UNION ELECTRIC COMPANY
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April 5, 1984

DONALD F. SCHNELL
VICE PRESIDENT

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Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton: ULNRC-787

DOCKET NUMBER 50-483
CALLAWAY PLANT, UNIT 1
CALLAWAY PLANT TECHNICAL SPECIFICATIONS - APPEALS

References: 1) SLNRC 84-0029 dated February 10, 1984
2) SLNRC 83-0036 dated July 8, 1983

Attachments: 1) Specification 3/4 6.1.6 and justification
2) Specification 3/4 4.9.3 and justification
3) Specification 3/4 6.3 and justification

The purpose of this letter is to request an appeal meeting on three Callaway Technical Specification issues. As discussed in a meeting with NRC representatives (Messrs. Holonich and Anderson) this appeal meeting should be held the week of April 9 to support the Callaway licensing schedule. We are prepared to meet with NRC as soon as possible to discuss these issues.

The issues involved are Containment Vessel Structural Integrity (Spec. 3/4 6.1.6), Overpressure Protection Systems (use of RHR suction relief valves - Spec. 3/4 4.9.3), and Containment Isolation Valves (exemption of the provisions of Spec. 3.0.4 - Spec. 3/4 6.3). Attached are marked-up pages from these specifications along with justifications for these changes.

The changes on containment structural integrity were discussed at length with NRC in a meeting on January 5, 1984. The U.E. proposed specification was subsequently included in the Callaway Proof and Review and the NRC Final Draft Specifications.

The overpressure protection specification change and justification was previously submitted in Reference 1.

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The containment isolation specification change and justification was submitted by Reference 2. This change has been discussed with NRC and has been agreed upon from a technical point of view. We have recently been notified that this change cannot be made without prior review by the Committee to Review Generic Requirements. This is the basis for this appeal.

Please notify us when the meeting is scheduled. We are looking forward to meeting at your earliest convenience.

Very truly yours,

for Donald F. Schnell

DFS/cfs

Attachments

cc: J. Holonich
F. Anderson

STATE OF MISSOURI)
)   S S
CITY OF ST. LOUIS)

Robert J. Schukai, of lawful age, being first duly sworn upon oath says that he is General Manager-Engineering (Nuclear) for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on the behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Robert J. Schukai

Robert J. Schukai
General Manager-Engineering
Nuclear

SUBSCRIBED and sworn to before me this 5 day of April, 1984

Warren Schuchardt

MY COMMISSION EXPIRES JULY 25, 1987

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CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the structural integrity not conforming to the requirements of Specification 4.6.1.6.1, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity of the containment not conforming at a level consistent with the acceptance criteria of Specification 4.6.1.6.2, restore structural integrity or complete an engineering evaluation that assures structural integrity prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Tendons, End Anchorages and Adjacent Concrete Surfaces. The containment vessel tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 11 tendons (4 inverted U and 7 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include

at least 6 tendons (3 inverted U and 3 hoop). If more than one tendon has an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit, or with one tendon below 90% of the predicted lower limit, it shall be considered as evidence of possible abnormal degradation for the purposes of Specification 4.6.1.6.1g.;

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (inverted U and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire that:

1) The tendon wires are free of unacceptable (pitting of 1/64 inch or deeper and minimum of 1/32 inch in diameter) corrosion, cracks, and damage. The presence of unacceptable corrosion, cracks, or other damage shall be considered evidence of ~~possible~~ abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.;

2) There are no changes in the presence or physical appearance of the sheathing filler-grease. Abnormal changes in the presence or physical appearance of the sheathing filler grease shall be considered evidence of ~~possible~~ abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.; and

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test shall be considered as evidence of ~~possible~~ abnormal degradation of the containment vessel structure for the purposes of Specification 4.6.1.6.1g.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses adjusted to account for elastic losses exceed the average minimum design value given below:

Inverted U		139 ksi
Hoop:	Cylinder	147 ksi
	Dome	134 ksi

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SURVEILLANCE REQUIREMENTS (Continued)

e. Verifying the OPERABILITY of the sheathing filler grease by assuring:

