GPU Nuclear

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June 1, 1982
5211-82-132

Office of Nuclear Reactor Regulation
Attn: Darrell G. Eisenhut
Director of Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Pressurized Thermal Shock (PTS) to Reactor Pressure Vessel

This letter is in response to your August 21 and December 18, 1981 letters requesting information concerning the PTS issue. As we indicated to you in our March 17, 1982 correspondence, we have proceeded with our analysis in-house and with B&W by using a realistic mixing model as well as TMI-1 plant specific data. We have placed a high priority on completing the analysis so that we could report to you in June. Since you need this report to brief the Commission during the early part of June, we are, with this letter, summarizing the results of our analysis. We feel that this information, much of which is excerpted from what will be the final report, will provide you with sufficient information to formulate recommendations.

We will issue the final report, currently under detailed review, the last week of June.

The analysis is very encouraging in that it concludes the following:

Reactor Vessel integrity will not be compromised due to low to moderate frequency events and anticipated transients during the designed lifetime of the vessel.

The analysis includes significant conservatisms that add to the margin of safety in maintaining RV integrity.

The current rate of embrittlement of the TMI-1 vessel may be reduced further if the plant switches to a low leakage fuel scheme in near term reloads.

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Because of the concerns raised by the PTS Issues, operator response will be significantly improved through increased awareness and additional training.

We hope that this letter and attachment will assure the Staff that there are no short term or long term concerns in regard to PTS. The final report the Staff will receive at the end of June will elaborate on these conclusions in greater detail. Likewise, GPU Nuclear's ongoing participation in Owner's Group Materials Surveillance Program will provide the periodic confirmation of the integrity of the reactor vessel.

Sincerely,

H D Hukill

H. D. Hukill
Director, TMI-1

HDH:PGD:vjf

Attachment

Attachment 1

GPU Nuclear - TMI Unit 1

NRC Letter of August 21, 1981 - REQUEST FOR ADDITIONAL INFORMATION

Question

1. Geometry

Geometrical description including design and as-built (when available) dimensions of the core, assemblies, shroud/baffle, thermal shield, downcomer, vessel, cavity, and surrounding shield and/or support structure.

Response:

The following references contain the requested geometrical descriptions for TMI-1:

- TMI-1 FSAR - Chapters 1, 3 & 4
- BAW 1628 - "RV Brittle Fracture Analysis During SBLOCA Events With Extended Loss of Feedwater"
- BAW 1648 - "Thermal Mechanical Report - Effect of HPI on Vessel Integrity for SBLOCA With Extended Loss of Feedwater"

Question

2. Material Description

Region-wise material composition and material isotopic number densities (atoms/barn-cm) for the core, near-core regions and RPV, suitable for neutron transport calculations.

Response:

The following references contain the available material descriptions:

- BAW 1511P - "Irradiation-Induced Reduction in Charpy USE of RV Welds"
- BAW 1439 - "Analysis of Capsule TMI-1E From Met-Ed Co. TMI-1 - RV Materials Surveillance Program"

Question

3. Neutron Source

Present and expected EOL:

a) Assembly-wise and core power history (EFPY)
b) Rod-wise and core power history (EFPY) for peripheral assemblies.
c) Core average axial power history distribution.

Response:

The following references contain the requested neutron source information:

- BAW 1511P*- "Irradiation-Induced Reduction in Charpy USE of RV Welds"
- BAW 1485P*- "Pressure Vessel Fluence Analysis for 177 FA Reactors"

Question

4. Vessel Fluence

a) Description of available calculations of the vessel fluence including fluence values, locations, and corresponding power histories (EFPY), including 1/4T, 1/2T and 3/4T through the RPV.

b) Description of available capsule-inferred vessel fluences including fluence values, locations, and corresponding power histories (EFPY).

Response:

The following references contain the requested vessel fluence descriptions:

- BAW 1511P* - "Irradiation-Induced Reduction in Charpy USE of RV Welds"
- BAW 1485P* - "Pressure Vessel Fluence Analysis for 177 FA Reactors"

Question

5. Surveillance Capsules

a) Capsule materials, radial and axial dimensions and locations.

