associated activities should be prepared under GPU's direction for submittal to NRC. With respect to this report:

   a. If satisfactory to NRC, the report should use the same format at the "APSR Testing" and "Quick Look" safety evaluations.

   b. Maximum use should be made of the APSR testing and Quick Look safety evaluations. In particular, there should be no need for additional effort on the subject of criticality control other than to indicate that the APSR motion and Quick Look did not identify any new consideration not already covered by the previous reports.

   c. A revised discussion of boron dilution, reflecting head lift conditions, should be included in the report (see number 2 below).

d. A revised discussion of H_2 control, reflecting head lift conditions, should be included in the report.

e. The safety of the lifting operation itself from a load drop consideration, should be discussed. It is anticipated that this discussion should be able to show that the 1979 accident did not result in any changes to the safety considerations relating to a load drop other than those actions required to repair and requalify the crane.

f. The potential loss of coolant accident should be re-evaluated for head lift and post head lift conditions. If possible, it should be shown that a "dry" core does not present a problem.

g. Decay heat removal should be reviewed including a discussion of the need for a backup removal path and the validity of the analytical predictions. This discussion should review the methods that will be used to monitor the heatup, basis for temperature limits, time available to take corrective action (if required) etc.

h. The possibility of the existence of pyrophoric materials from the core should be discussed and the safety implications, if any.

i. Potential radiation exposure of operating personnel and the actions taken to minimize such exposure (ALARA) should be addressed.

j. Any discharges of radioactivity should be identified and quantified.

k. Because the plan may remain in its post lift configuration for an indefinite period of time, the general safety considerations of this static condition should be evaluated. This evaluation should include: 1) the possible effects of the 1979 accident and post accident conditions on the plant's long term safety; 2) the effects of the possible environment events (flood, earthquake, etc.) on plant safety. It is anticipated that it can be shown that many of the pre-accident evaluations of these events apply to the post accident conditions.

l. Identify and evaluate, as appropriate, any new short or long term water chemistry considerations.

m. New fire safety considerations should be addressed.

n. If it can be accomplished without interfering with the schedule for the preparation of the report, the Safety Report should cover the safety of the removal of the plenum. This evaluation would be limited to an essentially normal removal process and would not include an evaluation of a contingency removal process, such as cutting up the plenum in place.

2. Reactor shutdown will continue to be dependent on the presence of boron in the primary coolant. The subject of preventing uncontrolled dilution of the boron was addressed in the safety evaluation for the "Quick Look". The methods used to preclude boron dilution should be reevaluated for the head lift. This reevaluation should include, but not be limited to:

   a. The possibility that the plant may remain in a post head lift condition for an extended period of time.

   b. Overhead sources of water at reduced boron concentrations should be identified and removed.

   c. Fire fighting procedures should be changed to preclude the use of unborated water in the head or overhead areas.

   d. Consideration should be given to include the use of commercially available direct alarming boron monitoring equipment to monitor the boron concentration in the pressure vessel after head removal. This may simplify other water inventory monitoring requirements.

   e. Procedures will be required to assure boron control in the event makeup water is added during a loss of coolant accident.

   f. Where possible, the process of positively isolating dilution sources by the use of disconnects and flanges, should be extended from those used for the "Quick Look".

3. Detailed step-by-step procedures should be prepared for the removal operations. These procedures, which will be in addition to the conventional head lift procedures, should include:

   a. Crane testing

   b. Primary and secondary water level control

   c. Boron chemistry control

   d. Water chemistry control

   e. Hydrogen monitoring and control

   f. Radiation monitoring

   g. Long-term head storage

   h. Long-term layup of the plant after the head is removed

   i. Contingency procedures (see Subsection D)

4. The validity of the predicted heatup rate of the core during and after the removal of the head should be confirmed by experimental data from the reactor plant. GPU should insert into the primary coolant, through a leadscrew opening, two remote reading temperature sensors. These sensors combined with changes in the water level could be used to obtain the necessary data (see Section V).

5. Fluid systems connected to the primary coolant system that may contain contaminated post accident water should be flushed, if possible, into the primary coolant system prior to head removal.

6. Industrial safety is an important consideration now and during subsequent defueling operations including head lift. It is recommended that GPU arrange to have an independent audit of their industrial safety activities.

7. GPU recognizes the overall importance of the crane repairs and the lifting procedures to be used, to the safety of head removal. GPU's consultants, NRC, and others, will review in detail this aspect of the head removal. TAAG members have provided and will continue to provide information to assist in assuring the safety of this operation. It is recommended that:

   a. The environmental conditions that will exist during the head removal operations - radiation fields, special clothing, respirators, special contamination barriers and radiation shields, revised crane controls, communications equipment, etc. - should be considered in revising the existing head lift procedures and supporting training program.

   b. The training, maintenance, and QA programs to support the head lift operations should consider the lifting and handling problems that have occurred in the past in industry. GPU should review the problems others have had and assure they have learned from these events.

   c. Contingency plans should be developed for the head lift that recognize the unique conditions that may exist during this transfer (see subsection D).

8. Materials used inside the containment including polyurethane, herculite, etc., should be fire retardant.

9. A monitoring program should be established to assure that hydrogen pockets do not develop inside the containment.

10. A detailed Training and QA program should be used to support the head lift operation.

11. A cover plate should be provided to cover the pressure vessel opening. It is considered desirable, but not mandatory, that this cover plate should be leak tight or be able to control leakage in the event the primary system is refilled (unpressurized). This cover plate should have provisions to sample continuously the primary coolant and have inspection ports. This plate should be designed to be brought in through the personnel access hatch.

12. Prior to the removal of the head, the seal plate should be installed. It is recommended that the plate design be revised to provide a higher probability of a leak tight seal that could be relied upon for a period of several years. An all welded installation is preferred.

13. GPU should review the existing Unit 2 technical specifications to determine if any changes are required. Any changes should be identified promptly. Action should be initiated by GPU at that time to make any changes required.

14. In view of past difficulties, communication equipment for communication within, and to the outside of containment, should be improved prior to the head lift.

15. Any necessary modifications requiring access to reactor control outputs to the installed fuel transfer equipment should be completed prior to the head lift. GPU should expedite these modifications.

16. Special tests, if any, during pre-head lift or head lift operations should be limited to those tests required to support the head lift (such as the decay heat test proposed in this report). Tests to obtain accident consequence data should be limited to those tests that will not delay or interfere with the removal of the head. It is noted that EPRI is developing a test procedure for measuring head rotation at the time of unbolting. EPRI recognizes this procedure is not required for head removal and should not be allowed to interfere with the head removal schedule or operation.

C. Environment in Head Region

1. Radiations levels in the areas where work is to be performed should be controlled to less than 50mr/hr. These levels should be achieved by controlling water activity, through the use of shielding, or some combination of these two factors.

2. The airborne activity within the work area should be monitored by alarming Continuous Air Monitors (CAM's).

3. An enclosed environment with a clean air source should be provided to minimize the need for respirators and simplify contamination control (see Section IV).

4. The general work area should be under TV surveillance and the operations should be recorded.

5. In order to minimize the possiblity of uncontrolled boron dilution, all sources of unborated (below 3700 ppm) water in or above the head region should be drained and disconnected. A water exclusion area should be established.

6. In addition to CAM's, gamma alarms and monitors should be located in the working environment during the head lift.

7. The possibility of disturbing hydrogen pockets in the head volume during head lift should be considered. It may be desirable to maintain a nitrogen purge between the time the water level is lowered until the head lift is started. It may also be desirable to reestablish a nitrogen cover over the core after the cover plate has been installed.

8. The head storage stand should have provisions to accommodate limited head decontamination, limited head disassembly and auxiliary shielding.

9. The procedures and equipment to be used for the head lift should provide for contamination control during the lift and transfer.

10. During the lift, transfer and securing of the pressure vessel after the lift, no other testing or changes should be made to the reactor primary plant or supporting auxiliaries. The containment should be intact and closed.

D. Contingency Planning

1. Emergency <u>equipment</u> and <u>procedures</u> should be readily available to respond to:

   a. A contaminated injured man in a high radiation field (various incontainment locations).

   b. Sudden increase in airborne or activity.

   c. Fires within the water exclusion area.

   d. Loss of power.

   e. Loss of coolant (if required).

   f. A radiation spill (large).

   g. Loss of decay heat removal (if required).

   h. Reduction of boron concentration.

   i. Loss of communication (to crane or out of container).

2. The head removal and transfer procedure should be based upon the use of standard procedures modified as necessary to reflect specific post-1979 accident conditions. The procedures should not attempt to provide contingency steps in the event a significant operation does not go as planned; for example, stud removal. The procedure should require that operations be terminated and the plant placed in a safe condition (defined) in the event difficulties are encountered either with the execution of the procedures or with the support equipment. In the event of difficulties, alternate procedures can then be established for the specific difficulties encountered.

3. It is recognized that it is likely that minor changes to the procedures may be desired as the head lift operation progresses. To facilitate these changes and at the same time assure proper review, test group having representatives from the Unit 2 operations, engineering and safety organizations should be established. This group will have the authority to determine that a change is consistent with the original procedure intent, and does not introduce a new or change an existing safety consideration. Under these conditions, and subject to NRC approval, the group (under unanimous agreement) may approve on the spot procedure changes.

4. The source range neutron instrument monitoring limits, established for the APSR test, should be used for the head lift.

5. The capability to flood (partially) the canal and control the canal water chemistry and radioactivity levels should be provided prior to the head lift.

II. Locating and Identifying Fuel in the Systems

A. Background

Early TAAG inputs from may experts revealed surprising differences (tons vs kilograms) among some intense convictions regarding the amount of fuel debris outside the vessel (and in the bottom of the vessel).

TAAG's concern, particularly before Quick Look results were available, involved

- Possible impacts upon defueling approaches. (How much defueling is required; how much fuel is in the bottom of the vessel, etc.)
- Equipment and techniques required to flush or otherwise remove, encapsulate and ship ex-vessel fuel off-site.
- Evaluation of potential problems in decommissioning demineralizers and other components with resins or other materials, which might be significantly affected by years of exposure to irradiated fuel.
- Possible "surprises" from unexpected neutron levels during recovery activities.

