Westinghouse

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Date: March 5, 2010
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From: Systems and Equipment Engineering I
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Your ref:
1. NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors," September 13, 2004
2. NEI 04-07, Revision 0 "Pressurized Water Reactor Sump Performance Evaluation Methodology, Volume 1 – Pressurized Water Reactor Sump Performance Evaluation Methodology" December 2004
3. NEI 04-07, Revision 0 "Pressurized Water Reactor Sump Performance Evaluation Methodology, Volume 2 – Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02" December 2004
4. NRC Letter, Jonathan Rowley of NRR to Anthony Nowinowski of the PWR Owners Group Program Management Office, "Request for Additional Information Re: Pressurized Water Reactor Owners Group Bases For Licensee Debris Generation Assumptions for GSI-191," January 25, 2010. (ADAMS Accession Numbers: ML100050079, ML100050083, and ML100050086)

Our ref: LTR-SEE-I-10-33 Rev. 1

Subject: **Transmittal of Responses to NRC Request for Additional Information Re: Pressurized Water Reactor Owners Group Bases For Licensee Debris Generation Assumptions for GSI-191**

This letter was revised upon final review by PWROG personnel.

Generic Letter 2004-02 (Reference 1) requires all PWR licensees to evaluate the potential for debris blockage of screens used by the Emergency Core Cooling System (ECCS) for long-term recirculation of coolant to the reactor core post-accident. A uniform approach to perform that evaluation is described in NEI 04-07 (Reference 2) as amended by NRC's Safety Evaluation (SE) (Reference 3).

A fundamental step in performing the evaluation is the determination of the amount of debris generated by the postulated pipe break. The region about the break in which materials (insulation, coatings, etc.) are taken to become debris is called a Zone of Influence (ZOI). NEI 04-07 identified ZOIs for a number of commonly found insulation and coatings found inside PWR containments. The NRC SE on NEI 04-07 increased these ZOIs. Many licensees determined that these ZOIs were onerous.

Westinghouse performed a number of jet impingement tests at Wyle Labs for several licensees that were directed at reducing the ZOI for several materials, most notably, stainless steel jacketed NUKON and jacketed and banded Calcium Silicate. In all tests performed by Westinghouse, the same facility, procedure and approach to testing was used. The results obtained from these tests were used by about three quarters of the PWR licensees in the US to address GL 2004-02. NRC reviewed several of the technical reports (WCAPs) documenting the testing and reduced ZOIs and informally issued Westinghouse questions in twelve topical areas in December of 2008. Because of the common method of performing the testing and the wide-spread usage of the data, the PWR Owners Group developed a cafeteria program to address the NRC questions. On January 25, 2010 NRC issued a reformatted statement of their questions on the testing to the PWR Owners Group (Reference 4).

The non-proprietary responses to the reformatted questions in Reference 4 have been generated and are transmitted by this letter for submittal to NRC (Attachment 1). Proprietary references, along with appropriate affidavits of withholding, will be submitted under separate transmittal.

Questions regarding these responses may be directed to the author.

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ISSUE #1 Testing conditions did not adequately account for shock waves.

Issue #1A Statement:

Shock waves behave spherically (not like a jet).

- Significance: Equivalent ZOIs calculated by jet volumes during insulation testing is non-conservative for shock damage and may underestimate spherical ZOI diameter by more than a factor of 2.

Resolution To Issue #1A:

The accident conditions to be considered when assessing the possible damage to surrounding structures including piping insulation are those following the hypothetical rupture of a cold leg or a hot leg pipe. This would result in the depressurization (blowdown) of the high pressure, high temperature water coolant as determined by the two-phase, critical flow discharge of a flashing, steam-water mixture through the break. From Reference 1-1, the following fundamental characteristics have been identified for flashing steam-water mixtures:

1. Low quality, two-component, air-water mixtures have sonic velocities that are an order of magnitude less than the sonic velocities of either air or water.

2. Low quality, flashing steam-water mixtures, under critical flow conditions, have choking velocities (sonic velocities) that are approximately a factor of two less than the sonic velocities of equivalent two-component (air-water) mixtures.

3. Unlike single gaseous flows, over-expanded, supersonic flashing steam-water mixtures do not demonstrate the potential to generate standing shock waves in divergent nozzle geometries.

These fundamental characteristics lead to the following conclusions regarding the development of shock waves and blast waves in flashing steam-water mixtures.

1. The development of a blast wave requires a condition where the sonic velocities in the region following the initial expansion wave are greater than that in the surrounding air.

2. Because the two-phase mixture sonic velocities are an order of magnitude less than air alone, flashing steam-water mixtures cannot generate blast waves in air.

3. Damage mechanisms exist due to the expansion of the steam-water mixture as well as jet impingement. However, a blast wave is not one of these mechanisms. This is consistent with the observations from two-phase mixture compressibility experiments and blowdown jet measurements as well as steam explosion experiments and events.

Shock waves (overpressure) from a chemical detonation decay as a function of the distance from the explosion divided by the radius of the initial explosive mass (Sach's Scaling Law; see References 1-2 and 1-3). Hence, the L/D representation is consistent with the expected maximum overpressure (loading) from any significant compression wave. Therefore, the Wyle tests are representative of the reactor conditions based on the dimensionless distance L/D.

Regarding comparison of damage noted during the Wyle testing with that observed during air jet testing (Reference 1-4), the case has been made (Reference 1-5) that air, being a single-phase medium, is not a proper simulant for a flashing steam-water (two-phase) jet because air jets can form a blast wave in air. Hence, shock waves of the magnitude exhibited by rapidly-expanding single-phase jets are not formed in low quality, flashing two-phase steam-water jets. Moreover, since L/D has been demonstrated as the appropriate dimensionless distance for representing the depressurization of the expanding steam-water jet, the dimensionless distance to a target that is removed from the jet centerline is greater than the centerline distance and thus the impingement pressure is less as demonstrated by reduced damage/destruction in the Wyle insulation tests. The representation of a spherical expansion, combined with the observation of the maximum distance for damage along the centerline, captures this L/D aspect of the expansion and the local regions away from the centerline where the jet impingement pressure could cause significant damage.

<u>Issue #1B Statement:</u>

Testing was performed at Cold leg temps while Hot leg and PZR [pressurizer] temperatures are above the superheat limit of 577F for water.

• <u>Significance</u>: Vapor explosions are possible above 577F which could yield a large shock wave.

<u>Resolution to Issue #1B:</u>

As discussed in Reference 1-6, the role of homogeneous (spontaneous) nucleation is to make the response approach, or perhaps equal, the equilibrium jet expansion behavior calculated in the NUREG/CR-2913 (Reference 1-7). Damage from an expanding jet of low quality, flashing water is not caused by spontaneous nucleation of a fully depressurized water inventory. It is caused by the impingement forces developed by the jet and this directly related to the initial jet subcooling, i.e. the greater the subcooling, the greater the discharge flow rate and the higher the pressure profile along the jet centerline. Hence, the lower the water temperature is at a given pressure, the greater the destruction potential.

Issue #1C Statement:

The water temperature at the nozzle at test initiation was significantly lower than the water in the tank and resulted in a lower corresponding initial saturation pressure at the nozzle.

- Significance: Lower initial temperature minimized shock wave formation potential which likely resulted in less insulation damage.

Resolution to Issue #1C:

The lower initial water temperature results in a central jet core that remains intact for an extended distance as modeled in NUREG/CR-2913 (Reference 1-7). This is due to the subcooled nature of the axial jet that is discharged from the nozzle compared to the radial erosion rate of the jet as it flashes to the surrounding ambient pressure. Consequently, with the higher mixture density and discharge velocity for the subcooled jet, the dynamic head (and the damage potential) is maximized (along the centerline) due to these jet properties. Conversely, as discussed for other flashing experiments, the compression waves generated by a flashing liquid (Reference 1-3) are very small compared to the dynamic head of the jet.

Issue #1D Statement:

Rupture disks, like those used during testing, have a finite opening time which impacts shock wave formation.

- Significance: Acceptance that the test results included shock wave damage requires inherent acceptance by the staff of a finite break opening time for LBLOCAs, which are normally considered instantaneous.

Resolution to Issue #1D:

By their very nature, blowdown experiments will have an inherent break opening time characteristic. Consequently, the opening time will have a direct influence on the perceived issue (strength of the compression wave) that is to be addressed by the experiment. As a result, the issue of whether a significant compression wave would be formed in the analyzed condition for a LBLOCA is best addressed using analyses and experiments with ruptured glass spheres such as those performed by Boyer et al. (Reference 1-8), Esparza and Baker (Reference 1-9) and Esparza and Baker (Reference 1-10). These experiments show that the peak pressure waves measured when a glass sphere, filled with high temperature, high pressure liquid is ruptured is an order of magnitude less than when the same size glass spheres were filled with air and ruptured. This demonstrates that an expanding, flashing two-phase mixture does not develop a blast wave like that which can be formed by a high pressure, expanding single phase gas. In fact, the experimentally measured pressures for the flashing mixture tests were so small that they were at the limit that could be measured by the transducers. These experiments provide the most direct comparative demonstration of the unique behavior of one-component flashing two-phase mixtures.

Issue #1E Statement:

No method for scaling the shock wave from a test nozzle to a plant LBLOCA pipe size has been provided.

- Significance: Small nozzles used during testing underestimate the shock wave potential during a LBLOCA.

Resolution to Issue #1E:

As discussed by Epstein (Reference 1-3), the compression wave (shock wave) potential decays as a function of the pipe diameter to the radius traveled as presented by Sachs Scaling Law (Reference 1-2). This is the same scaling as the Wyle blowdown jet experiments that have been performed; i.e. the ratio of the distance to the target divided by the pipe diameter. Hence, the experiments are representative of the reactor accident condition.

