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

April 10, 1979

MEMORANDUM FOR: Victor Stello, Jr., Director, TMI Operations

FROM: Don Davis, Duty Officer Shift B

SUBJECT: RESPONSE TO REQUEST FOR HEADQUARTERS ADVICE ON COOLDOWN

Attached is the Headquarters Advisory Report on Cooldown from 280°F to 150°F at 1000 psia. Our review of secondary systems has not been completed and has been hampered to some degree by a lack of understanding of the planned operational modes and any plant modifications that will be made.

Don Davis

Don Davis, Duty Officer Shift B

cc: R. Mattson

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ADVISORY REPORT

REACTOR SYSTEM COOLDOWN

TMI-2

The purpose of this report is to summarize the recommendations of the HQ Technical Support Teams with respect to the cooldown of the reactor system. In particular, that phase of cooling down the reactor system from a temperature condition of about 280°F, to a temperature of 150°F at a coolant pressure of about 1000 psi (see attached figure).

The following are those subjects or areas that were considered:

1.0 Systems

2.0 Radiolysis

3.0 Potential Criticality

4.0 Engineering (mechanical capability of components)

5.0 Containment

6.0 Radiological Consequences

The following discusses each subject in detail:

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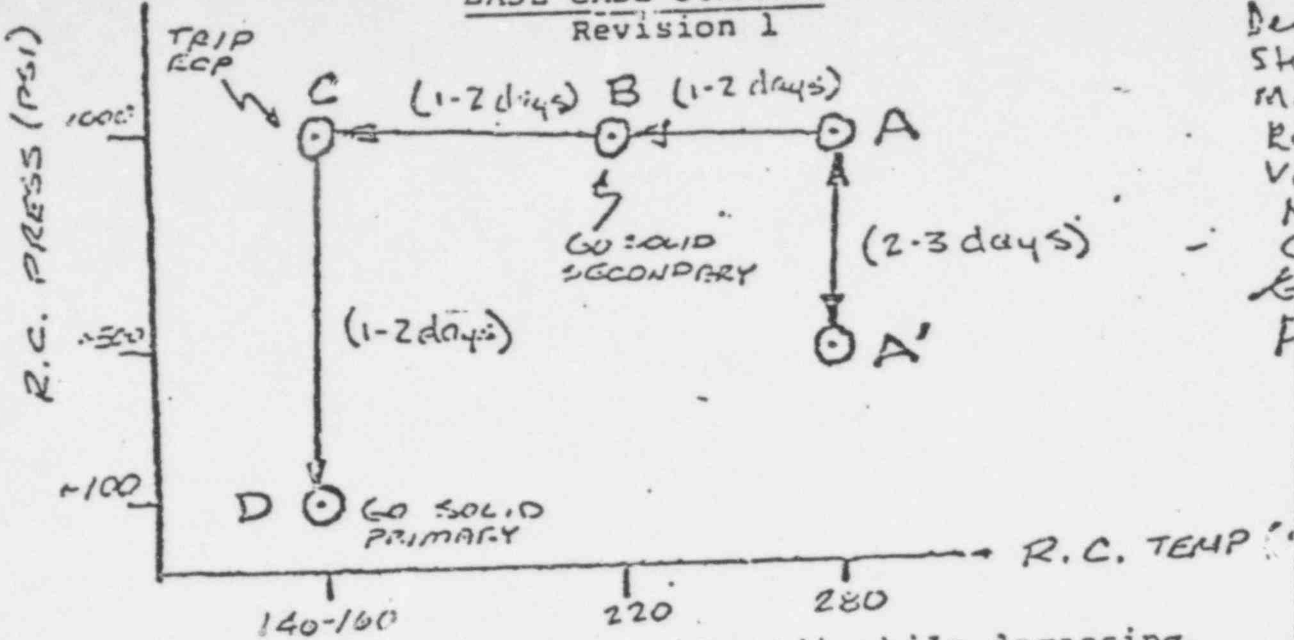
BASE CASE SUMMARY
Revision 1



(1) Degas at A; Lower Pressure (A→A') while degassing, then return to A.

(2) Continue Design/Installation of static and active systems for primary makeup/pressure control and secondary cooling system for "B" S/G.

(3) Reduce temperature (A→B) by steaming on "A" S/G

(4) Take "A" S/G solid - drop primary temp. to minimum (B→C

(5) Trip RC Pump "A" - Establish natural convection - Establish cooling to "B" S/G if available.

