GPU NUCLEAR

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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Three Mile Island Unit No. 1 Nuclear Generating Station (TMI-1)
Docket No. 50-289
Facility Operating License No. DPR-16
Meeting Request for Clarification of Denial of License Amendment to Support a Design Change to Permanently Remove the Reactor Vessel Missile Shields at TMI-1

References: (1) USNRC Letter, "Denial of License Amendment Request Related to Three Mile Island Nuclear Station, Unit No. 1 (TMI-1), (TAC No. M98471)", dated September 16, 1998.

(2) GPU Nuclear Letter, 6700-97-2106, "Request for License Amendment to Support a Design Change to Permanently Remove the Reactor Vessel Missile Shields at TMI-1", dated March 31, 1997 (as supplemented by GPU Nuclear Letters 1920-98-20283 dated June 3, and 1920-98-20389 dated July 13, 1998).

The purpose of this letter is to request a meeting with the NRC Staff to clarify and discuss the basis for the Reference 1 denial of GPU Nuclear's request for a license amendment supporting a design change to permanently remove the reactor vessel missile shields at TMI-1.

GPU Nuclear would like to determine, as a result of this proposed meeting with the staff, what, if any, additional supporting information could be provided to address the staff's concerns and provide adequate justification to achieve approval of the subject request.

GPU Nuclear, in association with the Babcock and Wilcox Owners Group (BWOG), has expended considerable resources in the preparation of the technical basis supporting our original request, Reference 2. GPU Nuclear believes that this license amendment would

provide numerous benefits. These benefits include the elimination of a design basis heavy load event, a reduction in radiation exposure, the improvement of industrial safety, the ability to perform CRDM electrical repairs without requiring a plant cooldown and a reduction in outage critical path time and the associated loss of generation. The proposed amendment also provides a risk-informed approach to eliminate the consideration of low probability missiles using FMEA and crack growth models and is endorsed by NEI as a risk-informed application that has potential benefit to other PWR designs in addition to the B&W NSSS plants.

To assist the staff in their preparation for this meeting, Attachment 1 to this letter provides GPU Nuclear's view on some of the topics related to this issue that we would like to discuss.

As you are aware, GPU Nuclear's submittals on this topic contained some information of a proprietary nature. It is GPU Nuclear's intent to present and discuss this topic in a non-proprietary manner. However, should the discussion with the staff necessitate the presentation of proprietary information, these discussions should be limited to separate closed session between the participants.

GPU Nuclear requests that this meeting be scheduled at the earliest possible date. Please contact Ron Zak, Corporate Regulatory Affairs at (973) 316-7035 to confirm a mutually agreeable meeting date or if you have any questions or comments on this matter.

Very truly yours,

for A. H. Rone
Vice President and Director
Engineering

c: Administrator, Region 1
Senior Resident Inspector
TMI NRC Project Manager

ATTACHMENT 1

In Reference 1, the NRC staff concluded that the proposed removal of the TMI-1 Reactor Vessel Missile Shields from the plant design based on the analysis presented in Reference 2 was not acceptable. The Safety Evaluation Report (SER) contains several statements that are not fully understood by GPU Nuclear. In addition, the basis discussion for the staff's denial of GPU Nuclear's request was unusually brief, considering that the reasons for denial were based upon issues which were not previously discussed with GPU Nuclear and were not included in the staff's request for additional information. Babcock and Wilcox Owners Group (BWOG) members and Nuclear Energy Institute (NEI) also have an interest in achieving a better understanding of the NRC staff's position on this issue.

Some of the areas for clarification are provided below:

(1) Reference 1 states: "To date, the staff has not considered evaluations other than fully quantitative LBB fracture mechanics analysis as a basis for removing the dynamic effects associated with postulated high energy pipe ruptures..."

GPU Nuclear seeks to understand the reasoning behind the staff position that seems to commingle the SRP guidance with the more advanced leak-before-break (LBB) methodology.

GPU Nuclear does not believe that only fully quantitative LBB analyses have been the bases for defining credible missiles. The same Statement of Considerations (SOC) supporting the modification to GDC 4 that allowed for use of LBB arguments to remove local dynamic effects also noted (Commission Response to Issue 7) that "...Standard Review Plan (SRP) Section 3.6.2 has been used for more that a decade to postulate the number and location of pipe ruptures in nuclear power plants. SRP 3.6.2 ignores or treats indirectly many factors, such as material properties, potential corrosion and the potential for water hammer..."

The technical work contained in our submittal evaluates the consequences of all credible missiles. None of the credible missiles have the potential to do damage to the containment with the exception of the entire CRDM. Therefore the missile shields do not provide any design protection for the containment unless the entire CRDM remains a credible missile.