References for Issue #1:

1-1 Fauske & Associates Report FAI/09-151, "Phenomena Contributing to Damage of Insulation During Blowdown and Jet Impingement," June, 2009.

1-2 Sachs, R. G., 1944, "The Dependence of Blast on Ambient Pressure and Temperature," BRL Report 466, Aberdeen Proving Ground, Maryland.

1-3 Henry, R. E., 2010, "Blowdown Compression Wave Overpressure", Fauske & Associates FAI LTR/10-01. (Proprietary)

1-4 Nuclear Regulatory Commission (NRC), 1998, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Bulletin 96-03, Boiling Water Reactor Owners Group Topical Report NEDO-32686, Utility Resolution Guidance for ECCS Suction Strainer Blockage, Docket No. PROJ0691, August 20, 1998.

1-5 Henry, R. E., 2010, "Comments on the Use of Supersonic Air Jets to Represent Flashing Two-Phase Flow," Fauske & Associates FAI LTR/10-03. (Proprietary)

1-6 Henry, R. E., 2010, "Relevance of the van den Berg Paper to the Two-Phase Blast Wave Discussion," Fauske & Associates FAI LTR/10-02. (Proprietary)

1-7 Weigand, G. G., Thompson, S. L., and Tomasko, D., 1983, "Two-Phase Jet Loads", NUREG/CR-2913, Sandia National Laboratories.

1-8 Boyer, D. W., Brode, H. L., Glass, I. I., and Hall, J. G., 1958, "Blast from a Pressurized Sphere," UTIA Report No. 48, Institute of Aerophysics, University of Toronto.

1-9 Esparza, E. D. and Baker, W. E., 1977, "Measurement of Blast Waves from Bursting Pressurized Frangible Spheres," NASA CR-2843, Southwest Research Institute, San Antonio, TX (May).

1-10 Esparza, E. D. and Baker, W. E., 1977, "Measurements of Blast Waves from Bursting Frangible Spheres Pressurized with Flash-Evaporating Vapor or Liquid," NASA CR-2811, Contract NSG-3008, National Aeronautics and Space Administration (November).

ISSUE #2: Adequate basis for applying test results to plant jacketing systems has not been submitted.

Significance: Test results underestimate LBLOCA insulation damage

A. Larger pipe sizes in the plant as compared to test configuration may result in different failure mechanisms.

B. Target size was too large/too close to test jet which resulted in less force on target edges than in the center as evidenced by center focused damage in test photos

C. Pipe jacketing failures are not the same as large component jacketing failures. Specifically, distance between band latches, number of latches, installation differences, band length, panel area under jet force, damage propagation by jet getting under large S/G panels)

D. Jacket tears due to lip lifting by the jet may not be the failure mode on large pipes. In addition, most plants use stainless steel jacketing which is stronger than tested aluminum, therefore lift forces may damage the banding and latches resulting in greater loss of jacketing.

E. Staff currently considers an open latch to be the same as a disengaged latch due to the random uncertainty in achieving this configuration. PWROG has not provided a basis for predicting repeatability for when a latch will open and disengage versus when a latch will open and remain engaged.

F. Potential for damage propagation outside tested ZOI on large components (i.e. getting under large S/G panels)

G. Axial jet impingement on insulation may be worse than perpendicular, especially for damage propagation along the pipe

ISSUE #4: Large uncertainties exist using ANSI model to calculate insulation loads

Significance: Use of ANSI model may yield non-conservative results.

A. Describe the procedure used to calculate isobar volumes used in determining equivalent spherical ZOI radii using the ANSI/ANS-58.2-1988 standard.

B. Explain why the WCAP-16710 analysis was based on 530F rather than the initial test temperature of 550F. Include an explanation of how the initial temperature differences between rupture disk water and tank water were incorporated into the jet sub cooling analysis.

C. Explain assumptions on how mass flow rate was determined considering potential for two-phase flow and temperature dependant water and vapor densities.

Resolution to Issues #2 and #4:

Specific aspects of the sub-points in Issues 2 and 4 have been discussed with NRC staff since the time that the NRC's issues were initially formulated. The following discussion will address Issues 2 and 4 in broad terms in lieu of explicitly responding to each sub-point of those issues, with the following exceptions that are addressed separately:

Issue #2C - Clarification:

In discussion with NRC staff regarding large non-piping components, it has been noted that all pressurizers have support skirts at their bottom ends that will physically block any jet associated with a surge line break from impacting insulation on the pressurizer itself. Thus, a licensee can refer to their own drawings showing the presence of the support skirt and thus can exclude pressurizer insulation as a debris source associated with surge line breaks.

Issue #2E - Resolution:

WCAP-16710-P reports jet impingement test results for stainless steel jacketed and latched NUKON targets at distances of 90", 124.8" and 174" from the jet nozzle outlet (presented in that topical report as corresponding to ZOI = 8D, 10D, and 13D, respectively, for a 3.54" nozzle diameter). The results of those tests are consistent in that the jacketing was not removed from the NUKON insulation, irrespective of which latches remained engaged or disengaged. Considered as a whole, this consistent behavior is considered a repeatable result.

Discussion of Remaining Items on Issues #2 and #4

Introduction:

Free-jet expansion tests were performed at Wyle Labs in January, 2010 to support developing a response to these two Issues. Pressure measurements were taken at specified planes perpendicular to the jet centerline using an instrument rake (Figures 2-1 and 2-2). The instrument rake provided for two sets of stagnation pressure measurements to be taken; five (5) on the "A" rake (closer to the nozzle exit) and four (4) on the "B" rake. The "B" rake was 12" farther from the nozzle than the "A" rake.

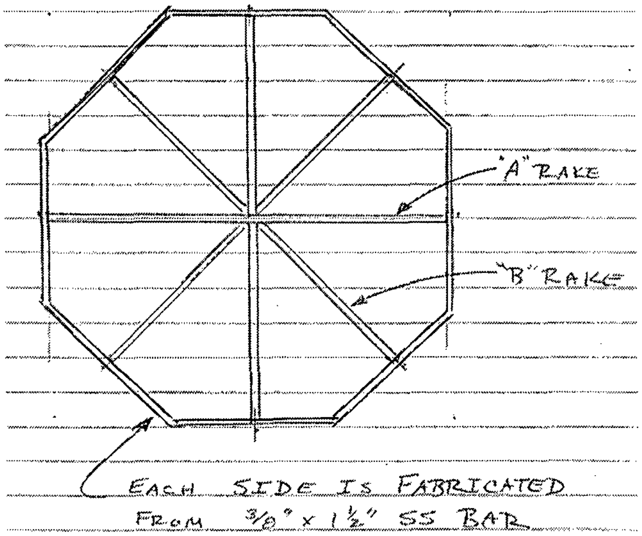

Figure 2-1
Front View of Instrument Rake

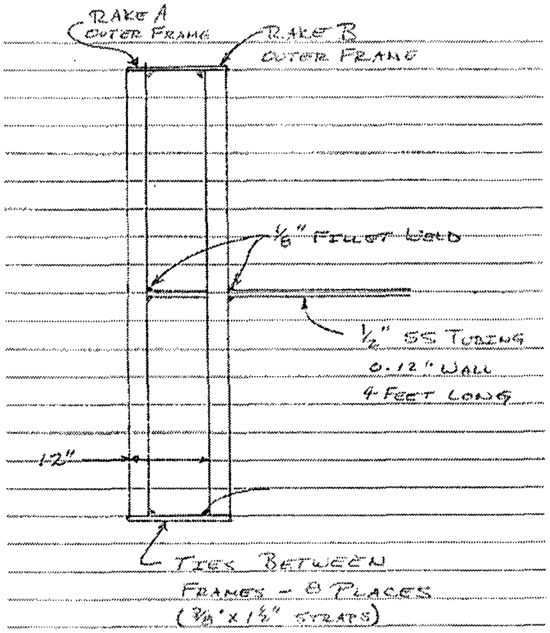

Figure 2-2
Side View of Instrument Rake

The instrumented tests provide sufficient data to both;

1. Support reducing uncertainties in the use of the ANSI/ANS 58.2-1988 jet expansion model to the insulation systems and materials tested using the Wyle High Flow Test Facility, and,
2. Justify application of the test data to plant jacketing systems for pipes larger than those tested.

A description of the application of the free-jet expansion test data, accomplishing the two points identified above, follows.

Objective:

The objective of this discussion is to demonstrate the use of test data obtained from jet impingement tests (observations of "no damage") and instrumented free jet expansion tests (stagnation pressures) performed using Wyle Laboratory's High Flow Test Facility to calculate zones of influence (ZOIs) for PWR materials inside containment.

Process Chart:

The steps to calculate Zones of Influence for PWR materials inside containment are outlined below.

Perform a Successful Jet Impingement Test at an L/D from Jet Nozzle	→	Perform an Instrumented Test at Same L/D from Jet Nozzle	→	Calculate a Zone of Influence Using both L/D and Measured Stagnation Pressure at Jet Centerline

Process Steps:

1. Perform a successful jet impingement test at an L/D from the jet nozzle;

   L = Distance of target from jet nozzle.

   D = Effective jet nozzle diameter.

   1.1 The L/D at which success is obtained is target (material) dependent; i.e., different insulation systems (targets) are expected to show success at differing L/Ds.

   1.2 The criteria for a "successful" test shall be considered as follows:

   - For jacketed Nukon, a test shall be considered successful if the jacketing material is not removed from the test item
   - For jacketed Cal-Sil, a test shall be considered successful if the jacketing material is not torn between the bands

- For unjacketed Nukon, a test shall be considered successful if there is no tearing of the outer cover of the insulation blanket.

2. Perform an Instrumented Test at Same L/D from jet nozzle as Step 1.
   2.1 Use instrument rake.
   2.2 Measure stagnation pressure distribution in jet field, including jet centerline pressure.
   2.3 Generate "map" of stagnation pressures.
      2.3.1 As a function of L/D (front rake and back rake), and,
      2.3.2 At the radii, R, for the two L/D values from the instrument rake.
      2.3.3 The radii can also be expressed as a function of the ratio R/D.