The large uncertainty seemed to call for rapid, qualitative screening for large amounts of debris in various locations. Chemical analyses, several gamma detectors, and neutron dosimeters were techniques considered. Chemical analyses of make-up filters gave the first evidence of significant loss of debris from the vessel, although subsequent checks revealed gram-size rather than kilogram quantities of fuel.

Below is a survey of candidate techniques, with TAAG conclusions and recommended actions.

B. Gamma Detectors for Fuel Debris Assaying

Alternative gamma measurement methods which have been proposed and have the possible advantage of being directonial include the Cd Te detector, the Ge gamma spectrometer, the Si (Li) Compton recoil gamma spectrometer and a Be-fission technique for generating and detecting gamma induced photo neutrons. A comparison of these various gamma detection methods (excepting Cd Te) is summarized in Table II-1.

C. Passive Dosimeters

Passive dosimeters have been useful in assessing neutron sources. The principal candidates are solid state track recorders (SSTR) or activation foils (AF). They are small, inexpensive, and relatively easily used for mapping large areas and for reaching

"inaccessible" locations. Their use is being considered for dosimetry studies of the Make-up and Purification Demineralizer System. This system was initially considered for SSTR dosimeter calibration by TAAG but the present program is independent of TAAG activities. However, the program may provide an adequate check on the feasibility of dosimeters for inexpensively locating any substantial quantity of ex-vessel (or bottom-of-vessel) fuel debris. This may be the most cost effective method. There is an on-going DOE/EG&G/GPU/EPRI evaluation of candidate passive detectors to check for possible concentrations of debris in other locations. The detectors are available.

D. Conclusions and Recommendations

Efforts to locate ex-vessel fuel debris must reflect the limited funding and higher priority tasks such as crane repair, head removal and decontamination. However, there are strong convictions among some members of TAAG and well informed engineers in GPU, EPRI and other organizations that, if a very limited amount of entry time (as expected) and a relatively small amount of funding (tens of thousands) are required, it is timely and appropriate to try to determine whether kilograms or tons of debris are to be expected at locations outside the reactor vessel.

Evaluations of Quick Look results by several groups of experts may help resolve some major uncertainties regarding debris location. Additional information also will be provided by the Demineralizer program, which may use both gamma and neutron detection methods.

TAAG/EG&G recommends a small scale check with neutron track recorders for debris which might have accumulated on the top of the tube sheet of one or both steam generators. Although the recorders operate in the presence of borated water, the present lowering of coolant from the generators would increase the sensitivity of the neutron recorders. "B" generator was initially recommended by TAAG. Steam generator "A" might be more interesting to explore because of additional fuel movement from "A" pump operations. Both generators should be considered. DOE/EG&G/GPU/EPRI and other personnel are moving ahead with plans to install the SSTR trackers.

TAAG concludes that actions to find an adequate gamma detector and to employ both gamma detectors and neutron detectors to locate fuel are appropriate a joint working group - EPRI, ORNL, GPU - is handling the gamma detector development.

Table II-1

Gamma Measurement Techniques for Fuel Determination

	Ge-Li	Si-Li	Be Fission Chamber	Be U^{235} Track Recorders
Function	Measures γ spectrum (energy & #)	Measure γ spectrum (energy & #)	Measures high energy γ's (#) >1.67 MEV	Measures high energy γ's (#)
Radiation Field per hour	0.1 R	5 R	100,000 R	100,000 R
Shielding Required	800 lbs. *	50 lbs.	None	None
Principle	Compton recoil pair production photo electrons	Compton recoil	$Be^{9}(\gamma, n)$ $U^{235}(n, f)$	$Be^{9}(\gamma, n)$ $U^{235}(n, f)$
Support Electronics	Yes	Yes	Battery Powered	No
Cooling	Liquid N_2	Air	None	None
Equipment/Data Available	(45 days/50 days)	30 days/35 days	30 days/35 days	2 wks/8 wks
Costs ($000's)	90 800# transporter	88 60# transporter	65 60# transporter	20 stringer
Concerns:				
1. Unkown source distribution scattering from tank contents	Counter saturated s/high Cs level	Scattering and collimation makes calibration uncertain	Response of detector to TMI environment	
2. Ce^{144}/fuel ratio uncertainty	More scattering from shielding makes calibration difficult		Ratios of Ru^{106} to Ce^{144}	

* NOTE: Maybe reduced by careful design and use of tantalum

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III. CONTROL OF WATER QUALITY IN THE RCS AND REFUELING CANAL

CONCLUSIONS AND RECOMMENDATIONS

Based on anticipated fission product leach rates, TAAG concludes that the ^{137}Cs concentration in the refueling canal can be maintained at or below a few hundredths of 1 μCi/ml, which would contribute a radiation level in the vicinity of 10-20 mr/hr over the canal.

Based on prior experience elsewhere and very limited data from TMI, the maintenance of adequate water clarity for direct viewing of defueling operations cannot be assured, so supplemental viewing should be provided for. However, the contribution to dose rate should be smaller from particulates than from soluble fission products.

Dose rate considerations, by themselves, do not provide a justification for a barrier to separate the region over the core from the rest of the canal. There may be other reasons for a barrier, such as control of suspended material and water clarity.

TAAG recommends the following actions to provide for the maintenance of water quality:

1. Examine the installed water handling and filtration equipment in the spent fuel cooling system for applicability to water clarification and collection of core debris. Modify and up-grade as necessary, or replace with a new filtration system of at least equal capacity (400 gpm).

2. Upgrade SDS throughput by a reconfiguration to four parallel lines of two columns each, with adequate pump capacity, piping, valves, etc., hr 10-15 gpm each line, 40-60 gpm total. This capacity is expected to be adequate for control of ^{137}Cs activity. The ion exchange column in the spent fuel cooling system is not appropriate for this problem.

3. Provide interface for a backup system of 50-100 gpm capacity (such as EPICOR), using the same chemistry as SDS. It would be used only if required by unanticipated activity levels.

4. The vacuum cleaner system should be installed as an integral part of the water processing system, with provision to route the vacuum discharge to either SDS or the main filter system [(1) and (2) above].

5. Provide proper system interconnections to route water from the vacuum (when operating), or from the core region (near the location of fuel removal operations), to the filter system and SDS, with return to the canal at a location remote from the core (either the refueling canal or the A spent fuel pool).

6. Continue consideration of a barrier for purposes of control of suspended material, containment of contamination within the canal, and work operations above the canal.

7. A process for removal of deleterious chemical impurities (notably chloride and sulfate) should be developed in case a need arises later on.

INTRODUCTION

Two problems, maintenance of water clarity and control of radiation levels, present problems during reactor refueling; and in a number of instances in which there had been corrosion problems or some degree of fuel damage, maintenance of water quality has been a severe problem. It is clear that TMI represents a level of fuel damage that has never been approached before, and there is no apparent way even to estimate the extent of the problem from solids that may become suspended in the water. On the other hand, the extent of fission product leaching from damaged fuel can be estimated, but only with substantial uncertainty. Accordingly, the water cleanup problem goes so far beyond prior experience that it really cannot be evaluated with the degree of confidence that one might desire.

It appears that a combination of a high capacity filtration system for clarity control and a decontamination system based on Submerged Demineralizer System (SDS) technology for dissolved fission product control is best suited to meet the requirements. (It should be noted that SDS, contrary to its name, does not demineralize the water.) The success of the cleanup approach used at TMI during the past year (SDS) and modified Epicor) has probably led to an underestimation of the impending problem, because the requirements during defueling will be rather different. Instead of decontaminating something like a million gallons in 6 months or a year, it may be necessary to process that much every week or two; and the filtration requirement may be ten-fold greater than that. At the same time, waste generation must be kept in hand.

This Section will consider the requirements for water quality control in the refueling canal, as best they can be estimated at this time, and the most efficient and available means to meet those requirements. The existing canal water filtration equipment in the spent fuel cooling system may be adequate for the task, but the canal water ion exchange system is not appropriate. The SDS ion exchange technology is suitable, but the existing system has inadequate throughput. Modifications to re-pipe the columns and incorporate other changes that may be required to increase capacity and to assure a very high availability factor are identified, but it is still not certain that the existing system can be upgraded sufficiently. Accordingly, a supplemental back-up system will probably be necessary. In addition, if a "vacuum cleaner" approach is used to recover degraded fuel from the primary system it will require its own filtration equipment and possibly also an ion exchange system, since this is the most likely source of increased fission product leaching.

WATER DECONTAMINATION SYSTEMS

SDS BACKGROUND

The SDS system was designed to remove cesium and strontium from the water in the reactor building after the accident. SDS utilizes inorganic zeolite ion exchangers, in contrast to the organic ion exchangers used in all other reactor water processing. Development studies that led to SDS were initiated in May, 1979, at Oak Ridge National Laboratory and the Savannah River Laboratory, under direction of the Technical Advisory Group chaired initially by B. Rusche. A process flowsheet was proposed, and further experimental evaluation was carried out after reactor building water samples were acquired in August, 1979, and again at the beginning of 1980. Based on the modified flowsheet, the SDS system was designed by Allied General Nuclear Services for Chem-Nuclear Systems, Inc., the prime contractor for fabrication, installation, and operation of the process equipment. Subsequently, GPU assumed responsibility for the SDS system. Initially, water processing was viewed with some urgency, but for a variety of reasons system construction was delayed, and operation did not commence until the late summer of 1981.

In the meantime additional development was carried out from time to time to increase the loading of the system and to improve its effectiveness for strontium (as well as for cesium) removal, and also to improve the performance of the Epicor system (which was being used at TMI for decontaminating low-level water) for polishing the SDS effluent by removing residual cesium, strontium, and anionic fission products. The original SDS flowsheet evaluation was reported in ORNL/TM-7448, and subsequent improvement studies and evaluations in ORNL/TM-7756, ORNL/TM-8333, and DOE/NE-0012 (Ref. 1, 2, 3, 4).