* BAW 1485P and BAW 1511P do not reflect new fuel shuffle scheme (i.e., 18 month LBP Low Leakage Cycle.)

b) Capsule fluence measurements, together with the accumulated power history (EFPY) and a description of the lead factors used to extrapolate the measurements to the peak wall fluence location.

Response:

The following references contain the requested information on surveillance capsules:

- o BAW 1439 - "Analysis of Capsule TMI-1E from Met-Ed Co. TMI-1 - RV Materials Surveillance Program"
- o BAW 10006, Rev.3 - "RV Material Surveillance Program"
- o BAW 1543 - "Integrated Reactor Vessel Material Surveillance Program"

Question

6. Vessel Welds

Axial and aximuthal locations of vessel weld-seams with respect to the core. Overlay of current fluencemap with weld locations. Identify the critical welds, vertical and circumferential, and give the weld wire heat numbers. Give weld chemistry for the critical welds. For each weld wire heat number, report the estimated mean copper content, the range and the standard deviation, based on all the reported measurements for that weld wire heat. The welds may be surveillance weldments for your vessel or others, nozzle dropouts that contain a weld, weld metal qualification data, or archive material. In the absence of any information, assume that ccoper content is at its upper limit (0.35 percent when using R.G. 1.99, Rev. 1) and that the nickel content is high.

Response

The following references contain the requested vessel weld data:

- o BAW 1511P - "Irradiation-Induced Reduction in Charpy USE of RV Welds"
- o BAW 1439 - "Analysis of Capsule TMI-1E from Met-Ed Co."
- o BAW 10006, Rev.3 - "RV Material Surveillance Program"
- o BAW 1500P - "Chemistry of 177 FA B&W Owner's Group RV Beltline Welds"
- o BAW 1485P - "Pressure Vessel Fluence Analysis for 177 FA Reactors"

Question 7.

Systems Analysis

a) Provide a list of transients or accidents by class (for example: excessive feedwater, operating transients which result from multiple failures including control system failures and/or operator error, steam line break and small break LOCA) which could lead to inside vessel fluid temperatures of 300°F or lower. Provide any Failure Modes and Effects Analyses (FMEAs) of control systems currently available or reference any such analyses already submitted. Provide the analysis of the most limiting transient or accident with regard to vessel thermal shock

considerations. Estimate the frequency of occurrence of this event and provide the basis for this estimate. Discuss the assumptions made regarding reactor operator actions.

b) Identify the computer programs used to calculate the limiting transient or accident. Indicate the degree to which the computer programs used have been verified and any other additional verification requested to demonstrate that the computer program models adequately treat the identified important physical models (i.e., ECC mixing, heat transfer, and repressurization).

Response:

There are a vast number of possible overcooling event sequences that can be postulated to occur. Design basis events, such as loss of coolant accident (LOCA) and main steamline break (MSLB) have already been addressed previously during the original licensing process. Thus, we have chosen a logical approach to identify those overcooling transients to analyze in detail - that is, we chose event sequences that provide an appropriate balance between likelihood of occurrence and severity of cooldown. Results of various independent programs have contributed to the selection processs and have resulted in a significant reduction in the number of overcooling and SBLOCA events that require detailed analysis, as well as a better understanding of system response once an event is postulated to occur.

Therefore, in response to Question #7 - Systems Analysis - we have chosen the following two classes of events that yield conservative but realistic scenarios that require evaluation of the vessel in terms of thermal shock:

Overcooling Transient

Overcooling of the plant occurs during events which result in increased steam flow with subsequent secondary side depressurization, reactor trip and continued feedwater injection.