3. Calculate a Zone of Influence

Methods of Calculating ZOI:

Two methods of calculating a Zone of Influence have been identified and are described below.

<u>ZOI Calculation Method #1</u>

It is noted that, to use the following method for calculating ZOIs, NRC staff has suggested that further supporting information (including more test data) would be required. This is currently being evaluated for a possible future submittal.

1. Use the L/D and measured jet centerline stagnation pressure from a jet impingement test.
   1.1 Test data is being used to calculate the ZOI for "no damage"
   1.2 The test provides for actual distribution of stagnation pressures at the target plane (i.e., the plane perpendicular to the jet centerline at the point where the centerline of the jet contacts the target).
   1.3 Therefore, the test data directly reflects the actual radial stagnation pressure distribution about the jet centerline.
   1.4 The use of test data in this manner eliminates the concern that the ANSI/ANS 58.2-1988 model under-predicts the stagnation pressures at a plane perpendicular to the jet nozzle; i.e., data from the Wyle January 2010 instrumented free-jet expansion tests show that measured stagnation pressures at discrete radial locations in a plane about the jet centerline are lower than those predicted with the ANSI/ANS 58.2-1988 model.
2. Use the method of calculating isobars shapes from the ANSI/ANS 58.2-1988 model.

3. Calculate volume contained within the isobar by rotating the isobar about the jet centerline.
4. Use the method that was utilized for calculating Zones of Influence of materials reported in NEI 04-07 to calculate an amended Zone of Influence based on test data. In essence, double the volume found in step 3.3 to account for a double-ended guillotine pipe break, set the resulting volume equal to the volume of a sphere and solve for the radius.

Verification of the efficacy of this method will require further jet impingement testing to establish target destruction distances and stagnation pressures at those distances.

ZOI Calculation Method 2:

Method #2 holds the position that ZOI adjustments should be made as follows:

1. The measured stagnation pressure at a jet centerline location where a test is "successful" must be used as an input the ANSI/ANS 58.2-1988 to calculate an isobar that will then be used to calculate a ZOI.
2. If the plant pipe is larger in diameter than the pipe used in the test, the damage pressure is geometrically scaled (i.e., reduced) to the larger pipe size based on projected area so as to maintain the same force on the pipe and insulation system. This scaling rationale was described in Appendix B of NRC's Safety Evaluation (SE) for the BWR Owners Group Utility Response Guideline (URG).
3. If the plant pipe is equal to or smaller in diameter than the pipe used in the test, no scaling based on projected area is needed.

<u>ISSUE #3</u> Staff does not accept statement that unjacketed Nukon was not damaged at 5D because this insulation was jacketed at the beginning of the test.

<u>Resolution to Issue #3:</u>

For any zone of influence testing done, the criteria for a "successful" test shall be considered as follows:

- For jacketed Nukon, a test shall be considered successful if the jacketing is not removed from the test item; i.e., the jacketing remains in place (latches may be sprung but the jacket remains in place).
- For jacketed CalSil, a test shall be considered successful if the jacketing material is not torn between the bands.
- For unjacketed Nukon, a test shall be considered successful if there is no tearing of the outer cover of the insulation blanket.

With this updated success criterion, the 5D and 6D tests on jacketed Nukon are not considered "successful".

Once "successful" test results are obtained, the methods discussed in the response to Issues 2 and 4 for application of the ANSI/ANS 58.2-1988 model and, for jacketed targets, the scaling method outlined in the response to Issues 2 and 4, must still be considered when calculating final ZOI values.

ISSUE #5

Issue #5 Statement:

Provide a detailed description of the test apparatus specifically including the piping from the pressurized test tank to the exit nozzle including the rupture disk system.

Resolution To Issue #5:

The tests were performed at the Hot Water Blowdown Test Facility at Wyle Laboratories (Huntsville, Alabama). A schematic is given in Figure 5-1.

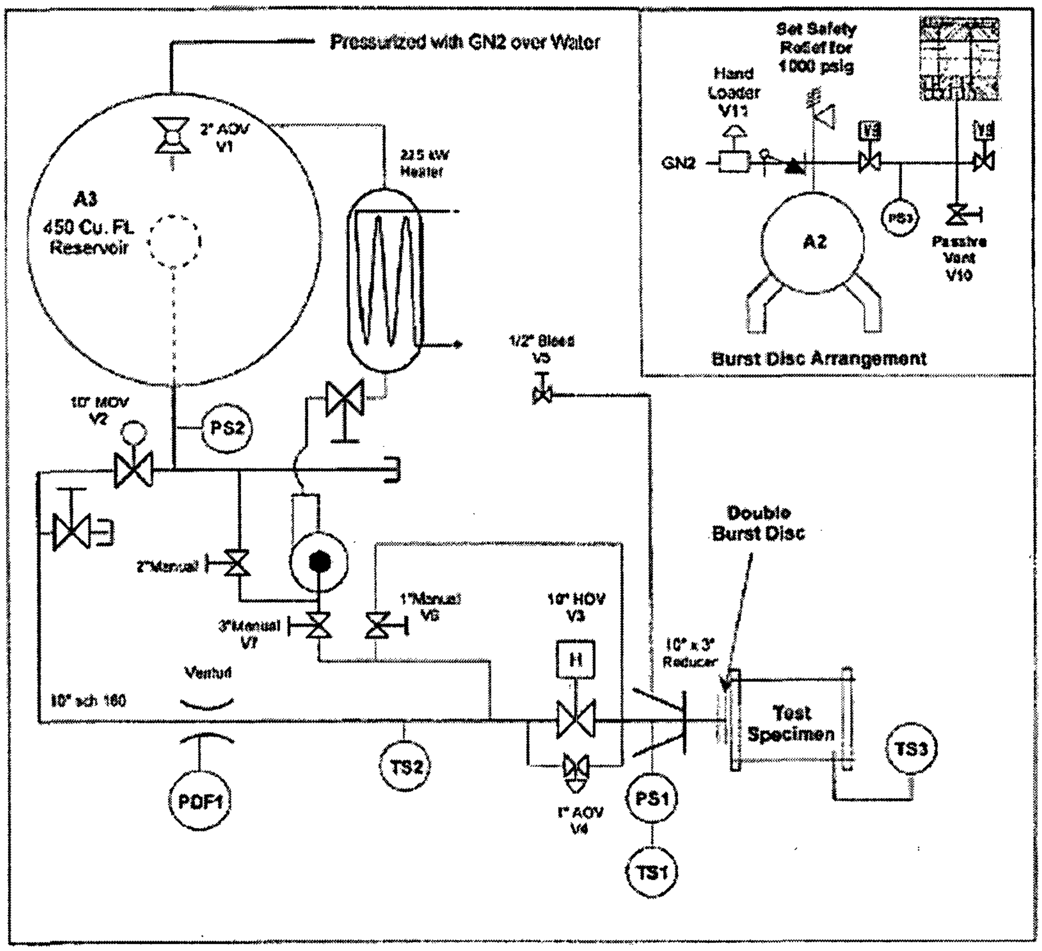

Figure 5-1
Schematic for Wyle Hot Water Blowdown Test Facility

Test loop instrumentation used in the jet impingement testing performed licensees is described in the table below:

Channel	Description	Range	Accuracy
PS1	Status pressure upstream of nozzle exit	0-5000 psig	+/- 0.3% F.S.
PS2	Tank (pressure vessel) pressure	0-3000 psig	+/- 0.3% F.S.
TS1	Fluid temperature upstream of nozzle exit	32-600°F	+/- 4°F
TS2	Fluid temperature near venturi	32-600°F	+/- 4°F
TS3	Ambient temperature near test specimens	32-600°F	+/- 4°F
PDF1	Venturi differential pressure for flow measurement	25 psid	+/- 0.5% F.S.

To support the development of a response to this request, drawings were requested from the test performer, Wyle Labs. In response to this request, Wyle Labs provided three drawings identified as Reference 5-1 through and including Reference 5-3. The drawing identified as Reference 5-1 is proprietary to Wyle Labs and will be provided to NRC by Wyle Labs with the appropriate non-disclosure affidavit. References 5-2 and 5-3 are included herein as Figures 5-2 and 5-3, respectively.

These drawings provide additional detailed information requested by NRC regarding the facility design, including piping from the pressurized test tank to the exit nozzle including the rupture disk system, beyond that given in the schematic diagram found in Appendix A of WCAP-16710-P regarding the layout and dimensions of the test facility.

From the drawings, the following is noted:

- There is a choke point upstream of the 3.54" nozzle
- The minimum dimension of the reducer feeding into the burst disk assembly is 2.313"

The minimum dimension controls the choked flow rate for a subcooled jet.