(6) Drop primary pressure to selected value (C→D)

(7) Take primary system solid - Control pressure & makeup with static or new active system

END POINT

Primary - Natural Circ, solid liquid, Long-term P/V Control

Secondary - Solid water, Long-term Heat Dump System

Approved for Issue:

R. Arnold

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1.0 SYSTEMS FAILURE MODES AND EFFECTS

Going From Point A to Point B in Base Case Summary (0?/06/79) - holding primary pressure at 1000 psig and reducing primary temperature to approximately 150°F.

Instrumentation Required:

1. Pressurizer level
2. Primary pressure
3. Incore outlet T/C
4. Loose parts monitoring system
5. Relief block valve position
6. Pressurizer spray valve position
7. Steam generator level

Areas to be Considered:

1.1 Loss of Secondary Cooling - Increases primary pressure (a help for NPSH for RCP and absorption of gases). Some corrective actions are for the operator to restore secondary cooling by turning on either another condensate pump, an auxiliary feedwater pump or use steam generator atmosphere dump valve as required. Relief valves in the secondary system may also be employed to dump steam to the atmosphere. If the steam becomes contaminated by primary system water due to a leaking steam generator tube, dumping to the atmosphere must be terminated and an alternate method of cooling such as HPI or the Decay Heat Removal System must be used. Monitor pressurizer level and pressure and stabilize conditions in primary system.

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1.2. Increasing Pressurizer Level - due to loss of letdown, increased MU flow or loss of secondary cooling.

Any of the above actions could lead to having the primary system water-solid with the potential for system overpressure if fairly rapid corrective means are not employed. For loss of letdown flow, modified Emergency Procedure 5 mitigation steps (not rest of procedure) should be employed. If increased makeup flow is detected, it may be reduced by throttling valves MU-V17 or MU-V18 or by temporarily turning off the makeup pump(s); if caused by inadvertent start of another makeup pump, shutdown the second pump.

Loss of secondary cooling is addressed in 2.1.1.

Isolation of the seal return line on the shutdown RC pumps may also be used to help mitigate the potential overpressurization of the RCS.

If the pressurizer level continues to increase and water-solid conditions appear imminent in spite of all actions taken above, the pressurizer vent valve (RC-V137) should be used to prevent overpressurization and if this is insufficient, the block valve (RC-V2) should be opened (use of RC-V2 should be kept to a minimum since continued or intermittent flow across the valve seat may effectively destroy the seat after a time since this valve may not have been

designed for flow modulation or multiple closures with large dif-
ferential pressures across the valve).

Note: the above procedure has not recommended tripping the oper-
ating RCP. Should the RCP be allowed to continue to operate when
in a water solid condition (as we recommend) the operators should
be prepared to open the pressurizer vent valve (RC-V137) to mitigate
any slow pressure transient due to tripping the RCP.

Note: Should the RCP be turned off and the MU pump started, it is
imperative that the makeup pump flow control valve MU-V17 be
closed initially to prevent overpressurizing or thermal shocking the
primary system. MU-V17 could then be carefully opened to control
system pressure. If the makeup pump bypass line is not available, it
may not be desirable to operate the makeup pumps when in a water-
solid condition.

3. Effect of opening safety valve - gives an uncontrolled blowdown
which may reform vessel bubble.

If the valve recloses, the system must be checked for any large gas
bubbles; these must be removed for they have the potential for pre-
venting the cooldown process. Degassing must continue, in any case,
to remove any small bubbles occurring throughout the RCS as a result
of inadvertent depressurization. If the valve sticks open, maintain

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water levels, with the makeup system. This method of cooling can only be maintained as long as the water inventory within containment is at a low enough level so as not to affect containment integrity or continued operation by flooding systems and instrumentation and only as long as there is a borated water supply. Eventually, continued cooling of the core would require the use of the decay heat removal system.

1.4 PORV Block Valve Open - Similar effects, but not as severe as 3.

1.5 1" Vent Line Isolation Valve Open - Similar effects, but not as severe as 4.

1.6 Loss of RCP w/Restart - No problems anticipated.

1.7 Loss of RCP w/No Start of RCP - assume there is no natural circulation and the RC pump trips. Assume cannot restart any RCPs. Go to EP-4 which involves makeup injection. An alternate method is use of the decay heat removal system.