1) If the installed quantity of grease exceeds that withdrawn by 5% or more, an investigation shall be conducted to assure that excessive leakage has not occurred in the tendon duct system,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

Failure to satisfy Specification 4.6.1.6.1e. 2) or 3) above for OPERABILITY of the sheathing filler grease shall be considered as evidence of [redacted] abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.

f. Determining through inspection that no apparent degradation has occurred in the visual appearance of the end anchorage or the concrete surfaces adjacent to the end anchorages. If apparent degradation has occurred in the visual appearance of the end anchorage or the concrete surfaces adjacent to the end anchorages, it shall be considered as evidence of [redacted] abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.; and

g. If evidence of [redacted] abnormal degradation of the containment structure is detected during the performance and/or evaluation of the results of the above tests, the following actions shall be completed:

1) Reported to the NRC within 10 days,

2) Perform an engineering evaluation demonstrating the continued ability of the containment structure to perform its design function. If continued containment integrity cannot be assured by engineering analysis within 90 days, ACTION a. required by Specification 3.6.1.6 shall be taken, and

3) Provide a determination of the cause of the apparent degradation and performance of any corrective actions necessary to ensure continued containment integrity.

4.6.1.6.2 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

TECHNICAL SPECIFICATIONS: CONTAINMENT VESSEL STRUCTURAL INTEGRITY

There are two fundamental differences between the Technical Specifications for Containment Vessel Structural Integrity required by the NRC and those requested by the Utility. Firstly, the NRC version requires that whenever any element of the tendon surveillance acceptance criteria is not satisfied, it be considered as evidence of abnormal degradation of the structural integrity of the containment vessel requiring that the integrity be restored or that the plant be shut down. The Utility version, however, would classify a failure to satisfy an element of the acceptance criteria as evidence of possible abnormal degradation and would permit an evaluation to be performed demonstrating the continued ability of the containment structure to perform its design function. Shut down would be involved only if, after the fixed evaluation period of 90 days, continued containment integrity could not be assured. The technical reasons for the insertion of an evaluation loop between discovering of a failure to satisfy a surveillance criteria item and the requirement to restore or shutdown is as follows:

1. The containment vessel post tensioning system is a passive element designed to provide a minimum level of prestress of 1.2 x the design accident pressure at the end of the 40 year design life of the plant. As such, there is a wide margin of safety at the beginning of the plant life which decreases to a minimum value of 1.2 at 40 years. There are 251 individual tendons involved in providing the prestressing of the containment. A level of prestress in any two tendons lower than the predicted lower limit and 90% of the predicted lower limit, cannot be considered as evidence of abnormal degradation of the containment vessel and should not be treated as such.

2. Periodic tendon surveillances are required by Reg. Guide 1.35. After the first two surveillances which occur at 1 and 3 years after the structural integrity test, surveillances are required every five years thereafter. The length of the intervening period of 5 years between successive surveillances is evidence of recognition of the extreme unlikelyhood of the deterioration of the post tensioning system to a level where vessel integrity would be affected in this time period. As such, there is no basis for requirement for restoration or shutdown within the time periods given in the NRC Technical Specification.

3. This same philosophy of evaluation to assure continued integrity has recently been approved by NRC for a plant with a containment similar to the Callaway containment. Since no fundamental difference in the post tensioning systems exist, which would affect this philosophy, a precedent has been established to accept this specification.

4. The present NRC Technical Specification for Containment Vessel Structural Integrity could result in unnecessary plant shutdown for reasons which do not affect the continued ability of the containment vessel to perform its design function. For example, according to the NRC Specification, if there are changes in the presence or physical appearance of the sheathing filler - grease, it is required that the vessel be restored to the required level of integrity within 72 hours or the plant be shut down. If it is not possible to replace the

grease within 72 hours, regardless of whether it is still performing its function (though of changed appearance) or whether the tendons are affected, then it would be required unnecessarily to shut down the plant.

The same sequence of events is set in motion if corrosion is found during tendon surveillance.