The system as installed consisted of eight identical columns (Fig. 1). There were two parallel lines, each containing three zeolite ion exchange columns in series; these were followed by two more columns in parallel, intended for polishing using organic ion exchangers. There was some flexibility in the column interconnections, which utilized flexible hoses fitted with Hansen disconnects. By the time the system was operated the column interconnections were changed such that one line consisted of four columns in series and the other used only two columns in series. All SDS columns are operated with zeolite ion exchangers which remove nearly all the cesium and strontium, but nothing else.

The "polishing" step with organic ion exchangers was carried out in the Epicor-2 equipment using a modified Epicor process that consisted of (1) removal of sodium ion, along with residual cesium and strontium, from the water using a hydrogen-form strong acid cation exchange resin, followed by (2) removal of anions and some colloidal activity with anion exchange and mixed-bed resins (Ref. 1).

Reactor building water was processed through SDS using conditions selected to yield maximum decontamination. This operation used a single train of four columns in series, all containing the same mix of two zeolite ion exchangers, and an unusually slow flow rate to improve strontium removal. After loading until significant breakthrough occured through the first column (but not the last two), the first column was removed from service; the second, third, and fourth were each advanced one position; and a new column was placed in the last position. For various reasons, a system throughput of 88,000 gallons (two batches of 44,000 gallons each) was finally selected for each cycle before column replacement, although substantially more water was processed in some early runs. The product water was further decontaminated by processing through the modified Epicor system using organic ion exchange resins. The goal was to decontaminate the water to the maximum practical extent.

During the time that reactor building water was being processed (and since then) substantial volumes of water containing lower concentrations of radionuclides were collected from various sources in the plant. This waste water was processed during intervals between processing batches of reactor building water, using the second line of SDS, but with only two columns in series rather than four. In this case, as before, a high decontamination factor was desired, so the product water was polished using the modified Epicor process.

Following processing of all the available reactor building water earlier this year, SDS has been used to decontaminate reactor coolant system (RCS) water; and to date five batches of about 50,000 gallons each have been processed. In this operation a large decontamination factor is not necessary as long as the bulk of the radioactivity is removed, so polishing by the modified Epicor process is not used. For the same reason, constraints on the SDS process have been relaxed somewhat (use of two columns in series instead of four), and further relaxation is clearly practical (notably increased flow rate). It is in this context that the SDS system can be considered for further cleanup of the RCS and for continued control of the refueling canal water throughout the fuel removal operation, because the process, as initially employed, had inadequate capacity, particularly for the latter task.

PROCESS BASIS

The SDS process differs from conventional ion exchange processes used in other nuclear plants (and at TMI prior to the accident) in that it uses inorganic zeolite ion exchangers instead of organic ion exchange resins. Zeolites are inorganic aluminosilicates with a rigid network of very small pores, defined by the particular crystal structure, and with cation exchange capability because of exchange sites inherent in the chemical composition. There is a large base of information on the equilibrium ion exchange behavior of many zeolites (Ref. 5), but some information essential to design of a practical system is missing-- notably that relating to kinetic factors such as diffusion rates of ions within the pore structure. Zeolites were selected for two primary reasons: (1) because of their extremely high selectivity for cesium and strontium (the overwhelmingly dominant radioactivities present in the water) relative to sodium ion (which is the cation competing for ion exchange sites), and (2) because of their resistance to radiation damage, compared to organic ion exchange resins.

Organic ion exchange resins show little selectivity for cesium over sodium, and they are relatively susceptible to radiation damage. As a result, although organic ion exchange resins are suitable for cleanup of solutions containing very low ionic concentrations and reasonably low radioactivity (which is the normal situation for RCS and canal water cleanup, for example), they are neither effective for decontaminating solutions containing appreciable concentrations of competing ions nor stable if loaded with high levels of radioactivity. In contrast, certain zeolites are effective even under the TMI conditions.

The RCS and canal water to be processed will contain about 800 ppm of sodium ion (about 0.035 M), which is a result of the requirements that the boron concentration be > 3500 ppm for criticality control and the pH be > 7.5. In contrast, RCS water normally contains boric acid at a lower concentration, essentially no sodium ion, and perhaps a few tenths of 1 ppm of lithium ion, for a cation concentration about 1000--fold lower than that in the TMI water. As a rough correlation, the capacity (before column breakthrough) of an ion exchange column for removing cesium is inversely proportional to the competing cation concentration, or about 1000 times smaller for TMI water than for normal reactor water. Thus, ion exchange column life, in either throughput volume or time at a given flow rate, would be some 1000 times shorter during the TMI fuel removal than during a normal refueling, and waste generation would be 1000 times greater. (In an analogous way, the strontium capacity is inversely proportional to the square of the competing ion concentration, for a factor of a million difference.)

In practice, then, when processing water of the composition of that in the RCS and refueling canal, using organic ion exchange resins, cesium would be retained by the ion exchange column for only a few bed volumes (or typically a few minutes operation). This projection is supported by the experience with the cleanup column for SDS leakage containment (which used organic resins) when there was a leak in an SDS system disconnect. In contrast, certain zeolites that are selective for cesium removal are effective for a few thousand bed volumes of throughput, and it is one of these (chabazite) that was selected initially for SDS operation.

Cesium is the dominant gamma activity in TMI water, but there is also a substantial amount of strontium activity. Strontium could be removed by either organic or inorganic exchangers, but another zeolite, Linde A, is extremely effective for strontium removal (chabazite is moderately effective). Accordingly, a mixture of these two zeolites was selected for SDS, and both radionuclides (but little else) are effectively transferred from the water onto an inorganic medium that is amenable to waste management. The mixture ratio is selected so that both cesium and strontium will break through the column at about the same time. This is not completely straightforward since the optimum mixture depends on both the composition of the water being processed and the processing flow rate.

Laboratory testing of the process was carried out at Oak Ridge National Laboratory, and system performance was projected by a computer program based on the second order kinetics model of Vermeulen (Ref. 6). Performance of the actual system was correlated with the model, and in practice sytem performance has been predicted rather well. Thus, parameters for operation of a system based on SDS chemistry can be calculated to satisfy a given set of requirements, such as those for TMI fuel removal, with reasonable confidence.

Performance of both the SDS process and the SDS system has been remarkably successful-- so successful, in fact, that a false sense of security has appeared. It is necessary to examine the constraints imposed on the process by its application to the long-term problem of maintaining the radiation level low enough in the refueling canal water. This will establish the requirements for the SDS process and delineate the actual operating conditions. Then a decision can be made as to whether or not the installed SDS sytem can be modified to meet these requirements. At the same time, the problem of water clarity must be addressed.

EXISTING CANAL WATER CLEANUP SYSTEM

The existing spent fuel cooling system includes canal water cleanup equipment that contains both filtration and ion exchange capability. As pointed out in the first TAAG report, at least part of this system is probably not suitable for the projected operation, namely the ion exchange column, which is a 21 cubic foot column rated at 180 gpm and designed for a mixed bed of organic ion exchange resins. However, the pumps (two rated at 1000 gpm each) and filters (two rated at 200 gpm each), may be useful if they are adequately shielded to handle the higher radiation level. Potential problems relate to the ability to replace, ship, and dispose of the filters, as well as the suitability of the filter material and design for the actual solids to be removed from the water.

As noted above, organic ion exchange resins are not suitable for decontamination of this water, and it is unlikely that zeolites could be used in the existing ion exchange column because they could not be transferred in and out. Even if they could be, their eventual disposition still would present a problem. Furthermore, since zeolites load radionuclides to a much greater concentration than organic resins, shielding and handling provisions for the existing system may be inadequate. Thus, it appears advantageous to use an ion exchange column that would also serve for shipping the waste to the disposal site so that problems from vessel-to-vessel transfers of the contaminated material can be avoided.

If the existing ion exchange column is not used, options are to replace it with a column of appropriate design, to use another existing system if appropriate, such as SDS, or to build a new system. As shown below, it may be practical to upgrade SDS sufficiently for this application. However, there is clearly enough uncertainty in all the assumptions that one cannot be sure the requirements can always be met. Accordingly, a back-up system of higher capacity and with adequate shielding should be provided, although it may never be required.

REQUIREMENTS DURING FUEL REMOVAL

The previous TAAG report stated there is a consensus that cesium leaching from the core now amounts to about 2 Ci/day, and that activity spikes (which are much more conjectural) may range between 20 and 100 Ci. It was also noted that water clarity can be a critical aspect, although no quantitative conclusion were drawn. Finally, it was pointed out that control of chemical impurities, notably chloride, may become a problem. These matters are evaluated further here.

Water management during fuel removal may become complex, and careful consideration is necessary to devise an effective system that is operationally simple while keeping the waste volume within reasonable bounds. Two factors that impact on this are (1) the effect of a "barrier" of some sort to partially isolate the water within the reactor vessel and a working region above it, from the rest of the canal, and (2) the impact that a water "vacuum cleaner" for recovering core debris may have on water cleanup.

CONTROL OF LEACHED FISSION PRODUCTS

It has been concluded that the steady-state source term for ^{137}Cs leaching from the fuel into the RCS over the last many months have been approximately 2 Ci/day. The hope that a better value could be derived during RCS water processing has not been fulfilled to date because other factors exceed and obscure this source. These factors include analytical limitations, slow mixing within the system, and other sources such as solutions partially isolated from the RCS mainstream. Indications to date are that the main RCS loops have a circulation time of the order of one day, presumably due to thermal convection, and water in the pressurizer mixes into this system on a much slower time scale--perhaps a few months. It is possible that a better description will be obtained for the source term and mixing within the system later on, after the RCS system is refilled and water decontamination is resumed; but at this time the 2 Ci/day figure stands.

The RCS also contains a substantial concentration of ^{90}Sr which apparently is not a result of fuel leaching. It remains at a fairly constant value, suggesting solubility of a slightly soluble strontium compound. The strontium concentration may not decrease significantly until this source is exhausted. Strontium leads primarily to beta exposure and airborne activity.

A simple mass balance equation relates the parameters involved with the cleanup system. At steady state the rate of introduction of activity into the water (leach rate) must equal the rate of removal in the cleanup system, or

$S = (5.5)\ (RFC)$, where

S is the leach rate in Ci/day,
C is the steady state concentration of the canal water in μCi/ml,
R is the fraction of the activity removed per cleanup pass,
F is the flow rate through the sytem in gpm, and
5.5 is the factor to systematize units.