EP-4 controls pressure from the MU pumps via PORV block valve and 1" vent valve isolation valve. We recommend using valve MU-V17 to control pressure as much as is practicable Use of PORV block valve is least acceptable since if the block valve stays open it will cause an uncontrolled blowdown. Should the PORV block valve fail closed, the system may overpressurize and cause the safety valves to open, resulting in an uncontrolled blowdown - addressed above in Section 3.

Actuation of an HPI train should be in a controlled manner through the manipulation of any HPI pump bypass lines and/or return test lines in order to minimize thermal shock on the primary system.

1.8 Inadvertent Start of Reactor Coolant Pump - additional differential pressures across the core from a second running pump might lift parts of the core, break free particulate matter and relocate pieces in other positions of the reactor coolant system. The additional 5 MWt of pump heat would create a heat imbalance. Immediate corrective action would be to secure the RC pump and stabilize primary system parameters. Removing power on standby pumps would reduce the potential for inadvertent startup; however, the immediate availability of the pumps is viewed as the more important consideration.

1.9 Inadvertent Start of Makeup Pump - this can pose a serious problem if: (1) the system is already water-solid, (2) the pump starts up without having valve MU-V17 control the flow to maintain the level in the pressurizer, and (3) there is no bypass operating so that full operational pump pressure is applied to the system. The maximum head attained by the pump would be applied to the reactor coolant system (RCS) under these conditions. When the system is not water-solid, the makeup pump adds water to the RCS, compressing and condensing the steam bubble in the pressurizer. Operator action time required to shutoff the pump after an inadvertent actuation may be insufficient for the operator to prevent RCS overpressurization

Safety valve action may not mitigate the overpressurization transient, particularly under water-solid conditions; therefore, a bypass flow line or test line should be utilized for each standby pump to prevent maximum HPI discharge pressure being applied to the primary system.

1.10. Inadvertent Opening of Pressurizer Spray Line - inadvertent pressurizer spray would cause a pressure decrease in the RCS. If the decrease is sufficient, RCP cavitation may occur, together with increased gas release from the coolant throughout the RCS.

Corrective action would be to immediately isolate the spray line by means of the spray block valve and then to stabilize primary system conditions. If the line cannot be isolated, venting of the pressurizer should be discontinued and additional pressurizer heaters turned on, if available, while observing pressurizer level and primary system pressure. If this action is insufficient to maintain the pressurizer pressure, it is recommended that the alternate RCP (the one without direct pressurizer spray) be used. If depressurization cannot be stopped, the RC pumps will have to be stopped before they cavitate and operation maintained by cooling in the natural circulation mode, by the method described in EP-4 or by means of the decay heat removal system.

1.11. Loss of Pressurizer Heaters - if all pressurizer heaters are lost, the plant may be unable to maintain system pressure. Under these conditions (unable to control system pressure), the plant should be operated in the natural circulation mode, or, if not available, the mode described in EP-4. The decay heat removal system may be employed if these other methods prove to be unsatisfactory for plant cooldown.

1.12. Boron Dilution - Potential return to criticality would exist by the addition of unborated water to the primary system. Periodic sampling of primary coolant boron concentration should verify shutdown margins. On-line SRMs/IRMs should be monitored to detect a loss of shutdown margin. All sources of unborated water should be carefully controlled administratively from the control room.

Immediate corrective action for any loss of shutdown margin would be to identify and secure the source of dilution and use borated makeup to restore margin.

1.13 Noise Diagnostics

Reduction of primary side temperature by steaming on OTSG "A" should not produce any sudden changes which can be monitored by the incore noise analyses system. Increasing coolant viscosity from the cooler temperatures would tend to produce larger differential pressures across restrictions, thereby increasing the potential for loosening parts. Virtually any anomalies heard with the noise analysis system would be cause for concern, and cooldown should allow time for feedback from the noise analysts. The detection of anomalies should be evaluated before proceeding in the cooldown.

To allow a more quantitative evaluation of the progress of the cooldown, a baseline "background" noise map must be available upon initiation of the cooldown. Periodic noise surveys thereafter could be compared to the baseline to provide a more positive trend indicator.

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1.14 Pressurizer Level Conditions

Cooldown of the primary system will cause shrinkage of the moderator volume. Since cooldown is expected to proceed slowly, pressurizer level changes would also be expected to occur slowly, and within the capability of the normal makeup system.

A [illegible] of all level indication during the cooldown evolution would remove a key piece of information necessary to this evolution. The immediate corrective action should be as indicated in EP-21. The predicted moderator volume shrinkage in going from 280^{o}F to 160^{o}F at a constant pressure of 1000 psia is approximately 540 ft^3. Since there is approximately 800 ft^3 of water in the pressurizer at the initial condition, loss of level in the pressurizer should not occur. Pressurizer sprays should be monitored to preclude activation which will cause a decrease in pressure.