Secondly, the NRC Technical Specification requires that inspection of the concrete adjacent to the end anchorages of the surveillance tendons be performed during ILRTs. This is consistent with R.G. 1.35, Rev. 2. However, SNUPPS is not committed to Rev. 2 of this regulatory guide, but rather to the proposed Rev. 3. Since the proposed revision 3 no longer requires inspection of concrete adjacent to surveillance tendon end anchorages, during ILRTs there should be no such requirement in the SNUPPS Technical Specifications. The SNUPPS proposed Tech. Spec. requires that the inspections of the end anchorage concrete be performed during the surveillances.

In summary, the NRC Technical Specification for Containment Vessel Structural Integrity fails to recognize the fundamental difference between a passive structural system and an active mechanical or electrical component in that almost immediate restoration is required for the structural system regardless of whether integrity is affected.

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LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two Residual Heat Removal (RHR) suction relief valves each with a setpoint of 450 psig ± 1%, or

b. ~~a.~~ Two power-operated relief valves (PORVs) with Setpoints which do not exceed the limit established in Figure 3.4-4, or

c. ~~b.~~ The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHQ suction relief valve inoperable, either restore ~~the inoperable PORV~~ two PORV's or two RHQ suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs and both RHQ suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event either the PORVs, the RHQ suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHQ Suction relief Valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

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4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

INSERT A

4.4.9.3.~~2~~ 3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

INSERT A

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve (RRSIV) 8701B is opened with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RRSIV 8702B is opened.

b. For RHR suction relief valve 8708A

1) By verifying at least once per 31 days that RRSIV 8702A is opened with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RRSIV 8701A is opened.

c. Testing pursuant to Specification 4.0.5

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LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) A day tank containing a minimum volume of 390 gallons of fuel,

2) A fuel storage system containing a minimum volume of 85,300 gallons of fuel, and

3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the spent fuel pool. ~~and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2 square inch vent.~~ In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5)), and 4.8.1.1.3.

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D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt Battery Bank NK11 and NK13, and its associated full capacity charger NK21 and NK23, or

b. 125-Volt Battery Bank NK12 and NK14, and its associated full capacity chargers NK22 and NK24.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible. ~~and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2-square-inch vent.~~

b. With the required full-capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Specification 4.8.2.1a.1) within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and associated chargers shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

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ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One division of A.C. emergency busses consisting of one 4160-volt and one 480-volt A.C. emergency bus,

b. Two 120-volt A.C. vital busses energized from their associated inverters connected to their respective D.C. busses, and

c. One 125-volt D.C. bus energized from its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible. ~~and within 8 hours depressurize and vent the RCS through at least a 2 square inch vent.~~

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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HEATUP (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

, two RHR suction relief valves,

The OPERABILITY of two PORVs or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

or RHR suct relief valve

Insert 1 →

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

INSERT 1

RHR RCS suction isolation valves 8701A and B are interlocked with an "A" train wide range pressure transmitter and valves 8702A and B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701B and 8702A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths.

In addition to opening RCS vents to meet the requirement of 3.4.9.3.c, it is acceptable to remove a pressurizer code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

SPECIFICATION 3.4.9.3

Westinghouse has recently completed an evaluation of the use of Residual Heat Removal (RHR) suction relief valves to mitigate the consequences of cold overpressure transients. The results of the evaluation can be summarized as follows:

1. The RHR suction relief valves have sufficient capacity to mitigate the consequences of the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50 degrees-F above the RCS cold leg temperatures, or the start of a centrifugal charging pump and its injection into a water solid Reactor Coolant System (RCS).

2. The design of the RHR suction relief valves and the nature of the above two transients are such that the pressure overshoot resulting from a design cold overpressure transient will be limited to a value less than the RHR suction isolation valve high pressure auto-closure setpoint. Hence, the cold overpressure transient cannot result in isolation of the RHR suction relief valves.

3. RHR suction isolation valves 8701B and 8702A must be opened with power to their valve operators removed. With this arrangement, a spurious actuation of the RHR auto-closure interlock or a single failure in an RHR suction line cannot isolate both RHR suction lines, which leaves one RHR suction relief valve open to the RCS for overpressure protection.

