Memo No. 1 H-6a.

TASK CLOSE OUT DOCUMENT

Task Scope ASSESSMENT OF CORE INTEGRITY

To: M. Levenson
S. Levy
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Task No. 6 Date Complete 4/13/79

Reason felt task is complete:

The assessment of immediately available information has been completed. Some additional analysis of the core heat-up transient during TMI-II accident is recommended.

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THREE MILE ISLAND

CORE INTEGRITY ASSESSMENT

INDUSTRY ADVISORY GROUP

Agreed to and understood by:

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SUMMARY CONCLUSIONS

The bottom 1/4 - 1/3 of the core is likely to have retained its basic physical integrity; e.g., rod dimensions spacing, etc., but probably has considerable clad surface oxidation.

The top 2/3 - 3/4 of the core volume is most likely to be in a heterogeneous state with a peripheral annulus of relatively intact (but highly oxidized, ballooned and perforated clad) fuel rods with a central core region which has experienced very substantial damage; e.g., substantial clad ballooning, clad perforation rod bowing, excessive clad oxidation (15 - 45% total core clad oxidation with locally up to 100%), and significant clad and oxide fragmentation.

In spite of a high degree of fuel damage in the upper central core region it is likely that the general core structure retains some physical integrity; i.e., control guide tubes which support the grids largely intact, remaining partial segments of distorted and oxidized cladding remaining intact, very possibly most of the grid cells are still occupied by cladding and the upper ends of the heavily damaged rods are probably hanging from the top grid spacer.

The highly damaged top central core region most likely has considerable debris (clad, clad oxide and fuel pellet fragments) pile-up at the grid spacers.

The resulting effective core flow area blockage due to the fuel condition based on flow resistance claculations, is substantial and may be as large as 80 to 90%; with local fuel assembly blockage in excess of 90%.

The degree of flow blockage inferred to exist, while substantial, should not preclude adequate core cooling under natural circulation conditions. However, local coolant boiling in the most highly blocked areas can be expected. We believe that the core is stable enough that it will retain coolability during natural circulation operation.

The entire core and especially the guide tubes which are still intact are suspected to have received enough oxidation so that oxygen incursion into the zircoloy caused the remaining clad to be extremely brittle. Hence, the fuel assemblies may not retain physical integrity during removal from the core, if removal is done in the usual manner.

INTRODUCTION

The above assessment of the Three Mile Island core integrity is based upon a combination of quantitative and qualitative observations and assumptions derived from both the deduced accident performance and subsequent steady-state performance in single loop-one reactor coolant pump operation. The evidence is largely circumstantial and thus these conclusions rely heavily on deduction from numerous indirect observations.

CORE COOLING DURING THE ACCIDENT

At least two different scenarios for core cooling during the accident have been put forth. These two scenarios are presented in:

a. The TMI Core Assessment by N. Kaufman of EG&G of 4/6/69, and b. the TMI Core Condition Assessment by D. N. Roy of B&W of 4/3/79. While these reports differ on the specific details of core cooling during the accident, they are in agreement that the core was partially or totally uncovered on at least two, and possibly three, occasions for periods of several minutes to several hours. Both also agree that the first period of significant uncovery occurred at approximately 5:45 A.M. on March 28, 1979 (approximately 100 minutes following the initiating transient). This initial uncovery apparently lasted for approximately one hour and was terminated when the pressurizer block valve was closed and pressurizer liquid flowed into the core quenching the fuel rods. Beyond this initial uncovery and recovery there were apparently two subsequent periods where partial uncovery is believed to have occurred.

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CORE DAMAGE DURING THE ACCIDENT

Because of the significant reduction in heat transfer coefficient attendant to core uncovery, reductions of two-to-three orders of magnitude in convecture heat transfer coefficient, the fuel rods undoubtedly experienced very significant overheating and damage during the initial uncovery period. The sequence of events during this core damage period is likely to have been manifested by initial clad ballooning, then perforation. Clad oxidation rate would have increased significantly as the clad temperature increased. The conclusion that significant fuel perforation occurred during the initial uncovery is supported by the steep increase in fission product activity in the reactor building the period from 5:45 to 6:45 A.M. on March 28, 1979. While the subsequent two periods of uncovery are likely to have resulted in additional fuel damage, the most major change of physical condition of the fuel is likely to have occurred during the initial period of uncovery when the fuel was initially unfailed. Attachment #1 letter P. W. Marriott to D. C. Ditmore "Mechanical Condition of Three Mile Island Core," 4/13/79, presents the results of a rough evaluation of the possible core damage scenario.