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1.15 INSTRUMENTATION OVERVIEW

1. Primary System Instrumentation

Primary coolant loop instrumentation appears to be powered from the 120 volt vital (safety-related) buses. These buses are normally supplied from a dc bus through inverters. The dc bus is normally supplied from the auxiliary ac system by means of static rectifiers and is backed up by a battery floating on the dc bus. The plant computer is also powered from the vital buses. Assuming the loss of offsite power, the aforementioned instrumentation should continue to monitor the plant parameters without any interruption.

2. Pressurizer Valves

VALVE	POWER SUPPLY	CONTROL Sw	INDICATION
RC-R2 Relief Valve	Safety-Related	Auto/Open/Close	Power Available Only
RC-V1 Spray Valve	*	Hand Indicating Controller	Controller
RC-V2 Relief Block Valve	*	Open/Close	Open/Close
RC-V3 Spray Isolation Valve	*	Open/Close	Open/Close
RC-137 Vent Valve	?	Open/Close	Open/Close

* B&W says presently an effort to connect back-up power for these valves in case of loss of offsite power. Not done yet.

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3. Incore Thermocouple Readout

With regard to the incore T/C being read by the plant computer, we have determined that there is no low temperature reading cutout built-in the analog conversion program in the computer. The value of the incore temperatures will therefore be computed and printed for the low temperature range. The reactor coolant loop hot and cold leg temperature indicators (0°-800°F and 50°-650°F) could be used as substitutes.

4. Reactor Coolant Pumps Motors

The close and trip circuits for the reactor coolant pump motor breakers include permissive and trip interlocks.

The RCP motors will trip on a number of electrical signals and either of two cooling water signals.

The cooling water signals include low seal injection flow or low cooling water flow to the seal heat exchanger. Consideration should be given to bypassing (jumped out) these trip signals. B&W informed us on April 10, 1979 that these two trips have been bypassed since April 8, 1979. The electrical fault protection on the motor should be retained to protect the containment electrical penetration assemblies. Other electrical trips, such as under voltage, not associated with interrupting fault currents, could by bypassed.

With respect to the permissives to start (in case of loss of the running pump), there are a number of permissives including:

4. Reactor Coolant Pumps Motors (Continued)

   1. RCP oil lift pressure
   2. RCP low cooling water seal flow to heat exchanger
   3. RCP seal injection water flow
   4. Upper and lower oil pot level
   5. Neutron power level (start-up)
   6. Core lift (start-up)
   7. Motor heat exchanger cooling water

Number 2 and 3 could be bypassed as it was done in the trip circuits.

Numbers 5 and 6 involve start-up concerns. These could be bypassed with no loss of function.

To prevent inadvertent signals being present, 1, 4 and 7 could be bypassed. However, additional precautions should be taken to assure that oil and cooling water pumps associated with the RCPs are started.

B&W also informed us that there will not be RCP clearance problems due to operation at the lower temperatures.

1.15 INSTRUMENTATION OVERVIEW

1.16 (Additional secondary side failure modes & effects are being considered and will be forwarded when complete).

2.0 RADIOLYSIS

Consideration was given to the potential effects of radiolysis of the reactor coolant water, i.e., the effects of decomposition of the water to form hydrogen and oxygen.

In gamma and neutron fields, 17 ccH_2/kg water is needed to suppress radiolysis of the primary coolant (Ref: US Patent 2937981, 5/24/60). In operating plants the usual concentration of hydrogen is maintained between 25 cc/kg and 35 cc/kg.

In the TMI plant the saturation concentrations of hdyrogen are listed below:

Operating Point "A" (280°F, 1000 psia)	1670 cc/kg
Operating Point "B" (220°F, 1000 psia)	1540 cc/kg
Operating Point "C" (140°F, 1000 psia)	1430 cc/kg
Operating Point "D" (140°F, 100 psia)	140 cc/kg.

These values are considerably higher than the concentration of hydrogen required for the suppression of radiolysis of water at the operating conditions of the plant. It can be concluded therefore, that no radiolysis will take place at the operating conditions defined by points A, B, C and D as long as the primary coolant remains saturated

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with hydrogen. Some radiolysis way be expected if the concentration of hydrogen is reduced below the saturation limit and it reaches the value lower than 17 ccH_2/kg but we cannot see how this would occur. Therefore, we conclude that the decomposition of the water into hydrogen and oxygen during cooldown is not of concern.