For example, since R is close to 1 if the system is operated properly, and S is assumed to be 2, FC = 0.36; thus, a steady state concentration of 0.01 μCi/ml requires a 36 gpm average processing rate, and 0.1 μCi/ml requires 3.6 gpm.

If the leach rate should increase during fuel removal, either the processing flowrate must be increased or the activity level of the canal water will increase, both in proportion. This relationship is independent of canal volume, but the rate at which the system responds to changes in any of the variables is dependent on volume, being slower the larger the volume. The relationship between F and C is shown in Fig. 2 for source terms of 1, 2, 5, and 10 Ci/day.

Additional insight into core damage, obtained during the Quick Look examination, suggests that water vacuuming will be used extensively to remove loose core debris. In addition, if mechanical operations are used to degrade larger pieces of the core for easier removal, they will most likely generate small particulates; and vacuum removal of these is probably necessary to maintain water clarity. Such vacuum operations will expose debris to rapidly flowing water and, at some point, will encounter newly-exposed surfaces that may yield higher leach rates than the present core. Thus, the vacuum, in the process of removing particulate material, may become a significant source of soluble activity. As such, it must be considered in the context of the canal water cleanup system, as well as just a tool for debris removal.

It is obvious that the vacuum not only must recover solids from the water, but it also must yield a very clear effluent. It now seems possible that a substantial fraction of the core may be removed in this manner, so it might be preferable to collect most of these solids in containers suitable for shipping or for transfer into shipping containers. Some fraction of these solids, however, will be present as finely divided material that will present a problem in regard to filtration. Thus, it may be necessary to have several stages of solids recovery, for large particles representing the bulk of the material, and for successively smaller-sized particles that constitute a decreasing fraction of the core material but the greatest threat to maintaining water clarity. Finally, if enhanced fuel leaching occurs in this system (which is possible after the initial removal of loose debris), it would be advantageous to decontaminate the clarified water via zeolite ion exchange before returning it to the canal. This would require that the vacuum discharge be treated in a dedicated ion exchange system or be fed into the main ion exchange cleanup system, rather than be returned directly to the canal.

TAAG recently initiated investigation of the characteristics of a vacuum system that might be applicable to TMI, but little specific information has been obtained to date. To some extent the problem can be placed in context if it is assumed that fuel pieces up to 1 cm diameter spheres (somewhat similar to single fuel pellets) are to be fluidized; a velocity of about 5 feet/sec is required. In a 1 inch diameter pipe this corresponds to about 12 gpm. The canal decontamination system could process this water since it should have a capacity greater than this.

The important conclusion is that the capacity requirement for the canal water cleanup system is reasonable for the anticipated steady state leach rate (2 Ci/day) and for water contamination levels that seem within the realm of practical fuel removal operations (probably 1 to a few hundredths of a μCi/ml); throughputs of 10 to 30 gpm may be adequate. However, a higher source term, which could result from extensive vacuum operations, especially if mechanical fuel disassembly methods are used (such as sawing, grinding, or drilling), would require a correspondingly higher processing capacity. As a result, the fuel removal operations may have a controlling influence on the canal water cleanup sytem.

CONTROL OF FISSION PRODUCT SPIKES

It was stated in the previous TAAG report that spikes of 20 to 100 Ci of ^{137}Cs may be expected. Other fission products may also be present, but cesium is the dominant gamma emitter. Upon further evaluation it appears that large spikes (near 100 Ci) are so unlikely that they should not be used for a design basis. At the other extreme, frequent small spikes are better treated as part of the steady-state leach rate.

Consideration of mechanistic models that could suddenly introduce 100 Ci of ^{137}Cs into the water provides some perspective on the likelihood of such large spikes. Water trapped since shortly after the accident (over 3 years) would contain about 300 μCi/ml, and 100 Ci corresponds to 87 gallons of such water. Possibly such a quantity could have been isolated and could be introduced into the RCS, but probably not as an inadvertent event resulting directly from fuel removal. Subsequent to the accident the RCS concentration decreased to about 25 μCi/ml by early 1981, 15-20 by 1982, and <5 since July, 1982. Thus, a 100 Ci spike would require a much larger volume unless the water had been isolated shortly after the accident.

The Quick Look inspection does not support the suggestion that such large pockets of water remain isolated within the reactor vessel, itself. Hardware that might contain trapped water that could be released by manipulation in the core includes the internal volume of the incore instrument assemblies (about 3 gallons total) and the volume between the assemblies and their guide tubes (about 100 gallons, but initially filled with clean water, so only a small fraction should be high concentration water). Therefore, it seems unlikely that sudden, large releases can originate from the core region.

Piping systems connected into the RCS but isolated at some time during or after the accident could be valved into the system, yielding a one-time spike. There are known volumes that are isolated or that mix rather slowly with the RCS. For example, the pressurizer was flushed with clean water during RCS processing, and analytical data indicate it contained cesium at a concentration characteristic of the RCS a few months earlier. The decay heat removal system is another possible source of high level water. It is important that all such systems be identified and backwashed with decontaminated water. There are only a few such systems; they are known; and it is essential that they be dealt with during RCS cleanup prior to fuel removal. In fact, this should be a prerequisite to head lift.

Dispersal of 10 kg of average (undamaged) fuel particles throughout the canal water could introduce 100 Ci of cesium, but it would have to be uniformly dispersed and remain suspended to provide the same gamma source as dissolved cesium. Fuel fines could be introduced by cutting or other operations related to fuel removal, but the fines should be collected as produced so that they do not enter the large canal system. Other removal mechanisms are natural settling and filtration with the canal cleanup system. Since the canal filtration capacity will be much larger (200-400 gpm) than the ion exchange capacity, removal of suspended radionuclides will occur much more rapidly; and recovery from a spike of suspended fuel would require much less time than from a soluble source. Other aspects of the problem from suspended solids are discussed below.

These considerations lead to the conclusion that a large spike (the order of 100 Ci) is extremely unlikely to result from core removal operations, but a spike could occur from manipulation of valves associated with the RCS. Such an event, however, should be very rare-- a one-time thing-- and not entirely unanticipated. As such, it does not provide a basis for normal system design purposes. A spike of the magnitude of 20 Ci is more conceivable. However, the cleanup implications of spikes of 10 to 100 Ci will be considered below.

IMPACT ON CANAL BARRIER

The impact of fission product spikes is strongly influenced by the use of a barrier to isolate to some extent the core, reactor vessel, and a region above them, from the rest of the refueling canal, as well as by the water management approach used. Here, it will be assumed that either the entire canal is well-mixed and processed for fission product removal, with no barrier being used, or only the volume within the barrier is processed. In the latter case there might possibly have to be another cleanup system to treat the canal water outside the barrier.

The ^{137}Cs concentration was calculated as a function of time after a spike occurs, for reasonable combinations of system volume, cleanup flow rate, steady-state leach rate, and spike size (Figs. 3, 4, 5, 6, and 7). Volumes of 40,000 and 400,000 gallons were assumed, the first corresponding to the reactor vessel and barrier region, and the second to an open canal with no barrier. Spikes of 10, 20, and 100 Ci were considered. For all cases, 2 Ci/day was assumed for the leach rate, and 30 gpm for the processing rate. For other processing rates the time scale would be changed in inverse proportion. The 30 gpm value is believed to be representative of the present SDS system modified as indicated below. (For removal of suspended activity by the filter system the processing rate would be much larger.)

It is apparent that the concentrations are initially much higher (4 to 9 times) for the smaller volume (use of barrier), and that they decay away much more quickly, reaching a lower value after 57 hours for these conditions. The integrated area under the curves (attributable to the spike) is the same for both system volumes. For example, with a 10 Ci spike and 30 gpm processing rate integration gives 1.46 in units of μCi/ml in the water times hours (a number that can be related to personnel exposure).

One may estimate the impact of a hypothetical spike on radiation levels near the water, where the people carrying out the fuel removal operation will be working. As a very rough rule of thumb, 1 μCi/ml of cesium in the canal water causes a radiation field of nearly 1 r/hour over the canal, but somewhat less as the surface area of the pool is decreased, as inside a barrier. On this basis, the time-integrated radiation field from a 10 Ci spike would be 1 to 1.5 r, and for a 100 Ci spike 10 to 15r.

It must be remembered that there will be a general but somewhat variable (with location) gamma background in this entire area, very likely in the range of 50 to 100 mr/hour. A reasonable desire would be for the canal to contribute an additional exposure small compared to this, perhaps 10 to 20 mr/hour. This would imply a cesium concentration in the canal water of 0.01 to 0.02 μCi/ml, or somewhat higher if the barrier concept is used. It is clear from the the figures that concentrations (and associated radiation fields) quite large compared to these may exist over the barrier region, especially in the case of large spikes which could give several hundred mr/hour. On the other hand, if the barrier is not used the greater dilution in the canal limits the increase in radiation field to a value comparable to the ambient, even after a 100 Ci spike.

If large spikes are excluded, then the increase in radiation field is rather small if a barrier is not used (generally 20 mr/hour), but comparable to the general background if the barrier is used. Qualitatively, it appears that there would be more incentive to interrupt work temporarily (evacuate), following a 10 to 20 Ci spike, if the barrier concept is used than if it is not. If only relatively small spikes are anticipated they will have an effect more like an increased leach rate, leading to a relatively small increase in radiation level.

The major impact, if a barrier is used will be localized radiation fields following a spike that are substantially higher, but that decay away more rapidly, yielding a lower radiation field after about two days. The choice, to a large extent, comes down to the effect of the short-term high radiation field with the barrier, as opposed to the longer-term but smaller increase in radiation level without the barrier. The magnitude of the effect is directly dependent on the size of the spike, and for smaller spikes the ambient background will dominate radiation exposure anyway. In that case, there appears to be little advantage, from the point of view of radiation exposure, to use of a barrier, and there could be more interference with ongoing operations. In the final analysis, factors other than radiation field (such as water clarity and spread of particulate material) will probably determine the choice.