Events of this type which result from pipe breaks are less likely to occur than those which occur from equipment malfunctions. In this regard, a single failure in the Integrated Pressure Control portion of the Integrated Control System (ICS) results in a full open signal to the Turbine Bypass Valves. Severity of a cooldown transient is judged on its initial cooldown rate, the minimum temperature, and the eventual repressurization of the RCS if this also occurs in the scenario. Events with a fast cooldown rate, low minimum temperature, and subsequent repressurization are considered most challenging to the reactor vessel's integrity. A comparison of the turbine bypass failure event with other potential cooldown events is provided in Table 1. This table is based upon previous analyses, engineering judgment and experience and shows that failures which have a more severe overcooling potential are less likely to occur. Similarly, more likely failures have a smaller potential for overcooling. The Turbine Bypass Valve Failure was chosen as the overcooling transient to be evaluated for reactor vessel thermal shock on the basis that it has a relatively large likelihood of

TABLE 1

COMPARISON OF OTHER OVERCOOLING EVENTS WITH THE BASE CASE TURBINE BYPASS VALVE FAILURE EVENT

DESCRIPTION FAILURE	RELATIVE SEVERITY	RELATIVE LIKELIHOOD OF OCCURRENCE	COMMENT
Emergency Feedwater Overfill	Less	Comparable	This type of event was analyzed in section 8.0 of the TMI-1 Restart Report
Stuck Open Secondary Safety Valve or Atmospheric Dump Valves	Less	Comparable	TMI-1 Restart Report Section 8.0
Stuck Open Turbine Stop Valve	More	Less*	More severe overcooling initially, but more likely to be terminated quickly by operator action i.e., the first two post-trip actions are to verify reactor and turbine trip. Therefore, overall severity judged to be less.
Small Steamline Breaks	Comparable	Less	
Large Steamline Breaks	More	Much Less	Analyzed in TMI-2 FSAR Appendix 15B (Analysis of Large Steam Line Breaks), which bounds TMI-1
Total Loss of Feedwater With HPI Cooling	More	Less	Analyzed in this report See Section for Small Break LOCA.
SMUD Event of 3-20-78	More	Much Less	NNI/ICS Power Supply Modifications Reduce Likelihood of this event.

*The turbine control and intercept valves would also have to fail open, which is considered less likely than a failure of the turbine bypass valves.

occurrence as well as degree of severity.

SBLOCA Transient

The criteria used for selecting the SLOCA transient are as follows:

1. The transient must be realistic. It may be a result of a normal operating transient and will evolve in a mechanistic manner.

2. The transient must have a reasonable probability of occuring. Systems must be challenged to fail and may contain single or multiple failures, i.e., an initiating event plus one or more failures.

3. The transient provides a realistic challenge to the reactor vessel. The transient and resulting failures provide a thermal shock to the reactor vessel.

The initiating event (i.e., loss of all feedwater) for the SBLOCA transient was chosen to provide a mechanistic manner in which to challenge the system and provide the necessary criteria for obtaining a thermal shock of the vessel, i.e., by providing a once-through mode of cooling. The mode of cooling is the cold HPI flow which is injected into the cold leg of the plant, passing through the core, and out of the system through a break. Therefore, the initiating event that will challenge the code safety in a mechanistic manner is a loss of all feedwater. In order to allow the code safety valve to be challenged during the transient, the PORV is assumed to be isolated.

The mechanistic break size chosen using these criteria was a single TMI-1 pressurizer code safety valve (0.018^{2} ft). The PORV break size was considered in BAW-1648 since it led to repressurization of the RCS. However, the transient provides a less severe thermal shock than the large safety valve break because the RCS temperature remains considerably warmer than with the 0.018-ft^{2} break, and the HPI flow is less than with the 0.018-ft^{2} break.

Break sizes larger than 0.018-ft^{2} result in more rapid depressurization to pressures at which the low-pressure injection (LPI) system provides makeup (with little or no repressurization). The transient response of these larger breaks is similar to that of the large break LOCA being considered under NRC Task Action Plan A-11.

An examination of the system response for various break locations shows that the limiting condition will occur for a break in the pressurizer or hot leg independent of break size. This results from the following considerations:

For cold leg breaks, the hot water leaving the core flows (1) through the hot leg, steam generator, and broken cold leg to the break; and (2) through the vent valve, downcomer, and broken cold leg to the break. The latter path has the least flow resistance and thus allows a large portion of the hot water to enter the downcomer for mixing. Furthermore, the

diversion of HPI water to the break reduces the total amount of HPI water entering the downcomer. For hot leg or pressurizer breaks, more HPI water is available to enter the downcomer. In addition, less vent valve flow occurs in a hot leg or pressurizer break, thus decreasing the amount of hot water available for downcomer mixing. The analyses that follow used a pressurizer break to evaluate system conditions.