References

5-1 Drawing D09044, Sheet 3 of 4, "Hot Water Loop Plan & Elevations" (Proprietary)

5-2 Drawing D09044, Sheet 4 of 4, Jet Impingement Test Details – 3" Test Set-Up"

5-3 Drawing D06042, "Sheet 2 of 2, "Jet Impingement Test Layout and Details Test Stand"

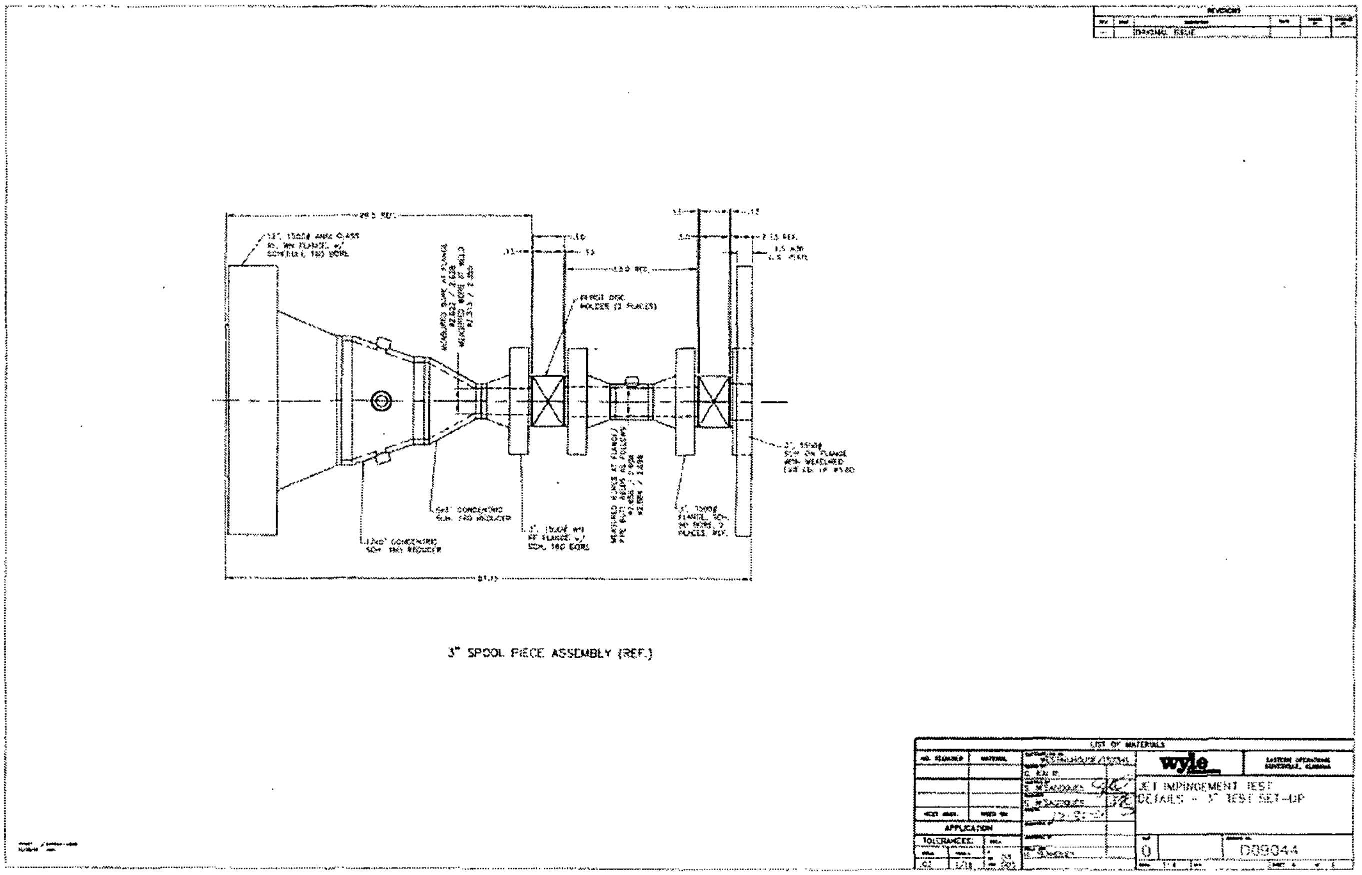
3" SPOOL PIECE ASSEMBLY (REF.)
LIST OF MATERIALS
APPLICATION
TOLERANCES:
wyle
JET IMPINGEMENT TEST
DETAILS - 3" TEST SET-UP
0
D09044

Figure 5-2

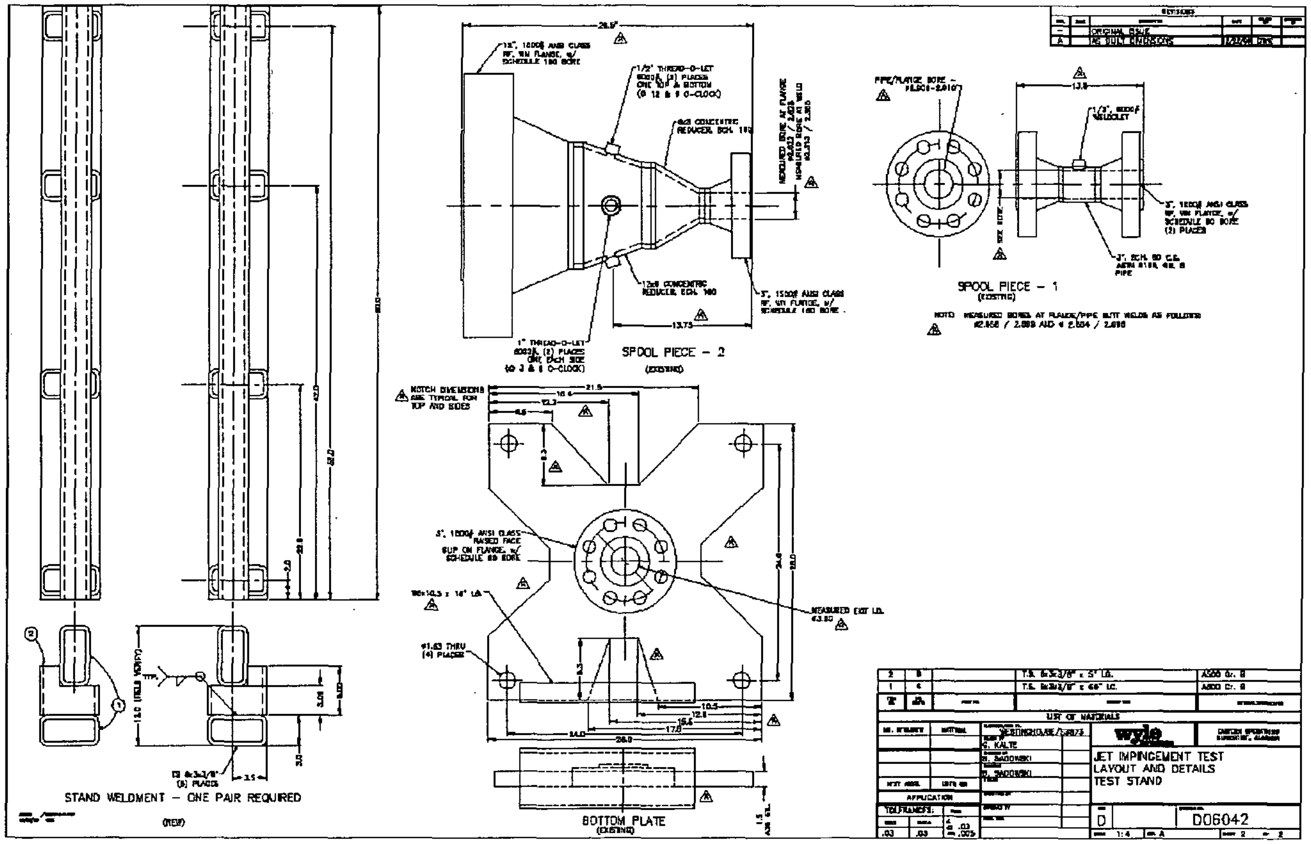
SPOOL PIECE - 2
SPOOL PIECE - 1
STAND WELDMENT - ONE PAIR REQUIRED
BOTTOM PLATE
JET IMPINGEMENT TEST
LAYOUT AND DETAILS
TEST STAND
D06042

Figure 5-3

ISSUE #6 Staff does not accept PWROG position that damage observed during testing is not likely in the plant.

Resolution to Issue #6:

For clarification, this issue deals with damage noted in jet impingement tests of jacketed and clasped NUKON insulation at ZOI = "5D" and "6D" performed using the Wyle High Flow test facility as documented in Revisions 0 and 1 of WCAP-16710-P and the reasoning put forward in that topical report that attempted to justify acceptance of those test results based on the non-prototypical nature of the test specimens. In the time that has elapsed since this issue was first put forth, the definition of a "successful" test has been reformulated, as given in the response to Issue #3 and restated herein:

> "For jacketed Nukon, a test shall be considered successful if the jacketing material is not removed from the test item."

With this updated success criterion, the 5D and 6D tests on unjacketed Nukon are not considered "successful".

ISSUE #7 Staff does not accept PWROG position that jet-ejected insulation panels during a LBLOCA would not be damaged due to subsequent collisions in the plant. Staff position is that containment conditions are significantly more crowded than the test conditions. Therefore more damage would occur to ejected insulation in the plant than the minor damage observed during the PWROG testing.

Resolution to Issue #7:

It has been previously discussed with NRC that the test results apply only to encapsulated Min-K panels deployed at the Callaway and Wolf Creek Nuclear Plants. As such, the treatment of this subject in Revisions 0 and 1 of WCAP-16710-P is specific to those two plants and only those two plants.

The insulation panels in question cannot be impacted by the jet based upon its location in the plant. Thus, there is no need to credit the test result of ejected insulation that survives.

The detailed justification for this response has been discussed with NRC staff and is presented herein for completeness.

INTRODUCTION

Insulation is used to mitigate heat loss from a pressurized water reactor (PWR) during power operations. There were three (3) areas of reactor vessel insulation that were of specific interest to the Callaway and Wolf Creek Nuclear Plants:

- Detector Well Panel (ex-vessel instrument tubes)
- Loop Piping Penetration Panel (adjacent to the hot and cold leg nozzles as it passes through the bio-shield)
- Reactor Pressure Vessel Top Head Panel (on top of the reactor head)

The insulation system used at the Callaway and Wolf Creek nuclear plants for these areas is encapsulated Min-K. The insulation material is Min-K and the encapsulation material is stainless steel sheeting that has been welded at the seams.

The encapsulated Min-K thermal insulation included in the test program conservatively represented the as-installed configurations at the Wolf Creek and Callaway nuclear plants. Three encapsulated Min-K panels were tested, one for each of the three areas of interest identified above. As discussed below, their test configuration was selected to conservatively represent their installation in the plant. The Reactor Pressure Vessel Top Head Panel (RPVTHP) was the item ejected from the test stand as a result of jet impingement testing.