3.0 CRITICALITY

The following discusses the need to increase the boron concentration in the reactor coolant system to prevent possible criticality.

Assuming the initial concentration of boron in the primary coolant system of 3000 ppm, precipitation of boric acid will not occur until the temperature of the primary coolant is reduced to below 32°F. Since there is no boil-off of coolant in the primary system the concentration of boric acid remains unchanged. It would be necessary to increase the concentration of boric acid by a factor of 24.5 at operating point "B" (220 F and 1000 psia) and by a factor of 8.5 at operating point "C" (140 F and 1000 psia) before any precipitation of boric acid could occur. It is concluded that no precipitation of boron in the primary system would occur when the operating conditions of the plant is changed from "A" to "B" and subsequently to "C".

In the letdown system the primary coolant is cooled from its initial operating temperature to about 120 F (FSAR value). Then it is depressurized to 14.7 psia. During the depressurization the liquid remains in a sub cooled condition and no boiling takes place. The concentration of boric acid in the letdown system stays the same as in the primary system and no precipitation is expected.

The reactor coolant is presently believed to contain 2200 ppm boron. The reactor core geometry if fully intact, at 150°F, 2200 ppm boron, all rods in and burnable poison intact, would have a K ~0.85. Rods are worth 10%~k; the burnable poison 4% Δk. Thus if rods are out, k goes to ~.95 and if the burnable poison is also out the k goes to ~.99 (at 2200 ppm).

If the clad is removed and borated water (2200 ppm) substituted (i.e., Zr to oxide and washed away) there would be little, if any, change in reactivity, since at this boron level and temperature the moderator reactivity coefficient is near zero)probably slightly positive). Similarly the reactivity state is not sensitive to the temperature in this range (150°F to 280°F).

Thus, if fuel has not redistributed, a boron level of 2200 ppm will keep the system subcritical even if everything but fuel is removed (small effect from thimbles and grids).

If fuel is redistributed (in addition to the above removals) the system could - under optimum conditions of moderation - go critical at 2200 ppm (note: moderation is required since solid UO2 spheres require enrichments over 5% to be critical). For the worst case - all fuel in a cylinder, or sphere - at optimum moderation - about 3000 ppm would be required to stay subcritical. The calculations are by the B&W Naval criticality group, using Monte Carlo (KENO) and tested cross sections. The calculation used pellet nuclear parameters (because these maximize reactivity) and fuel-water (boron) ratios which have been optimized by sensitivity studies.

The configurations which require boron levels above 2200 appear to be not very probable (unless it is probable that the rods are not there), but B&W believes (strongly) that there should be protection from redistribution criticality by going to 3000 ppm boron.

There are no problems with 3000 ppm from a physics viewpoint. The problems which have been expressed appear to be only the potential for systems blockage from boron at this level - a concern which appears to have no theoretical basis (see Section 1).

Detectability

If the reactor were to get to a k of about 0.95 subsequent changes of the order of 1%Δk should be reasonably detectable (25% increase in count rate for .95 to .96) on an excore startup detector if conditions are reasonably normal. It may, in these circumstances, be partially masked by (relatively) weak sources, disturbed geometries, and nonnuclear changes (e.g., downcomer density changes).

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At 2200 ppm the worst configuration is estimated to be several percent supercritical (i.e., k~1.03). It is very difficult (if not, impossible) to estimate the power level such a system would maintain to compensate for this reactivity. However, going to a water density of about 0.5 should (at least) do it. It is, of course, not generally possible to predict the (hypothetical) rate of reactivity addition to the case of redistribution (when detectable above 0.95) and thus provide the needed boron insertion rate.

Conclusions and Recommendations

At this time we do not have sufficient concern either to require boration or to prevent it. A primary coolant sample would likely provide information that would allow a more definitive recommenda... (e.g., the presence or non-presence of control rod material and a better value on existing boron concentration).

There is nothing inherent about the cooldown process which would in itself affect the reactivity. If boron blockage is not a problem, however, the boron level should be taken to 3000 psi to cover all (remote) possibilities. If decision to borate is made, a 100 ppm stepwise addition with monitoring of potential letdown blockage should be considered. If blockage is sufficiently strongly suspected to cause problems so that this is not carried out it would be highly advisable to be sure the startup range instrumentation is likely to be in good order (would be wise in any case) and it would also be advisable to be sure that boration at some "reasonable" rate is available.