It is estimated that the peak cladding temperature increased from less than 1000°F to the range of 2000 to 2600°F in the first 10 minutes of the initial uncovery period. It is estimated that most fuel rods experienced cladding temperatures in this range within the next thirty (30) to fifty (50) minutes. The exception may have been that fuel clad in lower power bundles may have been below 2000°F; i.e., peripheral core bundles may be less damaged. The fuel cladding would be expected to experience perforation in the range of 1400 to 1500°F for the internal pressurization of the TMI-II fuel rods and the measured reactor system pressure during this period. Prior to perforation individual fuel rods would have experienced considerable clad ballooning possibly up to 2X the initial diameter. Clad ballooning and perforation is not likely to have been co-planar across the core although preferential ballooning within the hottest core regions axially would be expected. Ballooning is typically somewhat random due to statistical variations in the cladding. It was suggested that co-planner ballooning might have been possible considering the postulated slow rate of initial uncovery. However, experience and a simple calculation indicate that clad overheating would not be expected until the fluid level dropped well below the midplane, thus greatly reducing the likelihood of localized co-planarity. The zirc cladding is believed to have experienced significant oxidation during the rise in temperature with the highest rates experienced above 1800°F clad temperature. This postulation is consistent with calculations of the amount of evolved hydrogen which imply oxidation of from 15 to 45% of the fuel cladding (see attachment #1, "Assessment of Cooled Hydrogen," by J. W. Thiesing, 4/12/79). It is likely that local cladding oxidation of essentially 100% of the wall could have occurred at perforation locations during the first uncovery, and possibly at other locations in the subsequent periods of uncovery.

After approximately one (1) hour during the initial uncovery period. The core was recovered with liquid. This occurrence is likely to have quenched the hot clad and clad-oxide causing high thermal stresses and resulting in significant fragmentation of the most highly oxidized cladding.

The two subsequent uncovery and recovery incidents would be expected to have oxidized additional cladding and resulted in additional fragmentation on quenching.

This fuel rod and cladding damage sequence is likely to have led to a condition of substantial buildup of zirc clad, clad oxide and fuel pellet fragments (debris) at the grid spacers, which would present a very significant flow blockage, particularly when combined with the local effects of rod ballooning and bowing. The greatest damage would most likely exist in the hotter central (radially) region of the core, and the upper portions of the core which were likely uncovered for the longest periods.

Experience with core heat-up analysis during a loss of coolant accident suggests that the other oore components; i.e., control rod guide tubes, will have experienced temperatures comparable in magnitude to the peak fuel clad, and thus those zirconium components will be highly oxidized and brittle.

OBSERVATIONS SUBSEQUENT TO THE ACCIDENT

A number of observations of plant performance during single loop -- one reactor coolant pump operation provides additional circumstantial evidence which lends some indirect support to this core damage scenario.

Self Powered Neutron Detector (SPND Damage)

With the exception of one detector at level 5 (there are seven levels of SPND's starting in the lower core region) all detectors above the second level are not working. The reasons for their malfunction is unknown. However, since essentially all SPND's above the lower 25% of core volume are not functioning, it is likely that this portion of the core (the top 2/3 - 3/4 of the core volume) reached higher temperatures and thus experienced more damage than the lower 1/4 - 2/3 of the core volume. This may be a consequence of lower temperatures during the accident due to shorter periods of uncovery and lower heat generation rates.

In-Core Thermocouples

Fifty-one (51) of the original fifty-two (52) core exit thermocouples apparently survived the accident, of these, forty-nine (49) remain intact and are now recording temperatures at the core exit which appear to be accurate. This fact is significant because the thermocouple leads pass up through the core region in the center thimble of 52 of the fuel assemblies (locations shown in Attachment #3) to a detector (junction) approximately 4 - 6" above the core exit. These thermocouples travel up the same tube as the SPND's. The thermocouple wire melts at 2500°F. The continued availability of these thermocouples suggests that the center thimbles and other non-fueled tubes; i.e., control rod guide tubes, have remained intact so that the core has retained some structural integrity in the upper regions in spite of an expected high degree of fuel damage; i.e., distortion, clad perforation and oxidation and fragmentation.

Traveling Incore Probe (TIP) Insertion

The insertion of the single core TIP was attempted. The probe stuck at a level of 3.5 feet into the core from the bottom. No further travel was possibly and the probe remains stuck. This is additional qualitative evidence of the degree of core damage being less in the bottom 1/4 to 1/3 of the core volume. It indicates that this tube, while possibly highly distorted, retains enough integrity to prevent further movement of the TIP.

Core Flow Resistance/Flow Blockage

B&W and PNL have performed independent assessments of core flow resistance which lead to a conclusion of significant flow area blockage. B&W has estimated the increased core flow resistance could be as high as approximately a factor of 60X. This estimate was apparently confirmed in two ways; 1) comparison of reactor coolant system flow meter readings, with one pump operation, before and after the accident, and 2) a post-accident single-loop operation core heat balance using the estimated decay heat and measured core coolant temperature change to determine flow. These assessments provided a range of core resistances up to a factor of 60X increase over the pre-accident valve. A factor of 60X increase in core flow resistance is approximately equivalent to an effective core flow area blockage of ~90% $\left\{\frac{(k/a^2)_2}{(k/a^2)_1} \sim 60;\ \frac{A_2}{A_1} \sim \sqrt{1/60};\right.$ $(1-\sqrt{1/60})100 \sim 87\%$.