The design of the Callaway and Wolf Creek Nuclear Plants precludes jet impingent on the RPVTHP from a postulated large break LOCA; therefore, conducting jet impingement testing of the RPVTHP was overly conservative. That is, the RPVTHP jet impingement test was conducted to evaluate a scenario that could not happen at either the Callaway or Wolf Creek plant. Without the action of a jet, the RPVTHP will not be ejected. This is supported in the following evaluation.

Only the Callaway and Wolf Creek Nuclear Plants have taken credit for the Min-K jet impingement testing. The response to this question will include a discussion of applicable design features for those two plants. Other licensees choosing to take credit for this data will necessarily need to demonstrate the applicability of these test results to their plant design.

DESCRIPTION OF MIN-K INSULATION – REACTOR PRESSURE VESSEL TOP HEAD PANEL

The following text describes the ejected panel and its test configuration. This panel was 19 inches in height and had a slight curve to it. The inside surface of the panel was approximately 56.5 inches wide. The outside surface of the panel was approximately 57 inches wide. Figure 7-1 shows the reactor pressure vessel top head panel as it was received at the test facility. What appear to be "buckles" in the stainless steel encapsulation in Figure 7-1 is not damage due to shipping, handling or testing; rather it is normal buckling of the panel due to the manufacturing process.

Figure 7-2 shows a photograph of the test configuration of the RPVTHP in the test rig. The RPVTHP was mounted slightly below the centerline of the jet nozzle. A flat plate was installed at the leading edge of the RPVTHP to conservatively simulate the mounting of the RPVTHP on the flange of the reactor vessel top head. The permanent cavity seal ledge, extending beyond the flange associated with the reactor vessel, was not simulated in the test.

MAXIMUM BREAK SIZE AT HOT- AND COLD-LEG NOZZLES

The calculation method for determining the maximum break size at the RV nozzle for the Callaway and Wolf Creek Nuclear Plants is described in this section.

A review of the physical construction of the piping and reactor support systems was conducted to assess the potential for a double-ended guillotine break at the reactor vessel nozzle welds. This was done using plant-specific drawings for the Callaway and Wolf Creek Nuclear Plants as identified in the text below.

As the hot- and cold-legs pass through the bio-shield, the piping is held by a whip restraint as shown in the schematic diagram given in Figure 7-3. This whip restraint is commonly referred to as a "wagon wheel." This restraint limits lateral movement of the piping should a break occur at the nozzle welds.

From Figure 7-3, the maximum possible lateral movement of the pipe in the "wagon wheel' is noted as:

$$displacement_{COLD\ LEG} = 2.09375\ inches$$

$$displacement_{HOT\ LEG} = 2.3125\ inches$$

The thickness of the cold-leg and hot leg wall pipe wall, t, for the Callaway and Wolf Creek Nuclear Plants is $t_{COLD\ LEG} = 2.32\ inches$ and $t_{HOT\ LEG} = 2.45\ inches$, respectively. For a postulated pipe break, the reactor vessel restraints restrict the lateral movement of the reactor vessel.

The reactor coolant pump (RCP) tie rods preclude the movement of the RCP due to a postulated cold leg break. These tie rods also preclude the possibility of cold leg separation from the reactor vessel cold leg nozzles for such breaks. The design of the RCP tie rods is shown in Drawing 1459F07, Figure 7-4. An enlargement of the tie rod design that has been taken from Drawing 1459F07 is shown in Figure 7-5.

The steam generator (SG) lower lateral supports preclude the movement of the SG due to a postulated hot leg break. These lateral supports also preclude the possibility of hot leg separation from the reactor vessel hot leg nozzles for such breaks. The design of the SG lower lateral supports is shown in Drawing 1459F04, Figure 7-6. An enlargement of the design of the SG lower lateral supports that has been taken from Drawing 1459F04 is shown in Figure 7-7.

CONTROL ROD DRIVE MECHANISM (CRDM) EJECTION

In discussions, NRC representatives questioned if the ejection of a control rod drive mechanism (CRDM) could result in a jet that could damage the RPVTHP. Again, the design of the plant precluded this. The CRDM housings extend above the Reactor Pressure Vessel Top Head insulation. If a CRDM were to occur, the fluid issuing from the housing would be expelled above the Reactor Pressure Vessel Top Head insulation. Therefore, the ejection of a CRDM would not cause damage due to jet impingement loads on the Reactor Pressure Vessel Top Head insulation.

CONCLUSION

Thus, by design, only lateral movement allowed by the gap in the "wagon wheel" pipe restraint is possible. If the piping moves the maximum allowed by the "wagon wheel" pipe whip restraint, a gap equal to the maximum gap between the pipe and wagon wheel will open. Based on the displacement values given immediately above and the cold- and hot-leg pipe wall thickness, no separation between the hot- or cold-leg nozzles and the RCS piping would be allowed for a break at the nozzles. Therefore, in the plant, there would be no jet flow if the weld at either the cold-leg or the hot-leg were to fail. Without jet flow, there would be no ejection of the panel as was observed in the test, which conservatively assumed a pipe separation with a uniform 1/8-inch gap offset about the 360° circumference at the nozzle weld.

Also, considering the CRDM housing extends above the Reactor Pressure Vessel Top Head insulation, the ejection of a CRDM will not result in jet loading on Reactor Pressure Vessel Top Head insulation that would result in damage.

OTHER CONSIDERATIONS

As observed from Figure 7-2, the test configuration provides for the jet nozzle to impinge on the RPVTHP at the leading edge of the panel itself. The action of a directed jet at both the leading edge of the panel, and along the length of the panel, subject the panel to conservatively large jet drag forces associated with the two-phase flow of the jet. Both the Callaway and Wolf Creek Nuclear Plants have at least three design features associated with the reactor pressure vessel preclude this from occurring.

Reactor Vessel Flange

For all PWR power reactor designs, the reactor vessel (RV) outside diameter (OD) increases at the location where the vessel and the top head are joined. For the Callaway and Wolf Creek Nuclear Plants, the OD increases from 193.00 inches to 205.00 inches as shown in Figure 7-8 This added thickness of the reactor vessel is needed to accommodate the studs that fasten the top head to the reactor vessel. This added thickness is (205.00 – 193.00)/2 = 6 inches. As can also be observed from Figure 7-8, this change in OD is accomplished with a sloped, gradual increase in the diameter of the RV. If there were a jet issuing from a postulated break, this sloped, gradual increase in diameter deflects the jet away from the OD of the RV flange. For conservatism, the test modeled the flange as a flat plate and did not model this sloped increase in OD of the flange.

Ledge for Permanent Cavity Seal

Both Callaway and Wolf Creek Nuclear Plants employ a permanent cavity seal. The reactor vessel flange has a ledge that runs circumferentially about the flange and extends outward 6 inches. This ledge serves as a seat for the inside diameter of the permanent cavity seal. This ledge provides for additional protection against directed flow coming from a postulated break at nozzles in the reactor cavity region.

Nozzle Safe End Location Offset Relative to RPVTHP

The nozzle/loop piping break location is offset from the RPVTHP. The nozzle safe end extends beyond the edge of the flange of PWR. This offset, coupled with the limited displacement of the postulated break, preclude a directed jet from impinging upon the RPVTHP.

For Callaway and Wolf Creek Nuclear Plants, the RV nozzle safe ends for a hot- and cold-leg, referenced to the RV centerline, are located at radii as follows:

$R_{HOT\ LEG}$ = 123.06 inches

$R_{COLD\ LEG}$ = 131.22 inches

The maximum radius of the top flange for this reactor design is:

$R_{RV\ FLANGE}$ = 102.5 inches

From these dimensions, for the Callaway and Wolf Creek Nuclear Plants, the hot- and cold-leg nozzles extend from about 20.56 inches to 28.72 inches beyond the edge of the RV/top head flange. This offset, coupled with the lack of separation of the RV nozzle and piping, the geometry of the reactor flange and the ledge for the permanent cavity seal, precludes a directed jet flow past the RPVTHP. (Note: The test conservatively modeled this as a 1.78 inch offset from the directed jet centerline and the top of the RPVTHP. This slight offset in the test provided for conservative jet loading on the RPVTHP that would not occur in the plant.).

SUMMARY

The jet impingement test of the RPVTHP for the Callaway and Wolf Creek Nuclear Plants provided for a directed jet to impinge upon the RPVTHP in the test rig. The directed jet generated sufficient force on the RPVTHP in the test stand that the RPVTHP was ejected about 150 feet from the test stand.

Plant-specific design features of the Callaway and Wolf Creek Nuclear Plants preclude this from occurring in the plant for the following reasons:

1. The primary loop penetrations through the bioshield limit the displacement of primary loop piping to less that the thickness of the cold-leg and hot-leg piping, precluding the separation of the piping and nozzle and therefore preclude the generation of a jet. Thus, <u>a similar occurrence in the plant cannot occur</u>.

2. Since jet loading will not occur on the RPVTHP, the RPVTHP will not be extracted from its location and <u>will not impact any object inside containment</u>.

3. Since no impact would occur, <u>the test conservatively represents the most severe jet loading that the RPVTHP would experience, it also presents the bounding damage to the RPVTHP for the Callaway and Wolf Creek Nuclear Plants</u>.

4. In the Callaway and Wolf Creek Nuclear Plants, <u>since the RV nozzle and piping cannot separate, there is no ZOI for such breaks to affect the RPVTHP.</u>

5. Again, for the Callaway and Wolf Creek Nuclear Plants, since the RV nozzle and piping cannot separate, there is no jet loading. Without jet loading, <u>there is no mechanism to fatigue encapsulating material to the point of failure</u> and cause the release of insulating material.