4.0 ENGINEERING

We have studied the mechanical capability of the reactor system components for the conditions that would be experienced during cooldown. The results of this study are discussed as follows:

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4.1 FRACTURE MECHANICS

Fracture mechanics calculation have been performed for several cases that could be encountered in the planned cooldown of TMI-2. In all cases, the possible atypical weld metal in the lower head is limiting. Nevertheless, assuming reasonable mixing of the water, our calculations show that there is no need for concern about brittle fracture of the vessel unless extremely unlikely conditions would occur.

We first performed Appendix G calculations using all of the conservative Appendix G assumptions. These include a 1/4T flaw, the Appendix G bound K_{IR} curve, and a factor of 2 on pressure stresses. This gave a minimum temperature of 160^{o}F for 1000 PSIG pressure and a cooldown rate of 50^{o}F/hr.

Next, we calculated thermal stresses and stress intensity factors for the proposed cooling parameters. This gave a slightly higher cooldown rate, then slightly higher thermal stresses and stress intensity factors. Again, using the Appendix G factor of 2 on pressure, the K_{IR} curve, and the 1/4T flaw, the minimum temperature to comply with Appendix G was 170^{o}F.

If the pressure were reduced to about 900 PSIG, Appendix G requirements and margins would be met at 150^{o}F.

We also performed calculations asuuming a pressure increase to 2500 PSIG. Using K_{IC} instead of K_{IR}, with a factor of 2 on pressure stress, and a 1/4T flaw a temperature of 185°F would be required. With no factor of 2 margin on pressure, a temperature of 140°F is still tolerable.

Therefore, we conclude that there is a very low probability of vessel failure under conditions postulated to occur during the planned cooldown.

4.2 Solid Conditions in Steam Generator Secondary Piping

For water solid conditions on the steam generator secondary side, the piping systems affected, out to the first isolation valve, are the main steam line, main feedwater line, and the auxiliary feedwater line. The design of both feedwater lines is predicated upon being filled with water during operation and therefore, normal code allowable stresses will not be exceeded. While the main steam line is not filled with water normally, the additional dead weight contribution to the piping is accommodated within normal code limits for that portion inside of containment. The spring hangers (one on one main steam line and three on the other) will bottom out and act as rigid restraints. For the main steam piping in the auxiliary building, the spring hangers will be pinned so as to carry the additional dead weight load of the water in the piping within normal code limits. The information on the main steam lines is based upon verbal input from Burns and Roe, the architect engineer for TMI-2. At this point in time the architect engineer is re-evaluating the seismic capability of these lines; the results of this reevaluation are not yet available to the staff.

4.3 Steam Generator Tube Integrity

Steam Generator tubes are required to maintain their integrity during postulated design basis accidents including a LOCA or a steam line break in combination with an SSE. The design basis LOCA corresponds to a 925 psia secondary to primary pressure differential and the design basis MSLB corresponds to a 2200 psia primary to secondary differential pressure at approximately 600^{o}F. The required margins of safety against tube failure during postulated accidents are consistent with the margins of safety determined by the stress limits specified in NM-3225 of Section III of the ASME. Furthermore, a factor of safety of three against tube rupture is required during normal operating conditions which correspond to a 1250 psia primary to secondary pressure differential at approximately 600^{o}F. Babcock and Wilcox has provided results of laboratory tube burst and collapse tests. The burst tests conducted on specimens with defects up to 70% through wall resulted in no tube failures at pressures less than 3900 psi and the collapse tests on similarly defected tubes resulted in no tube failures below 3500 psi.

Three Mile Island Unit 2 conducted a baseline inspection of 100% of the tubes in both steam generators in November 1977 following the hot functional tests. Tubes with imperfections 40% or greater were plugged which is consistent with the basis delineated in Regulatory

Guide 1.121, to maintain the factors of safety described above and provide an additional margin for possible operational degradation.