PNL has employed the COBRA code to perform analyses of a 26 channel simulation of the TMI-II core to reproduce the measured core exit temperature distribution during single loop operation. From their work they estimate 60 - 80% gross core blockage, with local regions with blockage in excess of 95%. The local high blockage is implied by the wide variation in core exit

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coolant temperatures during single loop operation (attachment #3). With a variation of $\sim 100^{o}F$ from the peripheral core regions to the central core region.

A simple core heat balance using the average core exit thermocouple readings on the periphery of the core also leads to the conclusion of an effective core flow area blockage of ~90%. Specifically, the measure core coolant Δt_{core} in the core peripheral region was $\sim 8.5^{o}F$ (see attachment #3) on the average on 4/10/79 (this is typical of performance of the past week or so), with single loop - one pump operation. Without flow blockage the core coolant Δt_{core} should be $\sim 1^{o}F$ in single loop - one pump operation. This is confirmed by simple calculations (see attachment #4), and by measurement, since the measured cold leg-to-hot leg Δt during this condition is $\sim 1^{o}F$. The ratio of unblocked Δt_{core} to blocked Δt_{core} is reflective of the ratio of blocked to unblocked core coolant flow; i.e., $\frac{W\ unblocked}{W\ blocked} \sim \frac{1}{8.5} \sim .11$ or a flow area blockage of ~89%.

The trend of increasing temperature from the outer periphery to the central core region suggests again the possibility that the central core region is the most highly damaged. Further, the peripheral exit temperatures are close enough to the core inlet temperature that they suggest that the minimum flow blockage occurs in the peripheral region. It should be noted that the one exception to this central to peripheral trend is a core temperature reading in the center core region which is much below all temperatures, including the inlet. Another observation of significance relative to flow blockage resistance is that transfer of pumping from the S/G A1 to A 2 pump shifted the location of the highest exit coolant temperature to another quadrant in the central core region. This suggests variation in inlet lower plenum flow and pressure distribution attendant to single pump operation. This is consistent with analyses in the Oconee reactor performed by T. Mott of Technology for Energy Corporation, in Knoxville, Tennessee.

Letdown Flow Blockage

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The letdown line has given some evidence of flow blockage, increased flow resistance, following the accident. This flow blockage could result from carryover of some debris from the core region. Similar flow resistance has been observed, according to L. Noll TMI-I shift supervisor, when system crud has been shocked loose in TMI-I. As a consequence the observation is not conclusive, but is qualitatively consistent with fuel fragmentation. If the filter became clogged over an eight hour period, after the accident, it could be from crud. If it became clogged in a ten minute period, it is probably from large particles (such as a fuel). The actual occurrence should be checked. Also the ability of the suspected flow rates in the core to levitate fuel particles should be checked.

NATURAL CIRCULATION CORE COOLING ASSESSMENT

The previous considerations indicate the core configuration is stable. The calculations presented here show that natural circulation can cool the core in its present supposed state.

A number of natural circulation analyses have been performed with varying degrees of flow blockage from 0% to > 90% blockage by B&W and INEL. Typical results obtained are presented in Table 1.

These results indicate that adequate core cooling can be achieved with substantial core blockage; i.e., up to 95% without gross core boiling. Since the core condition in TMI-II is estimated to be no worse than ~ 90% effective core flow blockage, a stable condition with no bulk core boiling is expected. Some local boiling in the most restricted flow areas would be expected. However, the intended operation with high subcooling will preclude any bulk boiling.

RECOMMENDATIONS FOR FUTURE WORK

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The results from calculations being performed at various locations will be very important in verifying the reactor core model of the core presented here. For example, Carl Obenchain of INEL calculated the guide tube temperatures and showed that these temperatures closely followed the fuel cladding temperatures. This indicates that the guide tubes are probably brittle in regions where the cladding is completely destroyed. Further calculations could give a better assessment of the amount of core damage based on a better assessment of the core/fuel temperature time history.

TABLE I

NATURAL CIRCULATION CALCULATIONS

Source	Steam Generator Mode	Decay Heat	Core Flow Blockage	Core Flow lb/sec	Core ΔT of	Primary System Pressure
B&W	S/G's A&B Steaming	5MW	~87%	~330	10°F	?
INEL **	S/G A Steaming, S/G B isolated	"	85%	320	14°F	1000 psi
"	" "	"	95%	100	47°F	"
"	" "	"	99%	14	↑*	"
"	S/G A&B Solid	"	0-40%	429	~11°F	"
			0-40%	550	~10°F	"

*Had not achieved thermal equilibrium after 1500 sec. Expected to result in a steady state boiling condition in the core.

**Performed by Stewart Fiscler at INEL.

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