It is further concluded that breaks within the hot leg will respond in the long term in a fashion similar to those in the pressurizer. Thus, the combination of system response, break location, and break size determined that the evaluated accident would be the failure of a pressurizer code safety valve.

The SBLOCA transient thus analyzed, is a pressurizer code safety valve (0.018-ft2) failing open after being challenged by high pressure as a result of a loss of all feedwater. The valve cannot be isolated and thus continues the blowdown until the RCS can be placed on the decay heat removal (DHR) system. Since feedwater is not restored during the event, both forced and natural circulation RCS flow are assumed to be lost for the duration of the event.

Response to Question 7b

ANALYTICAL MODEL DESCRIPTIONS

Overcooling Transient

The Turbine Bypass Valve failure overcooling transient was analyzed by using the system analysis code RETRAN and the dynamic simulation code CSMP (Continuous System Modeling Program). For the first twelve minutes, during which a best estimate of primary and secondary parameters is necessary, the one-dimensional multi-node RETRAN program provides the level of detail required to model the event. The overcooling is essentially terminated by isolation of the steam leak and termination of the HPI; therefore, the long term system response is an energy balance calculation which can be efficiently performed by using the dynamic simulation language CSMP. Both models are discussed below.

The TMI-1 RETRAN Model

RETRAN-01 is the first released version of the computer code package developed by EPRI for analysis of light water reactor operational transients. It is based on the RELAP series of codes using homogeneous equilibrium flow equations and has been through extensive verification and benchmark against separate effects tests and plant transient data conducted by GPUNC on events which have occurred at both TMI-1 and TMI-2 for which actual recorded data was available.

The CSMP Model for Long Term Cooling

In order to "terminate" the overcooling event within the first few hours after initiation, a stabilized RC temperature must be maintained. The plant is controlled by using either the steam generators as a heat sink or by the primary side feed and bleed mode of operation. The transient

is slow-varying with time and there are no sudden variations expected between the various locations in both the primary and the secondary systems. Therefore, the nodal diagram can be simplifiedto two volumes - one for the RCS and one for the secondry side. Instead of using RETRAN, the mass and energy balance equations can be solved very efficiently using the dynamic simulation language CSMP (Contnuous System Modeling Program) for an extended transient time.

SBLOCA Transient

The CRAFT Model

An eight-node CRAFT model was developed to determine the thermal-hydraulic conditions existing in the RV for the SBLOCA analysis. The system noding for the eight-region model is described below:

Node 1: Reactor vessel downcomer and lower plenum
Node 2: Reactor core and upper plenum
Node 3: Cold legs between RC pumps and reactor vessel
Node 4: Hot legs
Node 5: Primary side of steam generators and cold legs between steam generators and RC pumps
Node 6: Secondary side of steam generators
Node 7: Pressurizer
Node 8: Containment

Node CFT: Core flood tank

The CRAFT code assumes homogeneous mixing of the liquids in a node and determines its thermodynamic conditions based on thermal equilibrium between the steam and liquid phases. This assumption will result in complete mixing of the cold and hot fluids entering the downcomer region from the cold legs and vent valves, so the downcomer node temperatures calculated by CRAFT are mixed mean temperatures. However, this temperature is not used in the mixing or fracture mechanics analysis. The primary purpose of the LOCA analyses was to determine the HPI flow rate, vent valve flow rate and temperature, and RCS pressure for use in the fluid mixing analysis.

The SBLOCA analyses used the CRAFT code only during the blowdown stage of the transient until the core outlet temperature became 100°F subcooled. After the RCS refilled with water, a steady-state analysis was performed to determine the RV conditions. This steady-state analytical method was benchmarked in BAW-1648 against the CRAFT analysis for the 0.007-ft2 pressurizer break with no operator action. The results indicate that the steady-state code predicts the downcomer temperature approximately 9% above the CRAFT prediction. The 9% deviation in the downcomer temperature was used as an adjustment factor for the breaks analyzed in BAW-1648 as well as the present analysis.