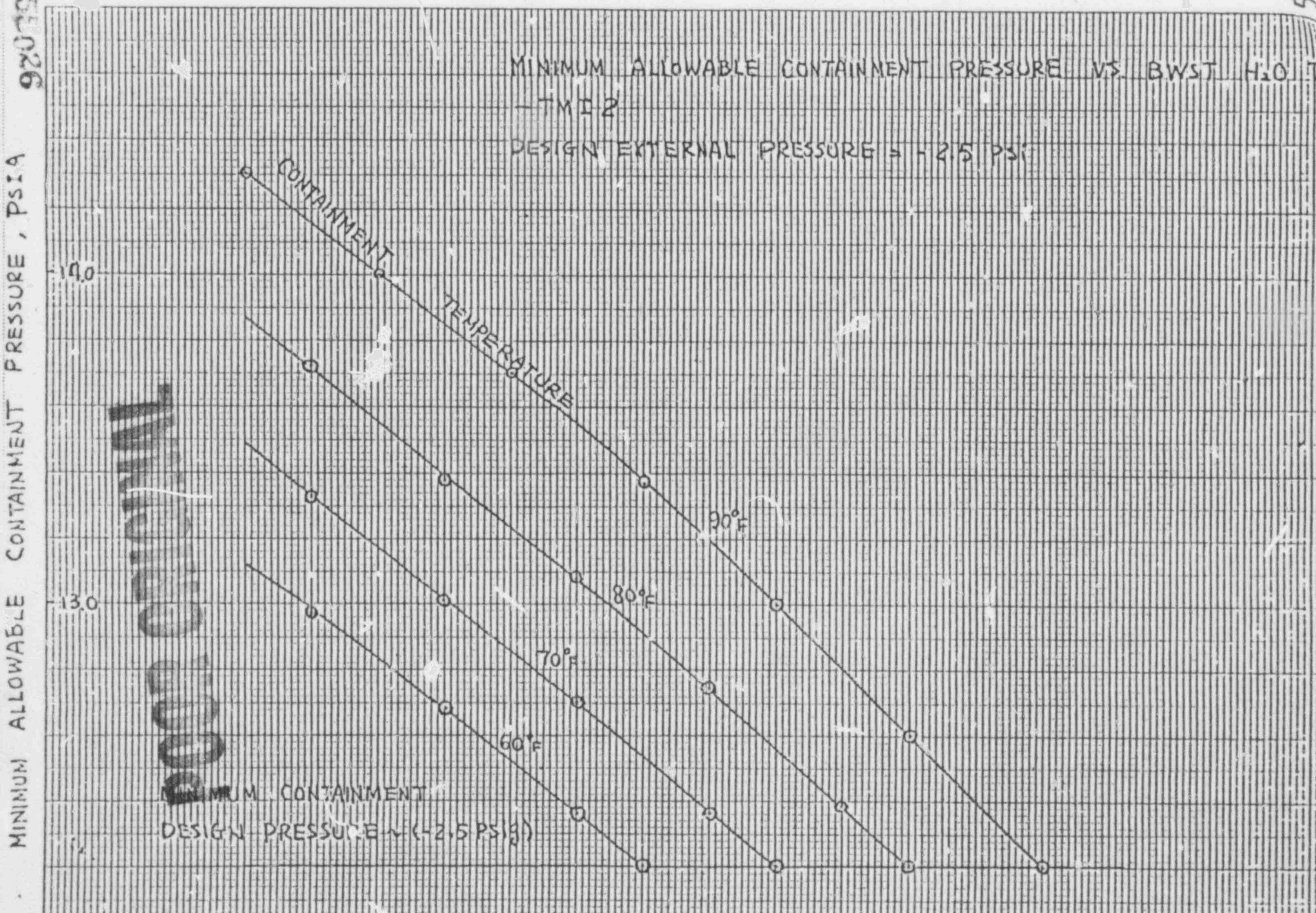
Based on the above design bases and the steam generator inspection and tube plugging which was conducted, the steam generator tubes will have conservative margins of safety against failure under the proposed condition of 1000 psi primary to secondary pressure differential at temperatures up to 600^{o}F.

Condenser Flooding

Potential safety concerns associated with flooding of the condenser were considered. Since condenser integrity is not normally included in our safety reviews, little information is available in HQ to determine the safety margins for static or dynamic flooding forces. If operation in a partially flooded condition is anticipated additional information as to the expected operating conditions, condenser design parameters and test results (e.g., hydro) is needed. Our contacts with Burns & Roe have not been successful in obtaining this information and further effort has been stopped pending feedback as to the potential operating modes in a flooded condition.