Westinghouse has determined that the preferred system for use in mitigating cold overpressure events is the RHR system. Incorporation of the revised specification improves plant operability by increasing the number of systems available for cold overpressure mitigation.

As a note Westinghouse has also determined that should the normal overpressure mitigation systems be inoperable, it is acceptable to remove a pressurizer code safety valve, or disassemble and remove the internals of a PORV and secure the block valve in the open position by removing power, or otherwise open the RCS to meet the requirements of a vent flowpath.

JHR/dck/14a8

1.0 POTENTIAL OVERPRESSURE TRANSIENTS

1.1 General

Potential overpressurization transients to the reactor coolant system (RCS), while at relatively low temperature (less than about 350 degrees-F), can be caused by either of two types of occurrences to the RCS: that is, mass input or heat input. Both types results in more rapid pressure changes when the RCS is water solid. Therefore, the description of the following two types of transients will imply that the RCS is water solid, at a relatively low temperature and is open to the residual heat removal (RHR) system.

Mass Input Type Transients

M1. Inadvertent safety injection actuation or

M2. Charging/letdown mismatch

Heat Input Type Transients:

H1. Actuation of pressurizer heaters or

H2. Loss of residual heat removal or

H3. Temperature asymmetry within the RCS or between the RCS and Steam Generators

1.2 Mass Input Transients

M1. Inadvertent actuation of safety injection considerations include full system (both trains), single train or single component within a train event. Each of the three types of events are discussed separately.

Full system actuation would include the opening of the isolation valves on all SI accumulators, startup of all low head and high head safety injection pumps and isolation of the normal letdown path to the Chemical and Volume

Control System. Such an event would result in unacceptable, large volumes of coolant being forced into the RCS. Therefore, such events are prevented by strict administrative controls which require the blocking of the automatic SI actuation circuits, immobilizing the SI accumulator motor operated isolation valves by locking out their power supplies and locking out power to the high-head safety injection pumps. (See Section 2.2). The low head safety injection (RHR) pumps are normally in operation, taking their suction from the RCS, during low-pressure, low-temperature plant operations. Therefore, even if a spurious start signal were received, the low-head safety injection pumps would not function in their safety injection mode.

Single train actuation is less likely than full system actuation since the signals which call for safety injection, both manual and automatic, are processed through the engineered safety features logic circuits such that a signal whether spurious or not will impact both trains. However, since the safety injection system is essentially immobilized at low temperature, single-train, inadvertent actuation is considered no more likely than full system actuation.

Inadvertent actuation of a single component would require that a human operat selectively unlock the power to the component then cause the component to be energized. The likely way this event would occur would be during the Technical Specification required periodic surveillance testing or during post maintenance check-out of the component. Opening of an SI accumulator isolation valve while the accumulator is pressurized is not considered probable because maintenance procedures for the valves require that the compressed gas in the accumulator be removed. Post maintenance check-out or periodic surveillance tests, however, might be attempted on the high head safety injection pumps providing an opportunity for operator error to cause an inadvertent single pump injection event. Therefore, this single pump event is considered a potential mass input transient but, since the RHR loop would be open at this time, an RHR relief valve has adequate relief capacity to mitigate the resulting RCS pressure transient.

M2. Charging/letdown mismatch events can be postulated to occur in a number of ways. One way would be the complete termination of letdown by operator error, such as closure of the letdown control valve, isolation of

the RHR/CVCS crossover path or closure of the RHR inlet isolation valves or by malfunctions of the control systems which result in the same interruption of letdown from the RCS. A second way would be to increase the charging flow by either operator or instrument error such that the charging flow exceeds the prevailing letdown flow.

The most severe mass input transient would occur if the letdown flow control failed to the zero flow condition while the charging flow control failed to the full flow condition. This last failure mode would result in the maximum mass input transient for the charging/letdown mismatch events but does not result in the isolation of the RHR relief valves from the RCS. Since the mass input is less than or equal to the mass input for M1 above, an RHR relief valve would mitigate this transient and prevent an overpressure condition in either the RHRS or the RCS.