In summary, the damage to the RPVTHP resulting from the test represents a bounding condition for the RPVTHP for Callaway and Wolf Creek Nuclear Plants.

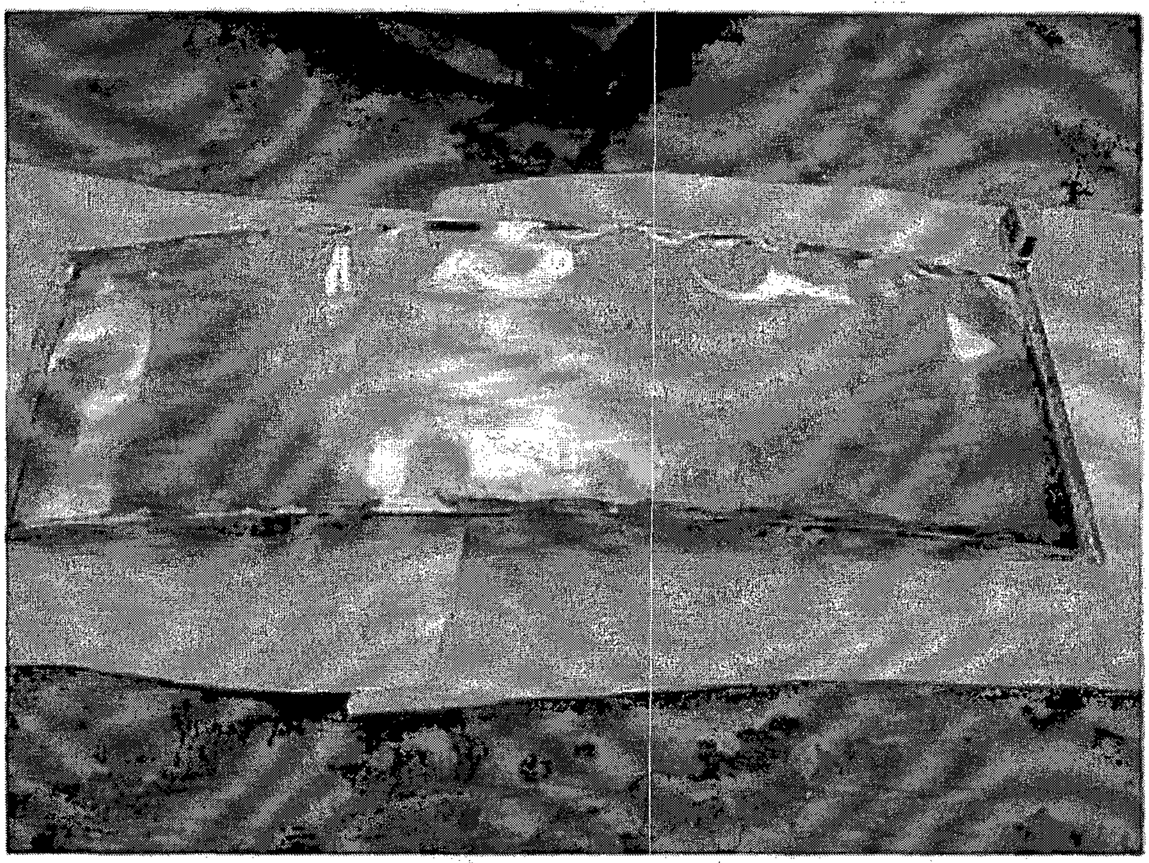

Figure 7-1
Reactor Pressure Vessel Top Head Panel

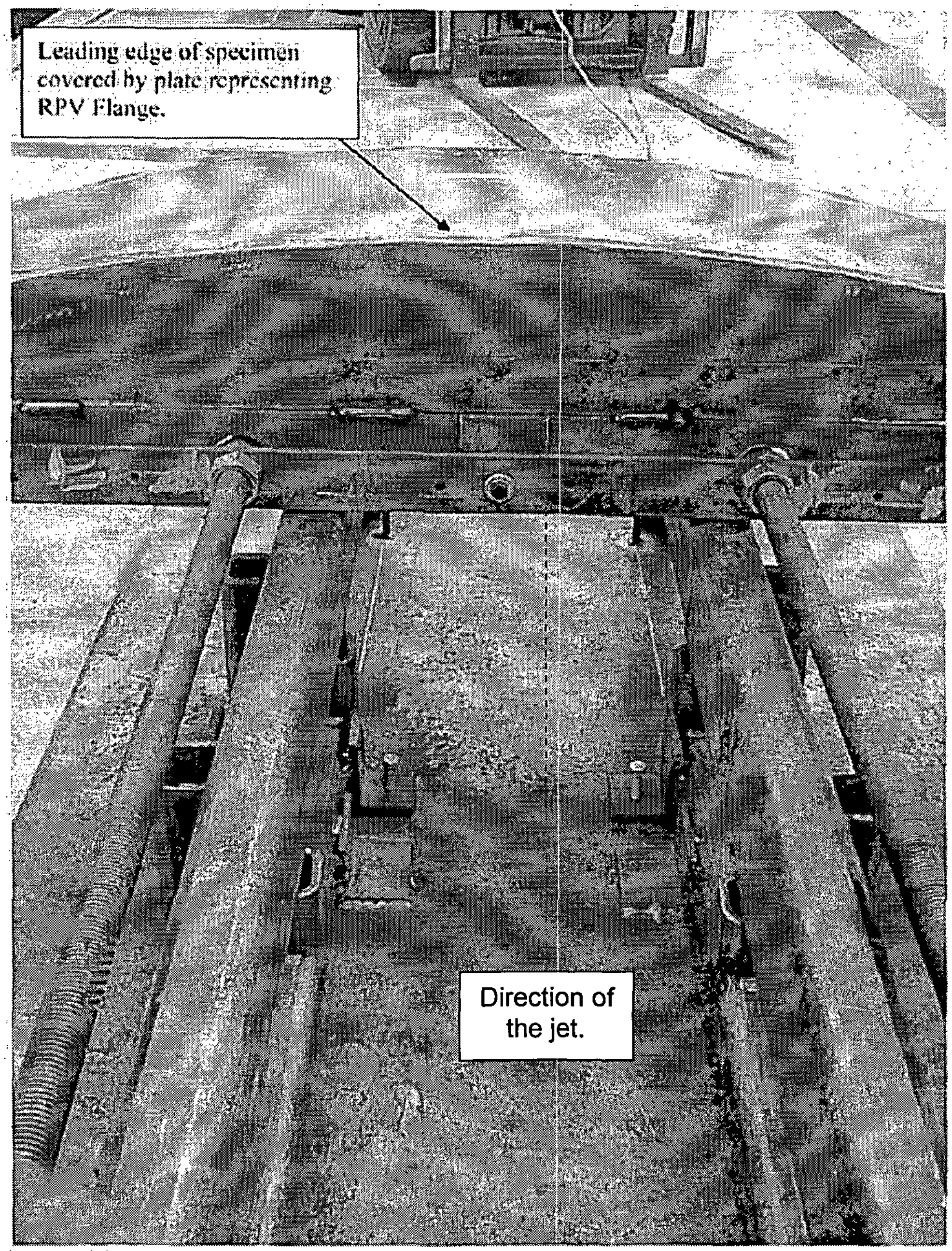

Figure 7-2
Reactor Pressure Vessel Top Head Panel in Test Rig

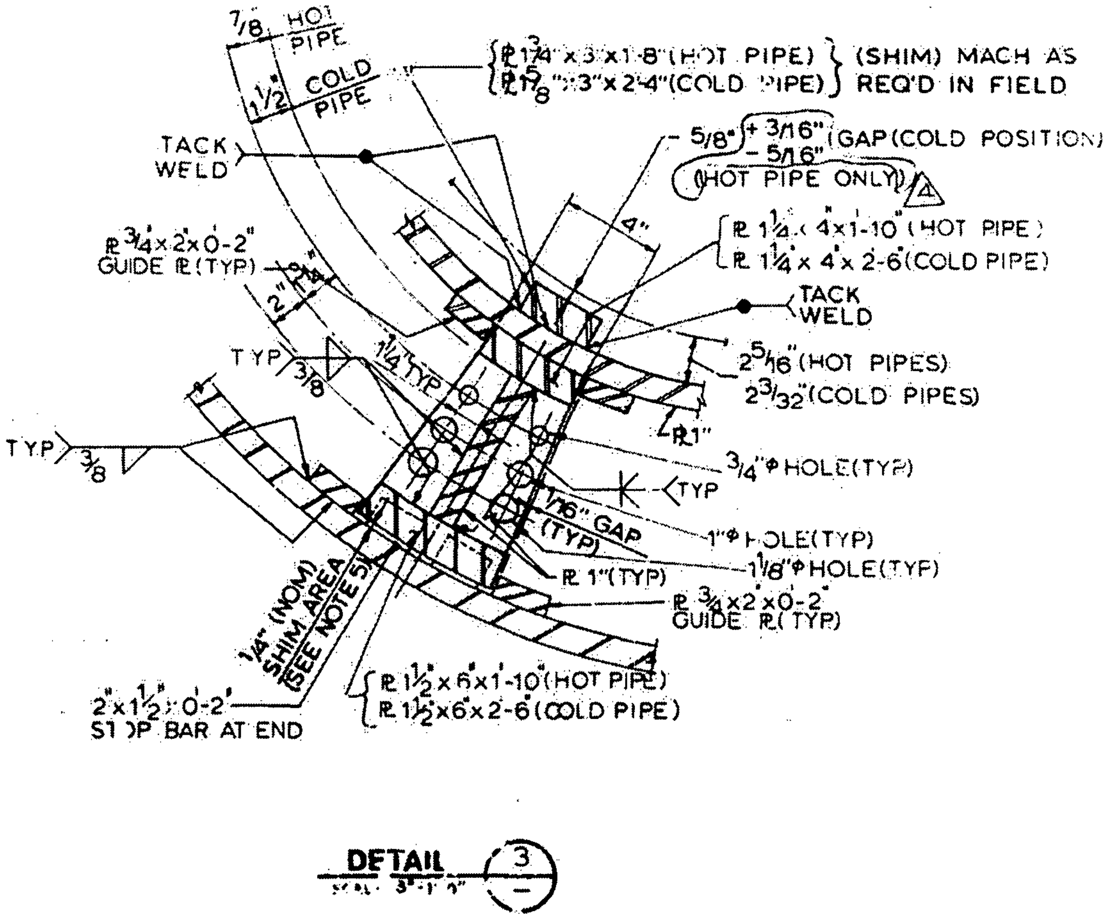

Figure 7-3

Schematic Diagram of "Wagon Wheel" Pipe Whip Restraint;
Design Gap between Shim and Loop Piping versus Loop Pipe Wall Thickness