5.0 CONTAINMENT

The containment internal pressure has been slightly lower than ambient pressure for most of the time since the accident. At the present time, the containment is at approximately a 0.9 psi reverse pressure differential. Since the design pressure is 2.5 psi, the current pressure is not of immediate concern. Current operating procedures indicate that the water flow to the fan coolers should be terminated if the reverse pressure differential reaches 2.0 psi. This action would effectively terminate further cooldown of the containment atmosphere thereby terminating the transient. In any case, this would be a rather slow transient allowing sufficient time for proper action. We believe, however, a more severe transient should also be considered. This transient is the inadvertent operation of the containment sprays. Initiation of the sprays would result in rapid cooling of the containment atmosphere causing a corresponding rapid decrease in containment pressure. The magnitude of the pressure decrease will depend upon the inlet spray water temperature (BWST water temperature). To assure that the containment does not exceed the design reverse differential pressure of 2.5 psi, the containment parameters should be maintained above minimum values as shown in the enclosed figure. The figure indicates that for a given inlet spray water temperature, the containment temperature as well as containment pressure should be maintained above minimim values. The pressure could be controlled by the addition of a noncondensible gas such as nitrogen or dry air. This procedure has apparently been followed previously to decrease the reverse differential pressure to below 1 psi. Control of containment


MINIMUM ALLOWABLE CONTAINMENT PRESSURE VS. BWST H2O TE
— TMI-2
DESIGN EXTERNAL PRESSURE = -2.5 PSI
MINIMUM ALLOWABLE CONTAINMENT PRESSURE, PSIA
14.0
13.0
CONTAINMENT TEMPERATURE
90°F
80°F
70°F
60°F
MINIMUM CONTAINMENT
DESIGN PRESSURE ~ (-2.5 PSIG)
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temperature could also be achieved by terminating the water to the fan coolers. Since the fan operation would continue, proper mixing of the atmosphere would be maintained while eliminating the heat removal mechanism. Since the consequences of exceeding the reverse design pressure differential is unknown, we believe it prudent to maintain containment conditions as indicated above to allow for inadvertent spray operation

6.0 RADIOLOGICAL CONSIDERATIONS

The potential radiological consequences of loss of let down flow use of the RHR system, and steam generator leakage have been identified for consideration in this section.

6.1 Purification Demineralizer Heatup/Degradation

Substantial radioactivity may have built up on the purification demineralizer such that if the flow is stopped, the bed will heat up due to decay heat. Rough calculations indicate that the relief valve will lift and discharge small amounts of water and possibly traces of steam to the Reactor Coolant Holdup Bleed Tanks (RCHBT) if the system is isolated. As long as some flow is maintained, there should not be any steam. If water and traces of steam are relieved to the RCHBT, the offsite consequences should be nil because these tanks vent to the waste gas vent header which can be placed at a negative pressure by venting back to containment. Procedures should exist for venting the waste gas vent header back to containment should this become a problem.

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Heat in combination with radiation damage could result in degradation of the demineralizer resin. Radiation degradation which would lead to physical property changes should not occur within the next few weeks. If there has been more fuel degradation than the 0700 3/30 primary coolant sample indicated, it is possible that the resins could physically break down. This could lead to plugging of the demineralizer lower retention screens, thus blocking flow. It is our understanding that the valve operator for the inlet to the purification demineralizer has failed thus making easy realignment of letdown flow difficult. We recommend that procedures be considered for flow blockage in the purification system.

The radiation exposure for the demineralizer resins will also decrease their ability to ion exhange. It is expected that decreased ion exchange is now taking place and that radioactivity could leach off of the resins in the future. This should not be a significant concern because downstream components are heavily shielded; however, radiation levels could increase.

6.2 RHR System Contingency Plan

If it is necessary to use the RHR system, leakage and resultant iodine releases could be a problem. A method to minimize radioiodine releases would be to install a skid mounted charcoal filter system in the RHR room. Such units already exist and could fairly easily be lowered through the RHR pump room equipment hatch. This should be considered for installation prior to reactor systems operation which could lead to a likelihood of RHR system operation.

The design flow rate of air from the RHR pump rooms is only 350 SCFM. This is a small flow and a small charcoal filter system could be installed in the exhaust ducting if room exists. This would supplement the large Auxiliary Building Filter Units which may become degraded with time. A small fresh charcoal filter would reduce iodine releases by at least a factor of 100 if RHR had to be used.

6.3 Contingency Plan For A Steam Generator Leak

Consideration should be given for methods of detecting "A" steam generator leakage with a flooded secondary side condition. Procedures should exist for minimizing releases should leakage occur--e.g., use of condensate polishers on recirculation to the hotwell and maintaining the condenser at a pressure negative to the condenser circulating cooling water.