The COMMIX-1A Model

The COMMIX-1A (advanced version of COMMIX-1) is a three-dimensional, transient, single-phase computer code for thermal hydraulic analysis. It

uses the ICE technique of Harlow and Amsden to discretize the conservation equation of mass, momentum and energy. The set of discretization equations are then solved using the cell by cell (point by point) iterative procedure. To permit the analysis of a flow domain with solid objects, the porous medium formulation with surface permeability, volume porosity, distributed resistance and distributed heat source are incorporated in the conservation equations. The "Force Structure" model and the "Thermal Structure" model in COMMIX-1A permit calculations of distributed resistances and distributed heat sources, respectively. These models are designed such that we can use different correlations at different grid loations. In addition, the code has various options permitting a large amount of flexibiility (e.g., use of Cartesian or cylindrical coordinate systems, various rebalancing schemes for speedy convergence, automatic time step selection, implicit energy option for low flow cases, etc.).

The code has been developed and refined over a number of years, and already a large number of computations for complex situations have been performed. The following is a list of some of the problems analyzed using COMMIX-1A.

A. Pretest Prediction of the W-1 SLSF Experiment
B. Hexagonal Fuel Assembly with a Planar Blockage
C. Nineteen-Pin LMFBR Fuel Assembly in A Hexagonal Duct with Power Skew
D. Flow Stratification in a Horizontal Pipe
E. Simulation of LMFBR Outlet Plenum Mixing
F. Analysis of LOPI transient
G. Simulation of P2 Transient Free Convention Test
H. CRBR Upper Plenum Under Thermal Stratification
I. German 7-Pin Flow Rundown Test
J. Solar Pond Heat Loss

Although the COMMIX-1A has been develped for thermal hydraulic analysis of LMFBR fuel assemblies, the code is designed to permit applications to other components and other reactor types, e.g., PWR. This can be seen from some of the applications listed above. In addition the COMMIX-1A code has been benchmarked and produced reasonable agreement with (a) experimental mixing tests conducted by B&W Alliance Research Center in Alliance Ohio, and (b) test data obtained from the one fifth scale model CREARE test facility that approximated the B&W 177 FA geometry.

NRC Letter of December 18, 1981 - Enclosure (2)
AMPLIFICATION OF THE "150-DAY" REQUEST
TO THE AUGUST 21, 1981 LETTER

Question

(1) Identification of the PTS events that were considered in reaching your conlcusions, and a justification for PTS events that you did not consider. You should include a quantitative assessment of the probability of occurrence of the various PTS events considered and not

considered and an accompanying assessment of the likelihood of vessel failure vs. EFPY for the events. The manner in which you considered multipie failures of systems, components, and those resulting from operator actions should be described in detail.

Response:

Most of our response is contained in the response to Question #7 of the 8/21/81 NRC letter.

Additionally, although GPU Nuclear has not embarked on a Probabilistic Risk Assessment (PRA) program at this time, work conducted by Duke Power Co. for Oconee III and reported to the NRC on January 15, 1982 in response to the PTS issue, has shown probabilities of vessel failure to be extremely low based on the relative threat of severe overcooling events.

GPU also believes, that as further refinements are made to the models used in the PTS analyses, the likelihood of events that may lead to vessel failure will further diminish.

Question

(2) A description of the steps, if any, you are taking now or plan to take in the near future to delay the rate of further embrittlement of your vessel, and your assessment of the effectiveness of those steps.

Response:

Currently TMI-1 employs a 12 month "Modified Out-In" refueling pattern. For economic reasons, GPU Nuclear is evaluating the transition to 18 month refueling cycles in future operations. Preliminary estimates indicate that the Lumped Burnable Poison (LBP) "In-Out-In" reloading, that can be used in the longer cycles, would reduce the vessel fluence rate by 25%.