CONTROL OF CHEMICAL IMPURITIES

The previous TAAG report considered the maintenance of water quality, and particularly the potential problem from chloride ion. Attention has been called to water quality because of the steam generator corrosion problem in TMI Unit 1. During the core removal operations there is some probability that chloride will accumulate to a concentration above 5 ppm in the RCS and refueling canal water. Should this occur, a process to reduce the chloride concentration should be available.

Demineralization (used normally to maintain water quality during refueling) is not effective at TMI because of the high concentrations of sodium and borate ions in this water (which result from the 3500 ppm boron content required to prevent criticality and the pH > 7.5 specification to control corrosion and fuel dissolution). Since water decontamination with respect to fission products is expected to be required throughtout the operation, a process incorporated in or compatible with the water cleanup would be advantageous.

A brief examination has been made of methods that might be used to remove undesirable constituents from Unit 2 RCS (or eventually the canal). The modified Epicor process used to polish SDS effluent from the cleanup of reactor building water could be used to remove impurities such as chloride and sulfate, but at the expense of generating a large volume of waste. The chemistry of this process is reasonably well understood, and there is much operational experience. The process is based on (1) removing sodium ion with a hydrogen--form cation exchange resin (which converts borate ion to non-ionized boric acid at the same time), followed by (2) removing anionic constituents along with some colloidal material with anion exchange and mixed bed resins. This effectively removes residual radionuclides (cesium, strontium, antimony, ruthenium) and anions such as chloride and sulfate. The anions are not removed unless sodium is removed first, because of competition for ion exchange sites by the great excess of borate ion.

The disadvantage is that this is an "add-on" process not required or planned for use for water decontamination and as such it will generate additional waste in fairly large amount (in the range of 2 or 3% of the volume of the water processed for chloride removal). The greatest part of this waste is cation exchange resin required to remove the large quantity of sodium in the water. (It may be noted that a very large volume of water must be processed. To remove half the chloride from 400,000 gallons of canal water, for example, it is necessary to process 280,000 gallons containing nearly a ton of sodium, which would require about 5400 gallons of cation exchange resin.)

A second possible option would be a modification of the SDS process that might remove chloride, along the cesium and strontium it normally removes, with only a very small increase in waste generation. Some form of the SDS process will presumably be used to maintain low radiation levels in the water in any case. The chemistry of the modified process is based on literature data, but mechanical problems with respect to operability are completely undefined. The process is undemonstrated and, in fact, it may not be operable in the TMI environment. Therefore, experimental testing is required.

The process is based on incorporation of silver in the zeolite mixture used in SDS (in addition to the IE-96 used presently to remove cesium, a smaller amount of a different zeolite, such as Linde X, would be added) to establish a controlled concentration of silver in the water processed through the system. The silver concentration would be maintained at a level chosen to precipitate chloride, as silver chloride, leaving a specified (but lower) dissolved chloride concentration in the water (in the range of 0.1 to 1 ppm).

The water would then pass through a subsequent column containing a different zeolite to remove the excess silver, yielding water suitable for return to the system. The latter column would probably contain Linde A, the zeolite used presently in SDS to remove strontium. The main uncertainty in the process relates to the mechanical behavior of the precipitated silver chloride--if it will precipitate in the columns and interfere with flow.

A laboratory investigation would be required to verify the projected effectiveness of the process, to estimate the existence or severity of any operational problems, and to define the flowsheet and operating conditions. Such a program is estimated to require approximately 3 months after approval, and to cost $30,000. The benefit, to be compared to the cost, is the much smaller waste volume (0.1% of the volume processed) if the process is successful. If it is sufficiently promising a full scale test may be warranted, probably at TMI.

CONTROL OF SUSPENDED SOLIDS

It was shown above that a water cleanup rate in the range of 20-30 gpm may be adequate for controlling soluble fission products (^{137}Cs) from the canal water if the fuel leach rate is (and remains) as low as expected. Furthermore, the SDS technology is particularly suited to this task. In this section the problems related to removal of insoluble material and the maintenance of water clarity will be considered.

The anticipated source term for suspended solids cannot be estimated with much confidence. The problem was considered at some length by an SAI group (Ref. 7), with the conclusion that the solids content of the water may be quite large. The mass of suspended particulates in RCS water was estimated at 40 to 500 kg to start with, and the source term during debris removal at 13 kg/day. Processing of RCS water since the estimate was made has probably reduced the initial solids content considerably. It should be pointed out that these numbers refer primarily to corrosion products (largely iron oxide) and not to fuel.

The problem from suspended solids relates primarily to clarity, and the statement is commonly made that 1 ppm suspended solids (corrosion products) is about the maximum that permits good visibility through 30 feet of water. This corresponds to about 1.5 kg in the refueling canal. The Quick Look examinations indicate quite limited visibility above the fuel (compared to 30 feet), but this may have been influenced by debris suspended during the operation. In general, samples obtained from the reactor vessel suffer from the same interference. At this time, there appears to be no satisfactory way to estimate the severity of this problem.

Fuel removal operations under "normal" conditions frequently encounter problems from water clarity, and with damaged fuel this problem, as well as that from increased radiation levels, has generally been significant. The implications from past experience are that water clarification processing rates of several hundred gpm will almost certainly be required, and there is no certainty that even this will be adequate. Higher processing rates can be achieved only with considerable difficulty.

If the above estimates are approximate correct water clarity probably cannot be maintained, and supplemental viewing means will be required. For example, to maintain 1 ppm suspended solids with a 13 kg/day source the clarification processing rate must average about 2400 gpm. With a filtration rate of 400 gpm (the installed system capacity) the steady state concentration (ppm solids) would be about half the solids suspension rate (kg/day). In addition, the solids may create a radiation exposure problem, depending on their fission product content. Thus, 1 ppm suspended fuel in the canal would contribute about 0.01 μCi/ml. For visual operations water clarity problems would probably prevent work before increased radiation levels would, but both must be considered. (During normal refueling radiation arises primarily from activated corrosion products, which should be a minor contributor at TMI.)

Because of the great uncertainty in defining the problem it is probably the best course at this time to thoroughly assess the applicability of the installed canal water filtration system to this problem, and if practical, to make any modifications necessary for its application at the design throughput of 400 gpm. This would include consideration of filter construction and media, shielding, transfer, and shipping, as well as radiation exposure problems elsewhere in the system. In the event the existing system cannot be made useful a new system of comparable capacity sould appear to be required. Such a recommendation is based largely on prior refueling experience, and one may expect no less of a problem here.

RECOMMENDED SYSTEM MODIFICATONS

There are three important systems that require upgrading if effective management of water quality is to be achieved. These are the water clarification system (including vacuum), the ion exchange decontamination system, and the piping interconnections and flow control system between these and the refueling canal. In addition, since the problem cannot be defined with certainty, there should be provision for a backup system of increased capacity, if required. The latter can be installed on a temporary basis, using canal water as shielding.

EQUIPMENT MODIFICATION

As pointed out in the previous section, the existing canal water filtration system should be assessed for its applicability to this problem, namely removal of finely divided core debris instead of corrosion products. The solids load may be much higher than normal, with a substantial amount of very small particles (down to submicron), and the presence of fuel fines requires criticality control. The filter may become much more radioactive because of the presence of fuel, so filter shielding and provisions for transfer, packaging, and shipping must be examined. If the existing system is not appropriate it should be replaced or modified to meet the anticipated requirements without sacrificing any processing capacity.

The ion exchange system to remove soluble fission products from the water should be based on SDS technology, utilizing an appropriate mixture of inorganic zeolite ion exchangers, because demineralization with organic ion exchange resins is not effective with this water. If leach rates remain near the value presently estimated the capacity of the existing SOS equipment can probably be increased enough to maintain a satisfactory fission product concentration. This would require changing the piping and valves so that there are four parallel sets of two columns in series, rather than two sets of four, and increasing the throughput of each set to at least 10 gpm, preferably with capability for 15 gpm. This may also require increased pumping capacity. A maximum processing rate of 40 (60) gpm, taking acount of down time and column change out. There should also be a standby system of greater capacity for emergency use if fission product concentrations get too high. This might be an Epicor liner loaded with the same zeolite used in SOS.

Water management should follow the practice of flow from areas of low activity toward higher activity (Fig. 7). Thus, the cleanup system feed should be taken from the region where core removal work is being carried out or from just above the core; the vacuum, when in use, would be appropriate. These are the locations of the highest dissolved and suspended fission product concentrations. A high water filtration rate should be available, probably at least 300-400 gpm time average, with return to the canal at a location remote from the core. If a barrier is used, the flow path is from the canal into the barrier (via leakage or controlled), then to the core region, and finally back to the cleanup system.

A vacuum used for removal of gross debris and water from the region where core removal operations are going on must contain a filter system, which presents its own problems. The core debris should be collected in a container compatible with fuel shipping requirements. Depending on the effectiveness of the filters on the vacuum system, the discharge could be routed either to the canal filter or to the ion exchange cleanup system. The preferred mode would be to filter the water properly in the vacuum and send it to the ion exchange system, since this water should have come from the region with the highest fission product leach rate.

The ion exchange cleanup system (SOS and possibly a supplemental system) should take feed from the filter system effluent (and the vacuum system when it is operating); however, this processing rate is only a fraction (around 10%) of the filtration rate. The ion exchange system would discharge into the same return line as the canal filter. The existing pumps and pipes may not be adequate for the increased SOS throughput, and additional hardware would be required for the supplemental system in any case.

WASTE GENERATION

The performance of the filter system has not been examined. This subject will have to be evaluated when the suitability of the existing canal water filter system is determined. Criticality constraints, as well as water clarification requirements, may strongly influence the system design. Since recovered solids represent fuel debris rather than corrosion products (the problem normally dealt with in canal water filtration) the more stringent

constraints related to fuel handling and shipping apply. A substantial fraction of the core may be shipped as solids collected on filters (or transferred from filters), mostly from the vacuum system but partly from the canal water filter system, also.