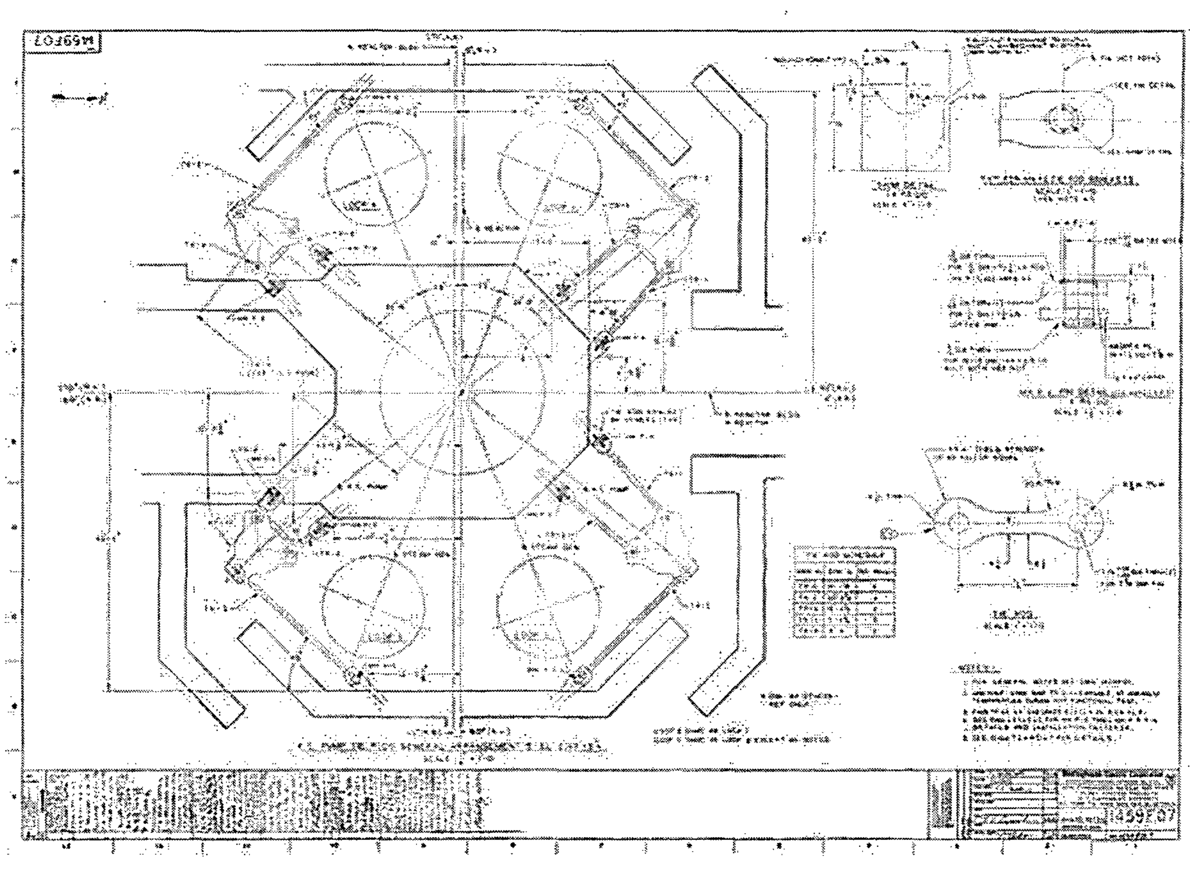

Figure 7-4
Drawing 1459F07 RCP Tie Rods General Arrangement & Details

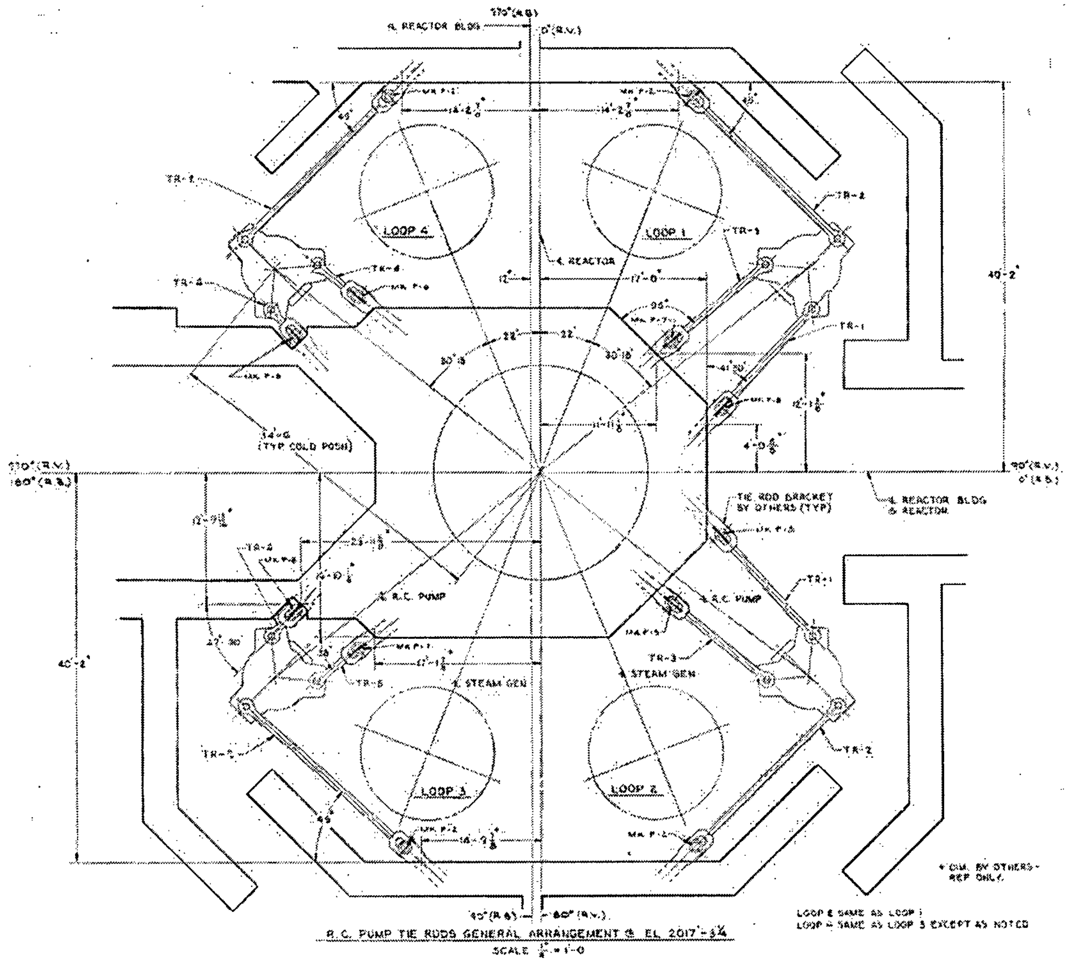

Figure 7-5
Detail from Drawing 1459F07 RCP Tie Rods

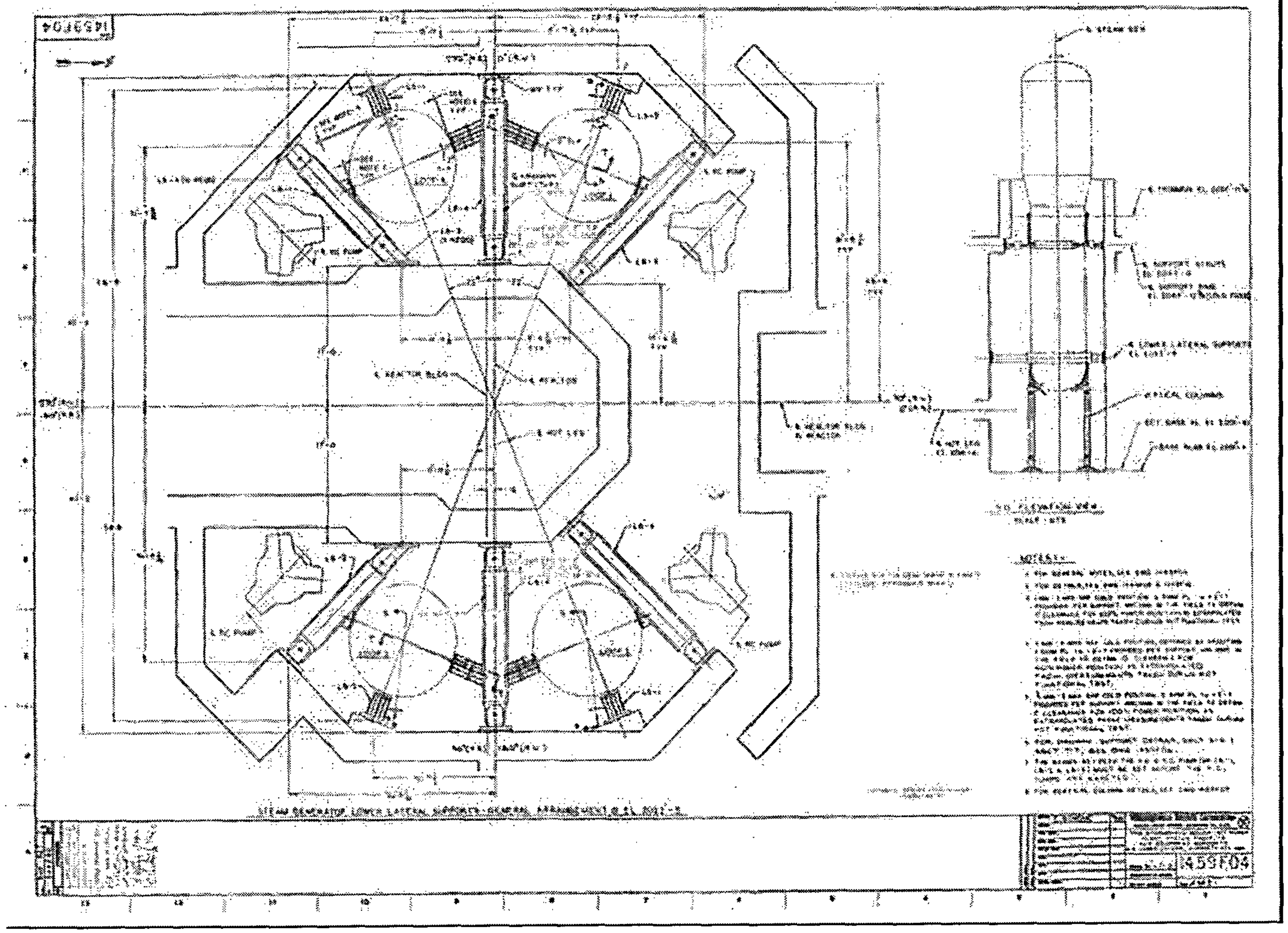

Figure 7-6