The submittal technically supports the claim that the occurrence of nozzle detachment (such as necessary to have a credible missile) is physically impossible during the design life of the BWOG plants. The evaluation of crack extension through wall and the time needed for a through wall flaw to grow circumferentially are based on FMEA and crack growth models and not primarily reliant on any LBB argument. The evaluation only adds LBB reasoning to demonstrate additional assurance that the crack growth is not an issue and to

provide the justification as to why more elaborate fracture analysis techniques or crack growth models are not necessary.

Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Head Penetrations," dated April 1, 1997, states that the NRC has reviewed the safety significance of Reactor Vessel Head Penetrations (VHPs) beginning in 1986 with reports of leakage pressurizer instrument nozzles made of the same Alloy 600 material as the Control Rod Drive Mechanism (CRDM) and other VHPs. The NRC determined that the cracking was not of immediate safety significance because the cracks were axial, had a low growth rate, were in a material with an extremely high flaw tolerance (high fracture toughness) and, accordingly were unlikely to propagate far. The evaluation concluded that these factors also demonstrated that any cracking would result in detectable leakage and the opportunity to take corrective action before a penetration would fail. After additional investigations in 1993, the NRC concluded in a safety evaluation that VHP cracking was not an immediate safety concern because 1) the Pressurized Water Stress Corrosion Cracking (PWSCC) cracks would be predominantly axial in orientation, 2) the cracks would result in detectable leakage before catastrophic failure, and 3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RV closure head would occur. Currently the NRC has accepted the generic industry response to GL 97-01. The NRC letter to NEI, dated March 21,1999, concluded that the integrated program proposed by NEI for VHP nozzles, which includes periodic inspections, is acceptable and that licensees responding to the GL may refer to the integrated program as a basis for assessing the postulated occurrence of PWSCC in the PWR-design VHP nozzles. GPU Nuclear understands that the NRC intends to issue letters to close this issue for individual licensee responses this summer.

(2) Reference 1 states: " Further, since CRDM penetration nozzles and housing assemblies are not high energy piping,"

GPU Nuclear desires clarification on why the staff believes the TMI CRDM motor tubes are not high-energy piping.

The SOC for GDC 4, high-energy piping is defined as those systems exceeding 275 psig or temperature exceeding 200° F. This same definition is applied within the Standard Review Plan Section 3.6.2. Reactor vessel head nozzles and CRD motor tubes meet both this pressure and temperature criteria. The stress analysis documentation from the original equipment manufacturer of the CRD mechanisms states "...These sections [of CRD motor tube] are basically uniform tubular section of various diameters and thickness with axially symmetric sections joining them." In other words, the CRD motor tube is pipe that was analyzed and documented to meet the minimum wall thickness, weld joint, and cyclic operational design limits of the ASME B&PV Code, Section III. Reactor vessel head nozzles at TMI-1, like all other BWOG plants were fabricated from the same product form hot-finished seamless tubing and from only 13 individual heats of Alloy 600 material. The Alloy was supplied by the Babcock and Wilcox Tubular Products Division or the

International Nickel Company and were ordered to the ASME B&PV Code, Section II, Specification SB-167 and relevant Section III requirements.

(3) The staff references November 1994 - NUREG 1061, Volume 3, "Report of the Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks" in the subject letter.

This NUREG describes general approaches for determining integrity (margin against full break) and states that other methods that can show equivalent crack stability margins will also be considered on a case-by-case basis.

GPU Nuclear feels the method presented by the BWOG and included in the TMI submittal was technically adequate to demonstrate integrity (margin against full detachment) for the purpose of eliminating the entire CRDM as a credible missile.

(4) Reference 1 states: " GPUN's proposal also does not provide any information to show that failure to remove the missile shield could adversely affect plant performance and safety, which was a consideration in the final rule SOC ... "

Presented below are some benefits of the change.

1) The elimination of two heavy load evolutions which, in effect, require six heavy load lifts per outage, would provide some, albeit qualitative, risk benefit. The change will eliminate one of the plant design basis heavy load drop events. TMI-1 is analyzed for the consequences of a heavy load drop of a missile shield as described in the technical analysis for NUREG 0612. An acceptable margin of nuclear safety is defined. The occurrence of a heavy load drop is more of an economic risk, because the unit may not remain economically feasible to operate after such an event.

2) The proposed design basis change would also eliminate the significant industrial safety risk associated with rigging and moving the 26 ton concrete missile shields.

3) The design change to remove the missile shields will also have significant performance benefit to the operation and maintenance of TMI-1. The specific benefits include a reduction of at least 10 to 12 hours in the critical path for outages that would normally require missile shield removal. The most significant performance benefit would be the reduction in cost and reduced need to take the plant to cold shutdown conditions to perform CRDM maintenance including position indication and electrical stator repair or replacements. Please note that the cost of radioactive waste from the processing of primary coolant water needed to reach cold shutdown conditions and lost generation are two of the major elements of this cost for each maintenance event.

4) The design change would also reduce personnel dose due to the reduction or elimination of activities described above.