The ion exchange system is better understood because of the experience with it in processing RCS and reactor building water. The individual SDS columns contain 60 gallons of zeolite, and they are capable of loading cesium and strontium from between 150,000 and 200,000 gallons of RCS water (possibly more if a greater breakthrough is accepted, which may be reasonable). On the basis of 150,000 to 200,000 gallons throughput and 30 gpm processing rate, one column would be exhausted every 3.5 to 4.5 days, or about 8 columns each month. The Curie is loading of each loaded column would increase directly with the concentration in the water being processed; at 0.02 μCi/ml ^{137}Cs it would be 12 to 15 Ci., along with whatever 90Sr is present in the water.

The auxiliary ion exchange system to provide increased processing capability if required, could be a shielded column such as Epicor, or a column located in the canal and using water for shielding. In either case it should be capable of handling 50 to 100 gpm of water to increase substantially the cleanup capacity above that of the upgraded SDS. Although Epicor columns may not be suitable, they can serve as an example of the loading that can be anticipated. If loaded with water at 0.02 μCi/ml to the same relative extent as the SDS columns (preceeding paragraph), the 4x4 Epicor (225 gallons) would contain about 50 Ci of ^{137}Cs, and the 6x6 (840 gallons) nearly 200 Ci. If loaded to capacity (to nearly complete breakthrough) the Curie content would roughly be doubled.

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IV. Reduction of Airborne Activity

A. Development of Pathway Concept

In the May 15 TAAG Report the use of a pathway concept to treat airborne particulate contamination was recommended. This included use of tunnels with appropriate air supplies and filtration. TAAG further considered the use of a pathway using appropriate contamination control enclosures (tents) for the reactor vessel head lift and fuel removal. A preliminary concept was developed by TAAG members from Newport News and Burns and Roe based upon experience in working on major radioactivity contaminated plant jobs. The concept is defined in attachment IV-A-1.

In summary this concept uses a temporary tent, located to provide the lowest dose rate path to the reactor head area, shielding as necessary, tent surfaces to cover transit surfaces that have high smearable contamination, air filtering to maintain low airborne particulate contamination in the tent and cooling to control temperature for personnel comfort.

The concept is intended to resolve the problem of heat stress and low worker productivity by attacking all causes of heat stress;

1). Eliminate the need for wet suits by covering contaminated surfaces

2). Eliminate the need for respirators by filtering the air in controlled volumes

3) Maintain tolerable temperatures.

In addition other ideas were included in the concept of Attachment IV-A-1 to alleviate heat stress and improve worker productivity are:

4) Use of the second personnel access hatch with a "once-thru" flow of people to speed entries

5) Use of the elevator to reduce the exertion of climbing from 305 to 347 elevation.

TAAG understands that GPU/Bechtel contemplates the use of some contained work areas later in the defueling. TAAG recommends that consideration be given to establishing all or part of the contamination control tent for the head lift. It could be enlarged later if found desirable.

TAAG's principal reason for recommending this approach to the heat stress/worker productivity problem is that it isolates the primary activity being conducted in the reactor building for a period of years. It provides protection from frequent interruptions of defueling that are otherwise likely to occur if the defueling is exposed to the general reactor building environment. Activities such as decontamination or reactor vessel head refurbishment, working in parallel with defueling are expected to create airborne activity periodically. Of particular concern from this standpoint are the condition in the 282' elevation. Substantial decontamination will be needed. These activities will be of the kinds likely to repeatedly produce increased airborne concentrations. Thus, this approach provides an avenue for continuity of the defueling activity.

B. Air Circulation and Filtration

In addition to the principal activity of head lift and defueling, many other activities will take place in the reactor building that would benefit from reduced airborne contamination and the desirability of working without respirators. Reduction of airborne contamination can be expected from increasing the flow of filtered air through the reactor building.

The reactor building purge system could provide 50,000 CFM continuously. However, it is not currently being operated continuously. One of the 2 trains of the system (25,000 CFM) are used now prior to each entry to purge the reactor building to reduce gaseous and particulate radioactivity and to assure an oxygen sufficient atmosphere for life support.

TAAG received data obtained from entries to date to determine whether continuous operation of the purge system would, in fact, reduce airborne particulate concentrations. The data is not conclusive, however some indications exist that reductions could be effected by continuous operation of the purge.

TAAG recommends that the reactor building purge system be operated continuoulsy to assist in reducing concentrations of airborne particulate activity and in treating any spikes of airborne activity that are released in the building. It is further recommended that 2 trains be operated whenever practicable to provide added filtration.

TAAG also notes that the supplementary air filtration system installed during the initial accident recovery remains on the auxiliary building. It had a capacity of 120,000 CFM, is equipped with HEPA filters and radioactivity monitors. It could be reducted to connect into the reactor building to provide additional purge air for reactor building ventilation if required for contamination control or establishment of capture velocities through radioactivity control barriers at the reactor building access batches.

D. Life Support Air Systems

As quick look and lead screw disconnect has progressed, the impedence of present respiratory equipment to personnel performance has been demonstrated. TAAG recommends that priority be given to installing and using systems for breathing and cooling air. There are systems that exist, (see Figure IV-D-1), that can be modified for TMI use in designing a Life Support Air System, which can be used for breathing and cooling. TAAG demonstrated to GPU/Bechtel, some devices which are available commercially (Air Supplied boods and vests) and can be used with this type of system. TAAG believes that a Life Support Air System can be installed economically to support most operations inside the containment, namely the work required for head lift. This system would have manifolds, at strategic locations, where individual air bases would be hooked up. To get personnel to and from work stations small individual air bottles can be utilized.

V. FOLLOW OF

A. Quick Look, Rod Motion Test and In-core Instrument Survey

In its initial deliberations regarding the defueling of TMI-2, TAAG recognized the broad spectrum of opinions which existed on the condition of the core. Planning for the defueling then required that the many possible conditions of the core be provided for. Rather than use that approach, it was the TAAG suggestion that three tests be performed in an attempt to learn more specifically the condition of the core.

These tests have now been performed and a clearer picture of the core condition is in hand. With this in mind TAAG has developed some suggested actions for future work using as a foundation stone the findings from the three tests.

This section of the report covers, in general terms, the findings from each of the three tests - then lists the conclusions from the tests and, finally, recommended steps for the future.

QUICK LOOK

The Quick Look inspection concept developed by TAAG for viewing inside of the TMI-2 Reactor Vessel proper was actually implemented in July and August 1982. See the TAAG report of March 1, 1982 for TAAG work in developing the concepts for the inspection and related tasks, such as primary and secondary water level control.

The first inspection took place in the center-most core position in 8-H on July 21, 1982. The second inspection on August 6, 1982 covered two core positions; The 8B position on the outer edge of the core and the 9E position which is midway between the edge and center of the core. The third inspection on August 12, 1982 included additional inspections into the 9E position with the TV camera using both the straight-on lens and the 90° lens to gain insight into conditions of the upper plenum's brazement assembly. Also during the third inspection both positions 9E and 8H were probed with a 1/2" diameter rod to determine the general condition of the core debris surface.

The technical director for the first two Quick Look inspections was a member of TAAG. The technical director for the third was from GPUN.

From these inspections the following key points and conclusions are noted:

1. A significant part of the top portion of the central core region appears to have been rubbleized. In this region, the top five feet of the fuel assembly are gone.

2. The debris bed over the remainder of the fuel assembly consists of loose rubble of various sizes and shapes. The initial checks of the debris indicate that at least the top 14" of debris is very loose. It appears to be mostly thermal shattered and oxidized material as opposed to melted fuel.

3. The upper plenum assembly appears relatively undamaged; there is no evidence of significant deterioration of the assembly. With regard to the upper plenum assembly the additional points are noted:

   a. The bottom ends of a few "C" tubes showed signs of localized melting; whereas other "C" tubes only a few centimeters away were undamaged.

   b. The top surface of the first (uppermost) support plate in the brazement assembly was relatively free of debris (i.e. appeared to have only been a light thin film of normal crud). The debris on this top surface should be indictive of the debris condition on the upper plenum assembly's top cover plate. Based on this evidence, the top cover plate should be relatively clean. Quick Look inspections further down the brazement assembly indicated that, starting with the third support plate, there is a layer of small flakes on the top surfaces. From the third support plate down, the thickness of the debris layer on the top of the support plates appeared to increase and got thicker the lower down in the brazement assembly. Portions of a non-fuel pellet are resting on top the 10th support plate (i.e. the lowest support plate).

c. The bottom surface of all the support plates in the brazement assembly were free of debris. Only a few gas bubbles appeared to be present on bottom surfaces. The fact that these bottom surface are clean is considered as hard evidence that the underside of the reactor vessel head should also be clean.

d. There is evidence that some fuel assembly upper end-fittings are jammed into the grid plate of the upper plenum assembly; therefore, there will be a need to push these end-fittings out (if fuel pins are still attached) before the plenum is removed.

4. The man-rem radiation exposure for the Quick Look inspection was considerably less than had been predicted when the concept was developed in January and February 1982. The total exposure up through the first Quick Look inspection was only 21.5 man-rem. The original predictions by Becthel were 404 to 1600 man-rem and by TAAG were 45 man-rem. From their experience it is concluded that efforts should be made to avoid radiation exposure extimates for work within the containment building being overstated.

5. The one problem encountered during the Quick Look Inspections which will impact on further in-containment work is heat stress. The heat stress problem must be solved for such operations as crane refurbishment, reactor head removal, etc, so people can work 4 to 6 hours without experiencing heat stress problems.

ROD MOTION TEST

The rod motion test recommended in the March 1, 1982, TAAG Report was completed in accordance with the procedures and Safety Evaluation provided. This test was intended to indicate:

1. If plenum distortion or melting had occurred to the extent that rod motion is interfered with or prevented.

2. The extent of damage to the control rods, i.e., the weight of the control rod remaining attached to the leadscrew.

The results of the test were provided in an EG & G data report. Of the 8 SPSR's, 4 moved significantly into the core and 4 others did not. The part of the test intended to measure the weight of the control rods could not be performed because the sensitivity of the load/input power characteristic of the rod drive motor to differentiate between normal running power and the weight of the remaining rod.