Question

(3) Your assessment of the sensitivity of your analyses to uncertainties in input values, such as initial crack size, copper content, fluence, and initial reference temperature at welds.

Response:

As part of the long range program on RV Integrity, the B&W Owner's Group is currently investigating quantitatively, the sensitivity to the various parameters used in the PTS analysis. Results of this analysis will be communicated to the Staff as part of the Owner's Group continued effort to keep the Staff appraised of significant phases in the materials program.

Question

(4) A list of assumptions relied upon in reaching your conclusions.

a. If this list includes "credit" for operator actions, describe the basic instructions given the operators (for example, if a "sub-cooling" band is used, describe it). Submit the procedures the operator will follow, and describe the training being given to establish operator readiness to cope with PTS events.

b. If the list includes credit for the effects of warm prestressing for some event sequences, include your justification and analyses showing that such events will follow a pressure-temperature pathway for which warm pre-stress is effective.

Response:

Overcooling Transient Assumptions

The transient is initiated by a secondary side upset (i.e., all six turbine bypass valves suddenly open) The blowdown is further increased by the assumption that the Turbine Control Valve remains full open. The initiating event and control valve failure result in a maximum blowdown of 127% of normal steam flow. This excessive heat removal causes the PCS temperature and pressure to decrease rapidly. Owing to the negative temperature coefficient of the core, excessive reactivity is introduced and the reactor is tripped on high neutron flux. The following is a complete list of the analysis assumptions.

Initiating Event -	Simultaneous opening of all Six Turbine Bypass Valves.
Reactor Trip -	105% neutron flux.
HPI initiation -	1600 psig primary pressure (ESAS)
Internal Vent Valves -	0
Core Flood Initiation -	600 psig primary pressure
Core Flood Temperature-	100°F
BWST/HPI Temperature -	40°F
Core Decay Heat -	1.0 ANS
Break Flow -	(a) 0.523 ft^2 Turbine Bypass Area maximum flow = 27% of full power steam flow. (b) Turbine control valve remains full open.
RC Pump trip -	ESAS plus 30 sec.
EFW System -	Maximum flow to OTSG level setpoint

SBLOCA ANALYSIS ASSUMPTONS

Consideration of various requirements for system response and its effect on thermal shock, as outlined, resulted in the selection of a transient that unfolds as a result of a pressurizer safety valve failing open with a loss of all feedwater. This results in the core being cooled in the once-through HPI cooling mode of operation. The cold HPI water is injected in the cold leg, enters the downcomer, and could result in large thermal gradients in the vessel wall.

The TMI-1 plant specific SBLOCA anlysis assumptions are as follows:

CRAFT Model	- 8 Node Model used
Break Size	- .018ft^2 , one TMI-1 pressurizer code safety valve
Location	- Top of pressurizer, code safety valve
Initiating Event	- Loss of all Feedwater at time zero
Reactor Trip	- Reactor TRIP on high RCS Pressure, 2355 psig
RC Pump	- TRIP RC pumps immediately on loss of subcooling margin or on ESFAS initiation, whichever occurs first
HPI Initiation	- Initiate HPI on ESFAS, 1600 psig
HPI System	- TMI-1 plant specific - 2 HPI pumps with venturis
BWST/HPI Temperature	- 50°F
Internal Vent Valves	- 8
Core Power Level	- 2568 Mwt
Decay Heat Assumption	- 400 EFPD with 1.0 ANSI curve
Structural Metal Heat	- Includes all structural metal heat
Break FLOW Model	- Subcooled, Bernoulli with C_D = .7
	- Saturated or steam, Moody Correlation with C_D = 1.0

OPERATOR ACTIONS

In both the overcooling and SB LOCA transients, the analyses assumed the operator throttled HPI to prevent the system from becoming more than 100°F subcooled. The operator controls the flow to maintain the system between 50°F and 100°F subcooled.

These actions are consistent with the plant HPI throttling criteria that requires the operator to bypass the ESAS signal and throttle HPI only if one or more of the following criteria are met:

1. HPI must be throttled to prevent violation of the applicable brittle fracture curve limitations.

2. HPI may be throttled if LPI flow is greater than 1000 gpm in each

line and stable for 20 minutes.