1.3 Heat Input Transients

H1. The inadvertent actuation of the pressurizer heaters when the pressurizer is filled solid will cause a slow rise in the water temperature (350 degrees-F/hr) with a consequent increase in pressure of the constant volume system. Since the temperature/pressure transient is very slow, the operator will recognize and terminate the transient before the relief valve setpressure is reached. The RHR relief valves will be actuated if the operator does not first intervene and stop the transient. This case is not considered significant compared to the design relief capacity of the RHR relief valves.

H2. A loss of residual heat removal cooling while the pressurizer is filled solid could be caused by a loss of flow malfunction in the service water or component cooling water systems or spurious closure of the RHR inlet isolation valves. The continual release of core residual heat into the reactor coolant with no heat rejection to the environs would cause a slow rise in the coolant temperature and pressure. Loss of service water or component cooling water will not result in isolation of the RHR suction relief valves, hence, the relief valves will be available to mitigate the pressure rise. Spurious isolation of both RHR suction relief valves simulataneously is prevented by removing power from one open RHR suction isolation valve in each suction path. A single failure combined with loss of cooling will not result in loss of overpressure mitigation capability (See Section 2.2).

H3. During plant heatup and cooldown operations, one reactor coolant pump is normally maintained in operation whenever the reactor coolant temperature is greater than 160 degree-F (See Section 2.3). Therefore, the large volumetric flow throughout the RCS will maintain an isothermal condition in the RCS. The steam generator secondary side water immediately surrounding the tubes will also remain at a temperature near that of the circulating reactor coolant on the primary side.

During normal cooldown operations, when the reactor coolant temperature has been decreased below 160 degrees-F the reactor coolant pumps may be stopped. Consequently, isothermal conditions in the RCS may no longer exist. The reactor coolant temperature will be decreased below 160 degrees-F by continued input of cold charging and seal injection water and by heat rejection to the residual heat removal loop. The steam generator contained water (both primary and secondary) may remain at a relatively constant temperature since there may be little circulation through the tubes. Therefore, a significant temperature asymmetry can be developed. If a reactor coolant pump is then started, the sudden heat input into the reactor coolant from the steam generators or the rapid warming of the stagnant pool of cold seal injection water will cause a rapid increase in reactor coolant temperature. If the event should occur while the pressurizer is filled solid, a rapid increasing pressure transient would occur. Normally, the RHR loop is open to the RCS when the pressurizer is filled solid, so that the relief valves in the RHR loop will mitigate the pressure transient. In addition, the Cold Overpressure Mitigation System has adequate relieving capability to prevent the over pressurization should the event occur with RHR suction relief valves isolated.

If the pump start event should occur when a steam bubble of normal volume is present in the pressurizer, the resultant expansion of the reactor coolant due to the sudden heating will be accommodated by an insurge into the pressurizer with only a small reactor coolant pressure change.

Under most plant operating conditions during heatup, cooldown or at cold plant conditions, the RHR loop is open to the RCS and heat input transients will be mitigated by the RHR relief valves.

2.0 TECHNICAL SPECIFICATIONS AND ADMINISTRATIVE CONTROLS

2.1 General

All reactor plant operations are guided by administrative controls which specify the availability of equipment, system alignments and ranges of process parameters. The administrative controls are intended to prevent deliberate plant operations which would result in potentially unsafe or unacceptable plant conditions. These administrative controls are identified in either; 1) the Technical Specifications issued by the NRC, or 2) the Precautions, Limitations and Setpoints document, special notes and limits in operating instruction guidelines and technical bulletins issued by the NSSS designer.

2.2 Deactivation of Safety Injection System

The standard Technical Specifications (STS) describe the ECCS requirements to have a minimum number of components available during Operational Modes 1, 2 and 3 (Sections 3.5.1 and 3.5.2) and also for Mode 4 (Section 3.5.3) to assure that sufficient core cooling capability will be available. However, the footnote to Section 3.5.3 also specifies that a maximum of one centrifugal charging pump shall be operable. Additionally Specification 3.5.3 requires that both high head safety injection pumps be made inoperable in Modes 4, 5 and 6 whenever the reactor vessel head is on by removing power from the pump motor. These controls ensure that at most only one ECCS high head injection pump could be started by a spurious SI signal limiting the mass injection to within the capabilities of the RHR suction relief valves.

