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TMI Program Office
Attention: Dr. B. J. Snyder, Program Director
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI 2)
Operating License No. DPR-73
Docket No. 50-320

Design Pressure for Containment and Future
Mechanical and Electrical Penetration Modifications

Following the submittal of Technical Specification Change Request No. 26 (TSCR. 26) there has been some discussion between the members of our respective staffs regarding what the realistic maximum pressure could be within the containment. Additionally a need to develop a standard criteria for containment penetration modifications was identified. The purpose of this letter is to summarize the results of several of the separate evaluations we have performed to bound potential containment pressurization in order to provide a conservative criteria for future modifications to containment mechanical and electrical penetrations. Additionally we are withdrawing our request to increase maximum internal containment pressure during normal operations to 1 psig (Item 6 in TSCR 26).

The Technical Specification concerning containment integrity was initially required to prevent a radiologically significant release to the environment occasioned by a containment pressurization and a concurrent potential release of radioactivity from the RCS to the containment building in a form and/or chemical state that would be releasable to the environment. In the present Unit 2 status, neither the high potential for a driving force (high Reactor Coolant System temperature and pressure) nor the high potential for a radiological release equal to that of the significantly more volatile radioiodines

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and noble fission gases is present. The transient that could potentially cause this statement to be invalid is core recriticality. But as discussed in your safety evaluation in support of our request for exemption from certain requirements of Appendix J to 10 CFR Part 50 this accident need not be designed against in reference to containment integrity based on Paragraph 4.1 of the PEIS which states that "the most probable (although very unlikely) cause of recriticality was found to be boron dilution, which would be a slow enough process that any approach to criticality can be detected and remedied".

To scope this analysis the radioactive material inside the containment has been subdivided into two fractions; that which is contained inside the reactor coolant system and that which is outside the system. We have investigated various mechanisms for pressurizing containment which results in an increase of airborne radioactivity in containment and eventual release to the environment.

The mechanism for release of radioactivity from the reactor coolant system to the containment would be a leak in the reactor coolant system. Our analysis for this event assumed that the 4000 cubic feet of water in the Reactor Vessel is at an initial temperature of 195ºF and is instantaneously released to the Reactor Building. This analysis yields a final reactor building pressure increase of less than 1.8 psig. This analysis is conservative for present conditions in that the average incore temperature is presently about 112ºF; a leak would occur over some extended period of time, not instantaneously, thus giving existing heat sinks time to absorb the heat released from the reactor coolant system. Thus realistically there would be little if any change in pressure resulting from this event.

For those radionuclides located outside the reactor coolant system the mechanism considered for simultaneously increasing the airborne radionuclide concentration and increasing pressure inside containment is a fire. Various fire scenarios have been evaluated. For example, we have evaluated the effect of a fire resulting in the release of heat from a 55 gallon drum of lube oil. Complete combustion of this quantity of oil would release approximately 8.53×10^6 BTU's of heat to the building. Using conservative analyses and accounting for the heat absorbed by the steel liner and the concrete D rings, this amount of heat could increase containment pressure by slightly less than 2 psi. An analysis which completely scopes the pressure versus time relationship of a postulated worst case fire inside containment, however, is very complex. But based on the example described above it is conceivable for a fire to increase pressure in containment to greater than 2 psig which could cause the failure of our most limiting penetration (Penetration R-626). Therefore the following actions are needed in order to reduce the potential sources for fire and radiological releases;

1. A method to control transient combustibles introduced i ~ the reactor building needs to be established; and

2. Reducing the amount of loose surface contamination that could potentially become airborne. This will be done during the upcoming decontamination

experiment and will continue through the gross decontamination phase of containment recovery.

Performance of the above listed items will reduce the radiological hazard of a fire in containment. However the possibility of a fire still exists, therefore we have also performed a highly conservative evaluation of the offsite consequences of a fire occuring in an area that will continue to be highly contaminated after completion of the decontamination experiment. For the purposes of this calculation it was assumed that a 14' horizontal section of the CRDM cable trays which are stacked vertically 8 high are all contaminated to a conservatively high contamination level of $3\ \mu Ci/cm^2$. Each tray is assumed (by estimation from in-containment photographs) to be 16" wide. All of the radioactive contamination associated with this area is assumed to become airborne and distributed throughout the containment. The resulting activity, 0.42 curies of total activity, results in an increased airborne concentration within containment of $7.4 \times 10^{-6}\ \mu Ci/ml$. Potential release of this activity from the reactor building to adjacent buildings, and then through station vent HEPA filters will reduce the activity released to the environment to less than $400\ \mu Ci$ assuming conservatively that all the airborne contamination remains airborne and able to be purged from the building. With dilution and filtering, the maximum long-term average concentration at the station vent is calculated to be $1.2 \times 10^{-9}\ \mu Ci/ml$. Further dilution during atmospheric release to the nearest site boundary conservatively reduces this effluent concentration by a factor of 10^3 to $1.2 \times 10^{-12}\ \mu Ci/ml$, a value below 10 CFR 20, Table II, column 1 values for the cesium and strontium radioisotopes present in containment.

Another potential scenario which could create a leak from containment is an increasing water level in containment which would cause an increase in pressure on containment penetrations which may become submerged. The assumed cause of this event is a leak in the Reactor Coolant System (RCS) and, using current procedures, charging the entire contents of the BWST (approximately 460,000 gallons) into the RCS and thence into the sump. To prevent this event from causing a leak from a containment penetration, penetrations which are below the 292.5 ft. elevation and thus potentially subject to water pressure, assuming an initial sump water level of approximately 3'5" (286'), will remain in their original design configuration which will withstand the increase in pressure. In support of this limitation the following additional observations are made:

1. Penetration R-401, which is potentially subject to water pressure, considered this in the design of its recent modification.

2. We will consider potential flooding of any penetration modification, using the event described above, for any proposed modification.

Based on the evaluation presented above and the withdrawal of our request to increase maximum internal containment pressure during normal operations, we believe that the present containment configuration is adequate with the design pressure for penetration R-626 (2 psig) presently being the limiting pressure barrier. Therefore, in the interest of developing a conservative criteria for on-going piping and electrical penetration modifications used for the recovery effort, The following general criteria will be used:

1. Piping or electrical penetrations modified for the recovery period only will be designed and built to withstand 5 psig and tested to hold 1.2 to 1.5 times this pressure for not less than 10 minutes in accordance with ANSI B31.1. Fluid hard piped piping through such temporary penetration will be designed, built and tested to ANSI B31.1.

2. The leak rate from modified penetrations, including flange and isolation valve leakage, will be limited to 100 sccm per 1 inch of pipe diameter. This value is similar to the acceptance criteria in approved TMI-2 Surveillance Procedure 2313 R-7 "Reactor Building Local Leak Rate Testing."

Be advised that future activities that involve modifications of the containment boundary other than piping and electrical penetrations will be evaluated and an assessment of offsite radiation doses will be made for other events besides fire. The quantities of radioactive material that may be released from the containment by postulated accidents will be derived based on the particular modifications and the design basis events. The results will be used to develop the criteria for that particular boundary modification.

Additionally, we are still evaluating the criteria we will use for the Containment Recovery Services Building's (CRSB) Containment Air Control Envelope. These criteria will be presented as part of the CRSB Containment Air Control Envelope. An appropriate Technical Specification Change Request will be made at that time.

If you have any further questions please contact Mr. J. E. Larson of my staff.

Sincerely,

J. J. Barton

J. J. Barton
Acting Director, TMI-2

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cc: L. H. Barrett,
Deputy Program Director