3. HPI may be throttled if the required subcooling margin (50°F except for OTSG tube rupture, then 20°F above 1600 psig, 50°F below 1600 psig) exists and pressurizer level is established 100".

NOTE: The margin to saturation is determined by the saturation margin meter and/or the average of the five highest operable incore thermocouples.

The above required actions are independent of the event and are posted in the control room.

In addition to throttling HPI, the analysis of the overcooling transient assumed the operator isolated the steam leak at twelve minutes into the event. We believe this is very conservative since we would expect the operators to identify an overcooling event within two (2) minutes using the Pressure-Temperature plot. Plant procedure EP-1203-24 requires the operator to isolate the steam leak as soon as it is identified. The first action the operator finally takes is to isolate the Turbine Bypass Valves.

OPERATOR TRAINING

The operators continually review the throttling criteria of HPI as well as the basis for the criteria. Included training activities include the studying of the procedures* in great detail in preparation for NRC examinations. As an example, one of the classes on mitigating core damage entitled "Potentially Damaging Conditions" deals with the HPI throttling criteria and basis supplemented by the use of the P/T (Pressure/Temperature) plot trainer.

Likewise, the operators received instruction at the B&W simulator on the basis of the Heatup/Cooldown curves where they also receive instruction on OTSG overfill and Non-LOCA overcooling events. Addtionally, the operators are trained on the simulator using procedures to respond to various LOCA transients.

During 1982, additional training will be conducted on PTS. In fact, our management is currently evaluating detailed lesson material under preparation by B&W, for an operator training program that familiarizes the operators specifically with reactor vessel thermal shock and the concerns, should a severe transient occur.

Implementation of the course material in the training program is scheduled for this summer and will generally follow this outline:

*The following procedures contain HPI throttling criteria: EP-1202-4, EP-1202-5, EP-1202-6B, EP-1202-6C, EP-1202-39, EP-1202-6B, EP-1202-2A EP-1202-2, EP-1202-14, EP-1202-26A, EP-1202-26B, EP-1202-36A, EP-1202-36B, OP-1102-16, OP-1105-3

- Reactor Vessel Thermal Shock (T/S) Description
    - Purpose of T/S Lessons
    - Discussion of T/S Scenarios
    - Description of Brittle Fracture Failure Scenarios
    - as it applies to Reactor Vessels

- Factors that Increase the Possibility of Brittle Fracture Failure of Reactor Vessels
    - Severe Cooling of Vessel Inner Surface due to transients
    - Irradiation Damage to Vessel Material
    - Assumed crack geometry
    - Operator actions
    - Mixing in Cold Leg & Downcomer

- Effects of these Factors and Why They Are of Concern to Thermal Shock

- Reasons for Operator Actions Assumed in T/S Analysis
    - Actions for SBLOCA
    - Actions for Overcooling Transients
    - Actions for LOCA

- Recognizing Symptoms and the Use of the Symptom Oriented Procedure That Will Minimize the Occurrence of Pressurized Thermal Shock

- Recognize RC System Response

- Use of Symptom Oriented Procedure

- Operator Actions in Symptom Oriented Procedures That Will Minimize the Occurrence of Pressurized Thermal Shock

<u>Warm Prestressing</u>

Finally, although we have not taken credit for warm prestressing, we feel that warm prestressing is a demonstrable and valid physical phenomena that has been experimentally proven. We feel that warm prestressing is a phenomenom that adds additional margin to the service life of the vessel and can be considered when certain highly unlikely but challenging transients are postulated.

NRC Letter of December 18, 1981 - Enclosure (1) - AMPLIFICATION OF THE "150-DAY" REQUEST TO THE AUGUST 21, 1981 LETTER

<u>Question</u>

1. RTndt

<u>Response</u>:

The Staff has accepted our values presented in response to the "60 Day" letter.

Question

2. Rate of Increase of RTndt

You have provided rates of increase in fluence per EFPY for your reactor vessel. We accept these values. However, please provide the rate of increase of fluence at the critical longitudinal weld location taking into consideration any contemplated changes in core configuration.