The Technical Specifications do permit the block of the actuation circuits from the pressurizer and steamlines instruments to avoid spurious actuations at low system pressure and temperature.

In addition to the administrative permission to limit or block safety injection provided by the Technical Specifications, Westinghouse has provided operating instruction (O.I.) guidelines and Precautions, Limitations and Setpoints (PLS) document which specify the following

additional requirements to avoid or limit potential RCS overpressure transients originating from the Safety Injection System:

During Plant Cooldown and Depressurization

- Block the automatic safety injection circuit when the reactor coolant pressure is reduced below the automatic unblocking setpoint.

- "Close the accumulator isolation valves and lock out the valve controllers when the reactor coolant pressure is less than 1000 psig and the temperature is less than 425 degrees-F."

- "When the reactor coolant pressure is less than 1000 psig, lock out the safety injection pumps and non-operating charging pumps."

During Plant Heatup and Pressurization

- "Unlock all safety injection and charging pumps after the pressurizer steam bubble is formed and the residual heat removal loop is isolated." (not addressed in STS)

- "Verify that the safety injection actuation block is automatically removed at the pressure setpoint." (STS Section 3.32)

2.3 Precautions When Operating in Water Solid Mode

The Standard Technical Specifications require that when in Mode 4 and when in Modes 5 and 6 with the reactor vessel head in place, that there be overpressure protection systems operable (Section 3.4.9.3). Sections 3.4.1.3 of the STS, which is applicable to Mode 4, recognizes, in a footnote, that restart of a reactor coolant pump in a water solid system can cause heat input type pressure transients.

The O.I. guideliens and PLS document provide additional caution statements which are intended to prevent the development of conditions conducive to overpressure transients in the RCS and thus the need to

challenge the water relief valves in the RHRS, or if the RHRS valves are not available, the COMS.

To avoid developing those conditions which might lead to a heat input type transient to the RCS or to transients caused by operator action, the following caution statements have been provided to the operators:

- "During cooldown, all steam generators should be connected to the steam header to assure uniform cooldown of the reactor coolant loops."

- "Whenever the reactor coolant temperature is above 160 degrees-F at least one reactor coolant pump should be in operation to maintain uniform temperature conditions in the system."

- "If all reactor coolant pumps have been stopped for more than five minutes and the reactor coolant temperature is greater than the charging and seal injection water temperature, do not restart the first pump until a steam bubble has been formed in the pressurizer. This precaution will minimize the pressure transient when the first pump is started and the cold water previously injected by the charging pump is circulated through the warmer reactor coolant components. The steam bubble will accommodate the resultant expansion as the cold water is rapidly warmed."

- "If all reactor coolant pumps are stopped and the reactor coolant is being cooled down by the residual heat exchangers, a non-uniform temperature distribution may occur in the reactor coolant system. Do not attempt to restart a reactor coolant pump unless a steam bubble exists in the pressurizer."

Additionally, the requirement to "and within 8 hours, depressure and vent the Reactor Coolant System through at least a 2 square inch vent" in the Action statements of Specifications 3.8.1.2, 3.8.2.2, and 3.8.3.2 has been deleted. This requirement is no longer applicable to the SNUPPS Utilities with the requirement in Modes 5 & 6 to have an OPERABLE RHR system as described in Specifications 3.4.1.4.1, 3.4.1.4.2, 3.9.8.1 and 3.9.8.2 and the use of the RHR Suction Relief Valves for cold overpressure protection.