Detail from Drawing 1459F04 Steam Generator Lower Lateral Supports General Arrangement

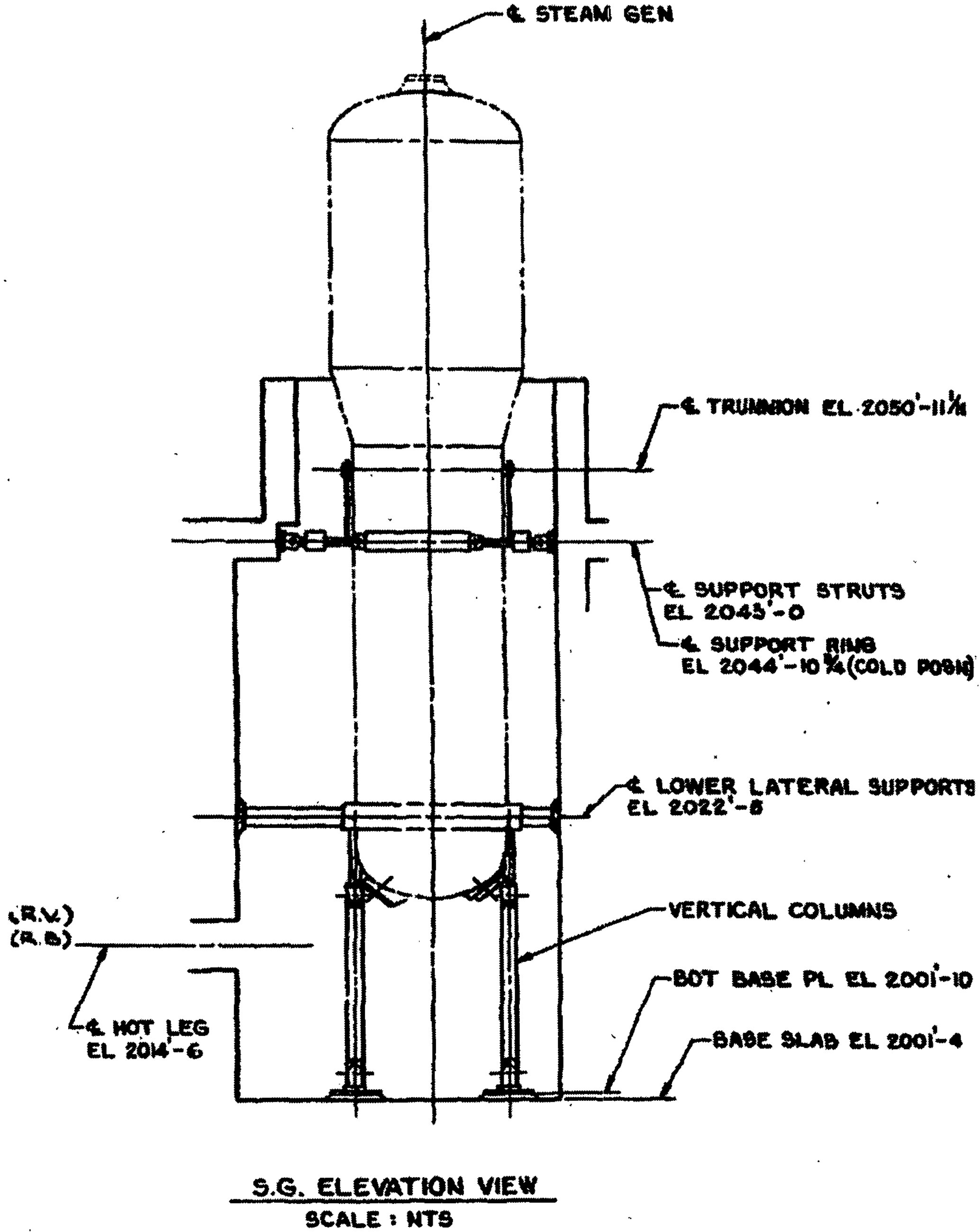

Figure 7-7
Detail from Drawing 1459F04 Steam Generator Lower Restraints

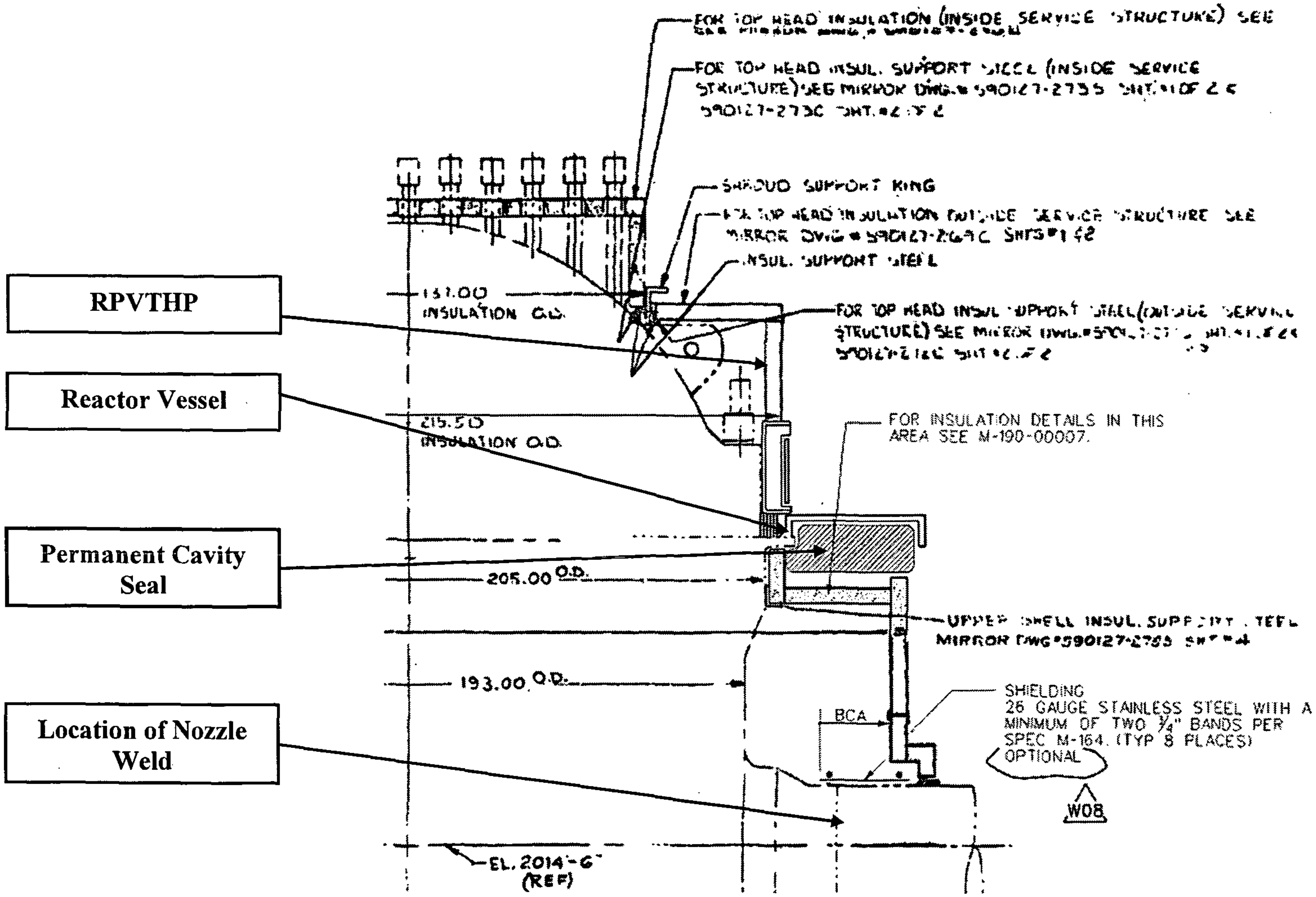

Figure 7-8
Reactor Pressure Vessel Top Head Panel as Installed