UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 18, 1979

(1) Copies to Kevin, Budnitz, George Lahr, George F.
(2) Return to me.

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

FROM: Al Kenneke, Acting Director, OPE

SUBJECT: COMPARISON OF NUREG-0600 AND NSAC-1

Per requests from Commissioners Gilinsky and Bradford the following is a preliminary comparison of the major differences between the Nuclear Safety Analysis Center's (NSAC) "Analysis of Three Mile Island Unit 2 Accident" and the USNRC (I&E) NUREG-0600. Although we have identified some specific differences, the reports tend to agree, especially with regard to the factual events of the accident. Both reports, in our opinion, are good ones which complement each other in presenting an understanding of what transpired during the first 16 hours of the accident. The NSAC and I&E reports are both considered preliminary in that they are not, nor were they intended to be, in-depth engineering evaluation of the causes, course, and consequences of the event. These types of analyses will be covered in future NSAC reports.

First, these two reports differ somewhat in both the scope and information base from which they draw. The NRC report is a more critical one which evaluates the performance of Met Ed personnel during the accident and preliminarily identifies areas of non-conformance with NRC regulations. Further, this NRC report utilizes, in addition to plant data, information obtained via direct interviews of the principals in presenting an accounting of events, both operational and radiological, and in analyzing personnel performance. The NSAC report appears to be considerably less critical of personnel actions but does make inferences in that direction. NSAC informational sources included plant records, computer data, logs, procedures and copies of GPU-conducted interviews of plant personnel.

Specifically, the two reports differ in the following:

- NSAC states (page TH-25) that the unavailability of auxiliary feedwater for almost 8 minutes at the beginning of the accident "... was not a significant direct contributor to the damage of the core ...." The report goes on to state (page TH-32) that even if auxiliary feedwater flow had been established within 30 seconds, as designed for, core damage would still have occurred, "although a few minutes later than in

CONTACT:
Bill Travers (OPE)
634-3302

8002100099

the actual case." By drawing this conclusion the NSAC group places heavy emphasis on the roles of the valve on the pressurizer (referred to variously as the electromatic power operated valve (EMOV), power operated relief valve (PORV), and electromatic relief valve (ERV)), and the high-pressure injection system (HPI) in this accident. It should be noted that an analysis to support this judgment was not included. The NUREG (I&E) report makes no conclusion as to the ultimate result of losing auxiliary feedwater during the accident. The question of whether the unavailability of auxiliary feedwater was a major contributor to core damage is, in our opinion, still open and deserves further analysis. There is, however, no question that the loss of feedwater was the triggering event for the accident.

- While both reports list the stuck open EMOV as an important aspect of the accident, the NSAC study states (page 4) that the accident might have been avoided within 100 minutes if operators had corrected the effect of this stuck valve by closing a block valve upstream of the EMOV, whereas the I&E investigation, in contrast, makes no such assessment. It is apparent that the role of the EMOV in the deterioration of plant conditions during the accident was a paramount one. If operators had realized, early-on, that the EMOV had not closed, and had acted to shut the block valve, a significant volume of reactor coolant would not have been lost and the severity of the accident would, almost certainly, have been lessened. It is important to note, per both reports, that the operators' failure to realize the EMOV remained open was based, in part, on a degraded pre-accident plant operating condition. Prior to the accident, leakage from the EMOV, a non-safety related component, or the code safety relief valves, which are considered safety related, had been allowed to continue in excess of plant procedural limits. This condition, resulted in higher than normal temperature indications in the exhaust piping, which is common to all pressurizer relief valves, and led the operators to place little importance on the high temperatures observed in the exhaust during the accident. This temperature signal was assumed to be the result of continuing pre-accident leakage and not a stuck-open EMOV involving large coolant volume.

- Problems and failures associated with EMOV's at TMI-2 and throughout industry are discussed in NSAC (page ERV-2). According to NSAC, valve failures, involving the Dresser Model 31533 VX-30, are known to have occurred at TMI-2, Oconee-3 and Rancho Seco. Over 700 Dresser valves are stated to have been delivered to industry (non-nuclear included) since 1971. Historical perspective is comparatively lacking in the I&E study. Both reports do indicate, however, that the cause of the EMOV failure during the TMI-2 accident cannot be determined until workers can obtain access to the valve.

- Only the I&E report (page I-1-52) highlighted the fact that operators were worried about the possibility of failure of the EMOV block valve used to mitigate the effect of the stuck open EMOV. This block valve is the only means of isolating leakage through the stuck-open EMOV. Although the block valve was opened and closed a number of times during the course of the accident, the significance of the operators' concerns should be included in any future analysis. In other words, was there any point at which the block valve should have been operated, but it wasn't due to the concerns of operators? In addition, can the present design of these valves accommodate their use for mitigating possible accidents in the future?

- The NSAC points out (page TH-25) that earlier accident analyses probably did not consider a sequence of events involving a stuck-open EMOV. It is explained that a "non-safety" class component (i.e. the EMOV) would not be considered to mitigate an accident by opening and, therefore, the possibility of its not closing would not be considered in an accident analysis. This philosophy, if reflected in the training of TMI operators and in the plant procedures, may have contributed significantly to failure of plant operators to identify that the EMOV did not close. Furthermore, the concept of ignoring non-safety equipment failures as adding to the consequences of an accident appears to merit reconsideration.

- Apparent conflicts, both internal (within the I&E report) and between these two reports, exist with regard to reporting on the operation of the high pressure injection (HPI) pump MU-P-1A. The I&E report lists an unsuccessful attempt to start the pump at 13 seconds (page 1A-7) but also states the operators started MU-P-1A (page I-2-11). NSAC sequence of events (page 5) lists the pump as being started at 13 seconds into the transient with no mention of any difficulties in starting. Given the significance of HPI to ECCS generically and MU-P-1A in this accident specifically, this question should be resolved. A broader question is raised here. We are not aware if the staff is compiling a check list of equipment malfunctions which require resolution prior to any TMI-2 restart. You may wish to ask the staff for their and the licensees' plans in this regard.

- NSAC reports the possibility that even after sump pumps were tripped a siphon may have developed between the reactor building sump and the auxiliary building sump tank because of positive pressure in the reactor building (page ROUTES-2). The I&E investigation makes no reference to this possibility. Although significant amounts of radioactivity are not believed to have been transferred to the auxiliary building via the reactor building sump the possibility of an unknown transfer of liquid from the containment should be studied.

- The I&E investigation has preliminarily identified (page I-4-26) an item of non-compliance with plant procedures that prescribe a system pressure set point for tripping the reactor coolant pumps (1100 psig). The NSAC report makes no mention of this. It should be noted that RC pump trip philosophy is presently under study by the NRC (I&E Bulletins 79-05C and 06C) and the nuclear industry.

cc: Leonard Bickwit
Sam Chilk
Harold Denton
Vic Stello
Dick DeYoung