See following page

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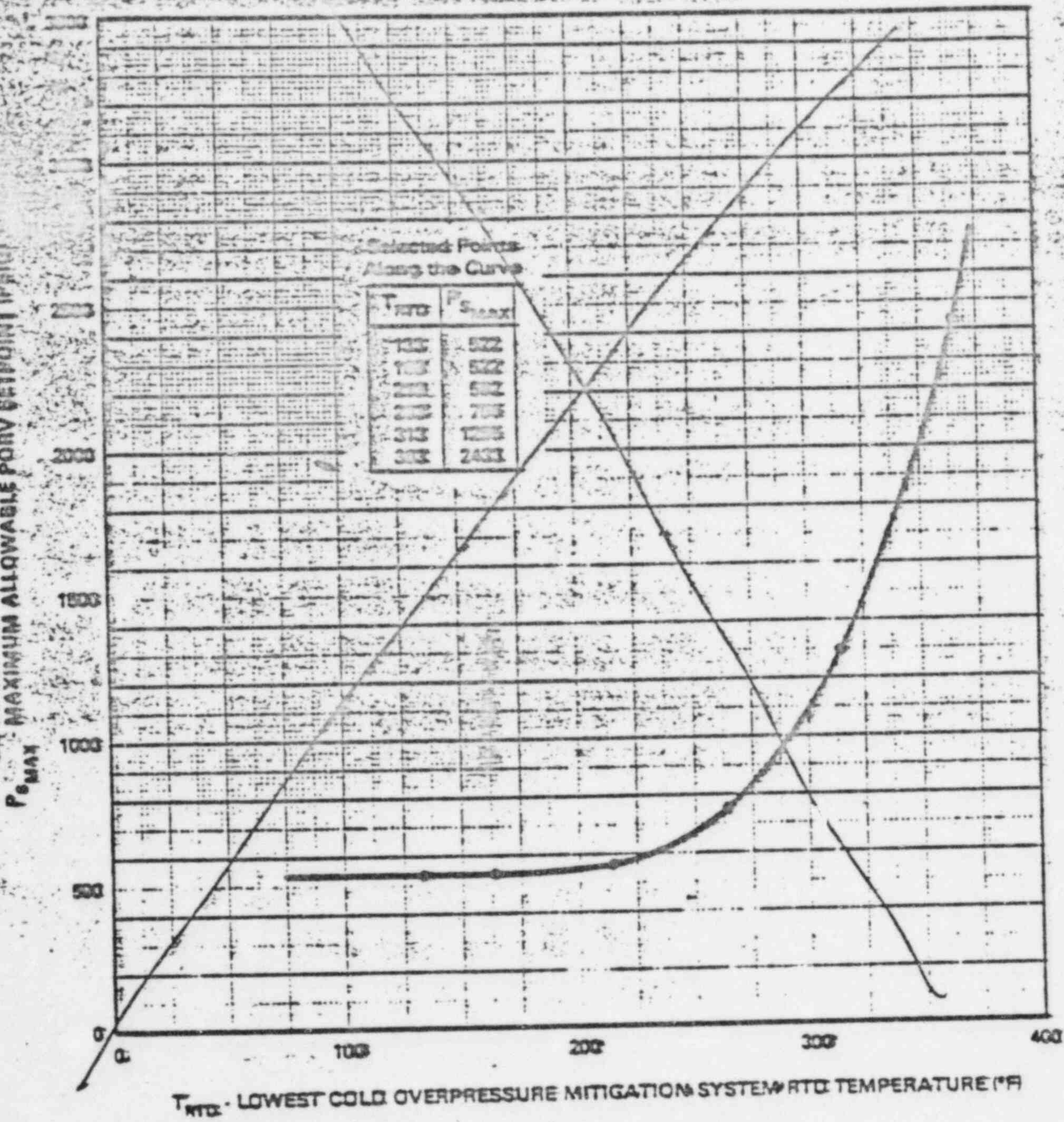
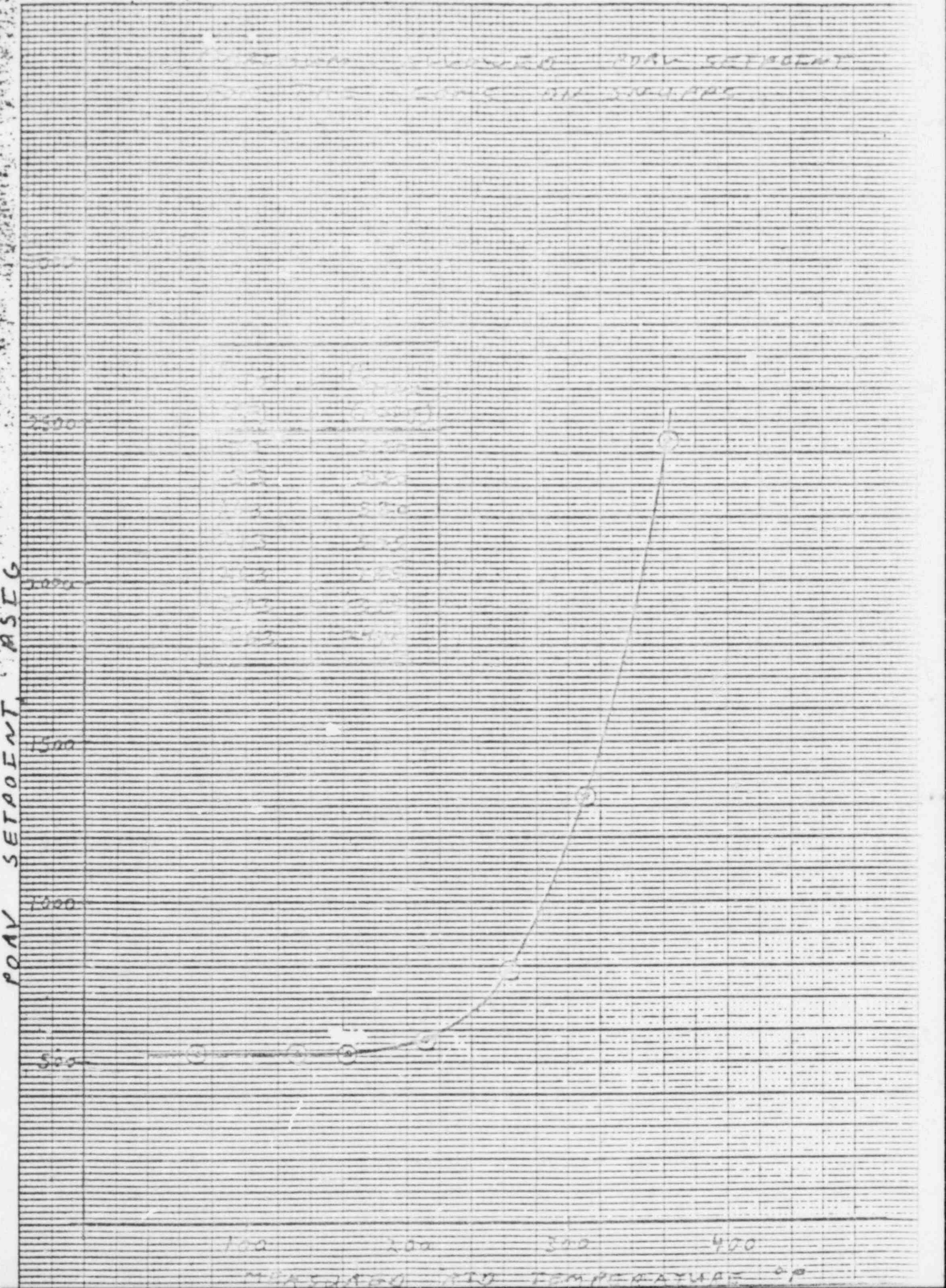


FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM


PORV SETPOINT, PSIG
2500
2000
1500
1000
500
100
200
300
400

DRAFT

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position and the provisions of Specification 3.0.4 are not applicable, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange and the provisions of Specification 3.0.4 are not applicable, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

Technical Specification 3.6.3 Containment Isolation Valves
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Justification -

Excepting the provision of specification 3.0.4 allow power operations to be reestablished and operation to continue once the effected penetration has been isolated and deactivated. Operations with some penetrations isolated can be accomplished safely by utilizing alternate flow paths or by simply not utilizing the effected penetrations function. The isolated penetration represents no additional safety concerns as it remains in its' safeguard or isolated position until maintenance can be performed and the affected penetration declared operable.