Response:

As we indicated in our response to Question (2) of Enclosure (2) of the NRC Letter of December 18, 1981, we are studying the implementation of an 18 month Lumped Burnable Poison "In-Out-In" fuel scheme. If implemented the 18 month LBP scheme would have the additional benefit of a reduction in fluence of approximately 25%. This is applicable to all critical longitudinal and circumferential welds.

Question

3.& 4. RTndt Limit and Basis for the Limit

Since the "60 day" response stated that you do not consider a limit on RTndt to be an appropriate basis for continued operation that, if implemented, would assure maintenance of a acceptable low risk of vessel failure from PTS event for the near-term, pending longer term results of more detailed analysis or research. We will be developing this criterion considering recommendations that you may provide in your "150 day" response.

Response:

GPU Nuclear feels that an appropriate criterion to assure an acceptable low-risk of vessel failure should derive from the work currently underway under the B&W Owner's Group Reactor Vessel Integrity Program. This periodic evaluation will allow for advance warning of any potential problems while accomodating advances in technolgy. These problems could be accomodated within the existing framework of Appendix G by allowing an alternate means of determining material toughness through emerging Elastic Plastic Fracture Mechanics techniques

Question

5. Operator Actions

We are aware through the TMI-1 restart hearing proceedings of the emphasis placed on the overall concern of PTS at TMI-1. We are aware that this issue is addressed in procedures, training and management involvement and that operators are sensitive to the thermal shock considerations. However, we cannot determine from your "60 day" response the degree of emphasis which is currently placed on the issue in training

and management involvement.

We request that you expand your response to provide us a more detailed discussion of what steps have been taken to ensure that your operators have a firm grasp of this issue and can be expected to cope with the events which serve to initiate PTS.

Response:

Please refer to the response to Question (4) of the NRC Letter of December 18, 1981 Enclosure (2).

NRC Letter of August 21, 1982, p.3

Question

You are also requested to submit a plan for Three Mile Island, Unit No. 1 to the NRC within 150 days of the date of this letter that will define actions and schedules for resolution of this issue and analyses supporting continued operation. We request that you include consideration and evaluation of the following possible actions:

(1) reduction of further neutron radiation damage at the beltline by replacement of outer fuel assemblies with dummy assemblies or other management changes;

Response:

The TMI Plant Specific Report will be submitted to the Staff at the end of June, 1982.

As we discussed in our response to Question (2) of Enclosure (2), NRC Letter of 12/18/81, we are evaluating the transition to an 18 month LBP fuel scheme. An approximate 25% reduction in fluence would result from the LBP refueling approach.

Question

(2) reduction of the thermal shock severity by increasing the ECC water temperature;

Response:

Consistent with the results of our analysis, we do not intend to increase the ECC water temperature. Our reasoning, based on the parameters in the analyzed transients (i.e., ECC temp. of 40° and 50°) and based on the benchmarked mixing Commix 1A model, indicate minimal advantage in raising the ECC temperature. As part of our PTS evaluation, we undertook a brief

study of actual BWST temp during the history of TMI-1 operations. The lowest temperature recorded was 58°F for one day while an 80°F temperature is the norm. Therefore, we conclude that our analysis has a degree of conservatism based on the historical data.

Question

(3) Recovery of RPV toughness by in-place annealing (include the basis for demonstrating that your plant meets the requirements in 10 CFR 50 Appendix G IV C).

Response:

GPU Nuclear within the B&W Owner's Group is continuing to monitor the current EPRI sponsored research program on in place annealing.

Question

(4) Design of a control system to mitigate the initial thermal shock and control repressurization.

Response:

As our analysis and our discussions on operator actions indicates, we consider no changes are needed to our existing control systems. Likewise the upgraded EFW, cavitating venturis in the EFW and HPI systems, redundant power supply to instruments and the upgrading of the position indication of the PORV and pressurizer safety valves, have all contributed to the enhanced system response to mitigate potential PTS challenges to the vessel.