The significant result from the rod motion tests, which was available befor the start of the Quick Look, was the indication that leadscrews were free to move. Plenum damage was inferred to be slight since relatively small distortions of plenum parts would close tight clearances and bind the leadscrew. This information indicated a high probability that leadscrews could be uncoupled - an initial step in the Quick Look and of the defueling procedure. The same information supports the idea that damage above the bottom of the plenum is slight, thus enhancing the probability that the reactor head can be removed readily and that plenum removal will be near normal in-so-far as binding or sticking from distortion.

IN-CORE INSTRUMENT SURVEY

The in-core instrument measurements recommended in the March 1, 1982 TAAG Report were completed. EG & G also conducted some laboratory measurements to support the interpretation of data taken in TMI-2.. This work is reported in a draft report "Preliminary Report of TMI-2 In-Core Instrument Damage" by N. Wilde and M. Yancey dated July 14, 1982.

In summary, the measurements confirmed and extended the earlier measurements of the lead-to-lead resistance and the insulation resistance of the instruments. Some thermocouples (TC's) were found to have open junctions, others had metal-to-metal junctions. The open TC's indicated electrical charging characteristics of a wetted TC. Lead-to-lead resistance of all TC's with metal junctions indicated the present length of the TC's as much as 20 feet shorther than the original length. The shortened length of the TC's indicates damage at the bottom of the core and as low as the bottom of the reactor vessel. Also as indicated later, this affects interpretation of core temperature measurements being reported.

The SPND's were similarly found to be variously open and wetted, shorted between lead and sheath or intact. As indicated by prior data, none of the 52 instrument assemblies had all 8-SPND's intact and about 75% of the SPND's at the lowest level in the core (about 8-inches above the bottom of active full) were failed. It is significant that the measurements taken in June 1982, confirmed those taken in April 1979 thus proving that the lower level SPND's in an instrument assembly are not progressively failing from corrosion of their sheaths. Rather the damage of the SPND's occurred at the time of the accident. The significance of the SPND damage distribution is that it shows general damage down to the bottom of the core. The only general area of SPND survival is at the core periphery up to level 2 (about 22 inches above the bottom of active fuel).

Although no additional observatons were obtained since the earlier TAAG report, it is also noted that the radiation dose rates, contamination and boron leakage residuals associated with the in-core instrument external guide tubes and service area confirm the indicated damage to instruments. Leakage of the instrument assembly sheaths and individual instrument sheaths was necessary to cause these conditions.

SUMMARY AND CONCLUSIONS

1. The aggregate of observations from the Quick Look, Rod Motion Test and Instrument Survey provide a substantial view of the extent of damage in the reactor and provide basis for progressing with defueling actions. As noted from each of the individual observations and measurements, the damage extends generally over the entire cross-section of the core. Only the peripheral fuel assemblies can be expected to remain approximately full length. A small proportion of the peripherial fuel assemblies yielded information through instrument measurements. Even the furthest, peripheral fuel assemblies had instrument damage.

2. The condition of the plenum indicates that reactor vessel head removal can proceed in a near normal procedure. The upper horizontal surfaces were relatively clean and should indicate the head is relatively clean. No distortion was observed in the upper plenum, indicating that there may be little distortions existing in the head.

3. The observations of the lower parts of the plenum showed slight damage but it appeared there were no distortions of the extent that would impede plenum removal. Thus removal of the plenum proximately following head removal is likely to be feasible without delays for special tooling or extensive development of plenum removal procedures.

4. The remaining major unknown about the extent of core damage concerns the lower part of the core. The most practical method of examination of this area, above and below the core basket, is to inspect and sample after the plenum is removed. This gives access both to the whole cross-section of the core and to the bottom of the reactor vessel through the annulus.

5. As indicated both by visual determination of cavity size and debris on horizontal plenum surfaces and by probe determination of consistency and depth of the loose rubble, a substantial fraction of the core is removeable by vacuuming. The loose rubble appears clearly vacuumable. In addition, the material lost from the cavity, which may be found within the original core boundaries or in the bottom of the reactor vessel is also expected to be vacuumable. Thus, the vacuumable fraction is in the range of 40 percent of the core.

RECOMMENDATIONS

1. Short term actions should be taken to continue progress toward defueling in an efficient and safe manner. These include:

   a. Start reprocessing of RCS thru SDS to lower CS level to as low as pratical.

   b. Disconnect leadscrews at positions 12N, 7K and 2F.

   c. Energize ASPR's and parts lead screws for head lift. At this point we can disconnect the CROM electrical and cooling water piping to the head.

   d. From the Quick Look the upper five feet of the fuel assembly have been rubbelized and the in-core thermocouples have been destroyed and their readings are not useful. It is recommended that thermocouples be installed through lead screw cavities to monitor core region water temperatures.

   e. Obtain water sample at the surface of Water (fabric on end of rod).

f. Lower RCS water (experiment)

1) Take radiation reading at predetermined locations on the head prior to and after lowering water level.

2) Take RCS water temperature with water level lowered.

3) Make sure SG water level is lowered; upon completion of above raise water level back to present level.

g. It has not yet been shown that the core debris does not contain potentially pyrophoric material. In addition, the presence of core debris on the upper steam generator support plate has not been ruled out. It is, therefore, recommended that the drained volume of the steam generator be kept under a blanket of nitrogen until it has been shown that pyrophoric material is not present in the core debris or that core debris is not present in the upper volume of the steam generator. Further, the blanket will inhibit corrosion of the steam generator tubing and surfaces.

Past studies have shown that the core materials of concern will not react exothermally if covered with water or an inert atmosphere. Therefore, it is also recommended that until the presence of hyrophoric materials is ruled out, GPU should carefully review the draining of any systems or components to assure a potential problem is not created.

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4 Removal of the reactor vessel head should continue to proceed at the earliest feasible time.

5 Preparations should start now to enable the removal of the plenum at the earliest time feasible after head removal.

6 Development of the equipment to vacuum the loose rubble from the plenum and from the reactor vessel should start now with the objective of availability of the vacuum equipment at the time of plenum removal. The water vacuuming equipment will be intimately interfaced with the RCS and canal water clean-up systems and the fuel shipping cannisters and require concurrent development.

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see N

B. Early Head Removal

TAAG has been studying the procedures, prerequisites, and other requirements for early removal of the TMI-2 reactor vessel closure head during its current meeting period. Attached are the most recent recommendations, broken into the following categories:

1. Prerequisites for the Reactor Vessel Closure Head Removal

2. Polar Crane Reactivation

3. Containment for Reactor Vessel Closure Head Removal

4. Reactor Vessel Closure Head Removal Method

Attachments follow the above sections.

I. Prerequisites for Reactor Vessel Closure Head Removal

In considering procedures for the reactor vessel closure head removal, TAAG has reviewed and considered the techniques used by various organizaions for similar tasks. This investigation has been conducted to ensure that related problems have been addrssed, and that the list of pre-requisites recommended for GPU use is comprehensive and complete. (See section I of this report for the recommendations.) Enclosed in this section of the report as an attachment is a sample list of prerequisites, the format for which is suggested for GPU use.

Additional informaion concerning testing prior to closure head removal is included in the "Reactor Vessel Closure Head Removal Method" section of this report.

II. Polar Crane Reactivation

Methods have been investigated for refurbishing the TMI-2 500-ton Polar crane electrically and mechanically during the current report period. Emphasis has been placed on making the restoration as quick and practical as possible, with minimizing the possibility of malfunction as the major goal. Recommendations for procedures leading to reactivating the crane are contained in Attachment B, which is a document that was submitted to GPU during the meeting of July 19-21, 1982.

TAAG has also investigated problems that have occurred during various reactor head installation or removal operations in order that specific areas of concern may be identified and addressed and problems avoided. The following are recommendations based on the experiences of several reactor servicing organizations:

a. Electronic Load Indicators (Electric Load Cells with Indicators).

Shipyards have experienced problems with electronic load indicators which produce erratic readings due to radio frequency interference in areas of operation. Shipyard experience has shown that these problems are minimized by filtering the power supply and input signals with systems of chokes and capacitors. In addition, load indicating equipment should be calibrated by qualified personnel, and only qualified personnel should operate this equipment during head removal.

b. Rigging. TAAG recommends that the rigging be trial fitted to the crane hook prior to use to ensure that all points fit and operate as designed. This rigging should be tested using a test load which is one and one half times the weight of the reactor vessel head, or is at least as great as the crane's static test load. If practical, this testing may be conducted in conjunction with the static load testing of the crane using the missile shield hold-down bolts. Following the load test and preceeding the reactor vessel closure head lift, the rigging should be subjected to, at minimum, a thorough visual inspection to ensure that it is in good operating condition.

c. Safety Devices and Communication Systems. Safety devices to be used during the head lift should be checked to see that they are operational immediately before the reactor vessel head is lifted. Among the safety devices included are limit switches, alarms, communication systems, etc. A backup communication system is essential for safe crane operation and should be designated and checked before beginning the reactor vessel closure head lift.

A literature search conducted at HEDL revealed a number of cases of crane incidents in nuclear plants. A categorized listing of these is shown in Attachment C for GPU use as reminders of the experiences of other organizations.

III. Containment for Reactor Vessel Closure Head Removal

TAAG recommends that a temporary containment tent be used inside the containment building of TMI-2 during the reactor vessel closure head and fuel removal. The purpose of using this type of tent is two-fold: It will allow personnel access to the reactor vessel closure head and fuel canal without the need of full-time personal respiratory protection, and it will allow separate operations within the containment building to take place simultaneously. A description of the recommended containment system is described in Section IV of this report.

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IV. Head Removal Method

TAAG has reviewed the presentations made by Bechtel, and all other evidence received to date, and recommends that the reactor vessel closure head be removed by the conventional dry removal technique. This method has been chosen over the "wet" removal technique which would require that either the containment around the closure head or the fuel canal be flooded to a given depth. TAAG also recommends that, subject to the findings in the test recommended in this section and in section V-A, the pre-head lift inspection be deleted, this decision is based on the evidence indicated in V, A of this report.

The decision to recommend the dry head removal technique suggested by Bechtel is based upon the following:

1. All information received to date shows that the plenum has sustained no damage, and thus can be removed by a conventional method.

2. No tests, including ASPR test and quick looks, have given any indication that dry head removal will not be successful.

3. TAAG believes that airborne contamination problems believed to be associated with the dry removal method can be addressed and resolved through the use of special work methods designed to control such contamination problems. The use of poly and herculite boots and shielding should be sufficient to control airborne contamination and radiation.

Prior to the removal of the reactor vessel closure head, TAAG recommends that the following tests be conducted to determine whether or not high levels of radiation and airborne contamination should be given serious attention:

1. TAAG recommends that test be run to determine whether or not large amounts of debris will adhere to the underside of the closure head. Specifically, TAAG recommends that

   a. A sample of the surface water in the reactor be taken by inserting a rod with absorbent paper attached into existing open motor tube locations. (Also recommended in section V-A)

   b. A sample of the lead screw material be taken from each of the lead screws.

   c. Radiation levels be measured at pre-determined locations on the closure head, before and after the vessel water level is lowered. (Also recommended in section V-A)

2. TAAG recommends that a test be conducted to determine whether the expected temperature of water in the reactor vessel will cause or increase airborne contamination. Specifically, a sample of the water in the reactor vessel should be taken using the absorbent paper technique mentioned above. The water and paper shoud be heated to the expected vessel temperature (130-135°F), and a radiations measurement should then be taken approximately 6" from the paper.

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TAAG further recommends that the following task be performed as soon as possible to reduce problems when the closure head removal takes place: A penetrating oil, such as oil of wintergreen, should be put on all closure head studs and retaining nuts to reduce the difficulty of removing them.

ATTACHMENTS

ATTACHMENT A: A detailed checklist of prerequisites required for reactor vessel closure head removal.

ATTACHMENT B: Newport News Shipbuilding file of recommendations for procedures leading to the reactivation of the Polar Crane.

ATTACHMENT A

THREE MILE ISLAND

PREREQUISITE LIST FOR

REMOVAL OF REACTOR VESSEL CLOSURE HEAD

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The prerequisite items listed on the attached list must be completed prior to proceeding with removal of the closure head.

CONCURRENCE

_____ DATE _____

_____ DATE _____

_____ DATE _____

_____ DATE _____

_____ DATE _____

PREREQUISITE FOR REMOVAL OF REACTOR VESSEL CLOSURE HEAD TMI

1. PROCEDURE CHANGES REQUIRED

	PROCEDURE	DESCRIPTION	SIGNATURE/DATE
A.	-------	Completing this prerequisite list will certify at that time, that all procedures covering operations until the next prerequisite point are issued and no changes are outstanding.	

The person designated to sign for an action verifies, based on personal observation, certified records, or direct report from watchstanders, and certified by his signature that the action has been performed in accordance with the specified requirements.

PREREQUISITE FOR REMOVAL OF REACTOR VESSEL CLOSURE HEAD TMI

2. DISCREPANCY REPORTS

	PROCEDURE	DESCRIPTION	SIGNATURE/DATE
A.	-------	All listed discrepancies must be cleared the clearing report number indicated, if applicable. This list contains all discrepancies are required to be cleared prior to proceeding with this list's prerequisite point.	

The person designated to sign for an action verifies, based on personal observation, certified records, or direct report from watchstanders, and certifiec by his signature that the action has been performed in accordance with the specified requirements.

REMOVAL OF REACTOR VESSEL CLOSURE HEAD TMI

	DESCRIPTION	SIGNATURE/DATE	
3.	Reactor vessel water level is at ____above the reactor vessel nozzel centerline and will be maintained IAW _____.		
4.			
5.	Pre-shift briefings have been completed.		
6.	All lifting and/or handling equipment for this operation has been tested and is certified ready for use.		
7.	Sufficient refueling personnel are qualified and trained to perform the special refueling plant evolutions applicable to this prereq-uisite list.		
8.	Communication system between crane operator and reactor vessel work area has been checked out and one backup system is available.		
9.	All procedures are up-to-date and ready for use.		
10.	Safety Analysis Report prepared and approved.		
11.	Negative ventilation established in contain-ment and work area.		

The person designated to sign for an action verifies, based on personal observation, certified records, or direct report from watchstanders, and certified by his signature that the action has been performed in accord-ance with the specified requirements.

REMOVAL OF REACTOR VESSEL CLOSURE HEAD TMI

	DESCRIPTION	SIGNATURE/DATE	
12.	Heavy lift crane has been serviced, is operational, checked out and certified for use.		
13.	Gamma monitor available, checked out and installed in work area.		
14.	Hydrogen monitor available, checked out and installed in work area.		
15.	System for raising H_2O level in reactor vessel installed and checked out.		
16.	Accordian sleeve for removing closure head fabricated, inspected and ready for use.		
17.	Continuous air monitor system installed in local work area and checked out.		
18.	Temporary head support stand available, inspected and ready for use.		
19.	Equipment and personnel access routes including emergency exits have been established and all personnel are familiar with these routes.		
20.	Leadscrew disconnection complete and verified using minature camera.		
21.	If required, mini-tank has been fabricated, inspected, installed and leak checked.		

The person designated to sign for an action verifies, based on personal observation, certified records, or direct report from watchstanders, and certified by his signature that the action has been performed in accordance with the specified requirements.

REMOVAL OF REACTOR VESSEL CLOSURE HEAD TMI

	DESCRIPTION	SIGNATURE/DATE	
22.	Closure bolts have been removed from the vessel flange and suspended from the head.		
23.	Reactor vessel temporary cover with sampling connections available.		
24.	All verification sign-offs up to this pre-requisite list are signed off and complete.		
25.	Appropriate retrieval tools including shears for severing herculite sleeve are available and ready for use.		
26.	Fire protection equipment available.		
27.	Water exclusion area established and will be maintained.		
28.	Back-up power available for crane, lights and instruments.		
29.	Containment equipment hatch has been secured and airtested.		
30.	Expected radiation levels under head have been determined by measuring through head and recorded.		
31.	Boron monitor system has been installed, checked out and is operational.		
32.	The weather has been checked and suitable conditions are forecast for the next 48 hours during closure head removal.		

The person designated to sign for an action verifies, based on personal observation, certified records, or direct report from watchstanders, and certified by his signature that the action has been performed in accordance with the specified requirements.

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All outstanding prerequisite items have been completed and no actions have been taken which would invalidate any verification signature on the prerequisite list.

_____
Signature - Date

JRG AGREEMENT NO. _____

The concurs in commencing operations

JRG CONCURRENCE

_____ DATE _____

_____ DATE _____

_____ DATE _____

_____ DATE _____

_____ DATE _____

The Chief Engineer directs proceedings past the prerequisite point.

_____
Signature - Date

The person designated to sign for an action verifies, based on personal observation, certified records, or direct report from watchstanders, and certifies by his signature that the action has been performed in accordance with the specified requirements.

C. Follow of Control of Radiation Exposure

In the May 15, 1982 TAAG Report, Section V discussed this subject, including reduction of gamma dose rates, reduction of dose rates from concrete surfaces and reduction of airborne activity. TAAG continued to follow each of these areas, however additional emphasis was placed in Mr. R. Arnold's instruction letter on the airborne activity. That part of this subject is, therefore, addressed separately in Section IV of this report. The work discussed in this section treats the occupational exposures of persons working in the TMI-2 reactor building.

Reduction of Dose Rates from Concrete Surfaces

Recommendations were made by TAAG in the May 15 report. They were based upon the preliminary results of the gross decontamination experiment. Subsequently Bechtel issued a final draft of this report. The final report utilized data obtained by EG&G on borings of reactor building surfaces and gamma spectrographic measurements of specific surface activity made by SAI. A major part of the decontamination experiment was intended for reduction of loose surface contamination, not for gamma dose rate reduction. As regard dose rate reduction, the final draft report states "The area dose rates were not reduced 70 to 150 mR/hr in the test areas in the reactor building as was planned".

TAAG continues to believe that the difficulty in reducing area dose rates is related to the difficulty of decontaminating uncoated concrete or concrete with degraded coatings. Further attempts at decontamination of surfaces in TMI-2, now underway, may resolve this question for the elevation 305' and 347' floors and interior walls. Ultimately, should relatively mild surface cleaning not remove enough activity to reduce dose rates, more severe material removal or shielding may need to be used.

TAAG is particularly concerned about the potential for radionuclide intrusion into the concrete in the 282' elevation. Dose rates below the 305' elevation floor are high, ranging from 60 R/hr near the 282' floor to 3 R/hr under the 305' floor. The sources of these dose rates are not specifically known. They may be largely from the sump water and solids debris on the floor.

However, review of literature and available experience on intrusion of radionuclides into concrete suggests such intrusion could have occurred and present a major source of these dose rates. In particular this applies to the concrete submerged for almost 3 years. It should be noted that the 282' elevation floor is a 2 ft. 6 in. thick concrete fill mat poured on top of the reactor building liner. It is caulked around the periphery, at the 'D' ring walls and at other cut-outs with a cork composition. This caulking would not be expected to be watertight for 3 years. Therefore, the fill mat would probably have been wetted on both sides and at numerous edges.

TAAG suggests that the possibility of radionuclide intrusion into the concrete in the 282' elevation be considered in the preparations for decontaminating this area of the reactor building.

Man-Rem Exposure Estimates

In preparation for the Quick Look exposure estimates of various alternative methods were made. Based upon the best dose rate information available at the time (about February 1982) the man-rem estimate by GPU/Bechtel was about 435 man-Rem. Subsequent to the Quick Look an initial summation of all the personnel exposures for this activity was reported to be less than 30 man-Rem.

The ALARA exposure estimates for the clean-up are an important aspect of the licensing process. The current occupational exposure estimate for the whole clean-up stated in the NRC Programmatic Environmental Impact Statement (PEIS) is 2000 to 8000 man-Rem. The initial report of the Quick Look exposures is encouraging relative to the estimate for that activity. However difficulty was observed in decontaminating to achieve target dose rates (TAAG recommended 20 to 50 mR/hr). This could have substantial effect on actual occupational exposures. Early identification of the differences from the PEIS estimate would be desirable.

TAAG recommends that the occupational exposure for work already performed in the reactor building be evaluated to determine whether the PEIS estimate is applicable.