Florida Power
CORPORATION

July 14, 1988
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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Emergency Feedwater

Dear Sir:

Florida Power Corporation (FPC) has reviewed the Confirmation of Action Letter dated July 6, 1988. The attached information is provided as a supplement to the correspondence submitted on June 28, 1988, concerning the emergency feedwater leaks and summarizes the July 8, 1988 presentation in Rockville, Maryland. The issues associated with the appropriate testing of check valves employed as containment isolation valves, such as FWV-43 and FWV-44, will be discussed further at a follow-up meeting yet to be scheduled. The issues associated with our corrective actions for previous events were discussed at the July 8 meeting and will be the subject of separate correspondence with Region II associated with Inspection Report 88-18.

If you have any questions, please contact this office.

Sincerely,

Ken Wilson

K.R. Wilson, Manager
Nuclear Licensing

KRW/REF/dhd
Attachment

xc: Regional Administrator, Region II

Senior Resident Inspector

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CONFIRMATION OF ACTION LETTER

SUMMARY OF JULY 8, 1988 PRESENTATION

Background on Current EFW Line Heat up Event:

During Refuel VI, several emergency feedwater (EFW) system valves underwent refurbishing. These efforts included the repair of EFV-33 and the inspection of EFV's 15, 16, 17, and 18 (see Attachment 1 flow schematic). Two (2) of these valves, EFV-18 and EFV-33, were identified as having body to bonnet leaks during Refuel VI startup activities on January 7, 1988. Efforts to secure the leaks were unsuccessful. The unit was returned to service on January 10, 1988.

Crystal River Unit 3 experienced a reactor trip on February 28, 1988, due to a malfunction in the feedwater system which resulted in an EFW injection (LER 88-06). The plant entered a MODE 3 outage for 4 days to investigate the cause of the trip. The unit was returned to service on March 3, 1988, but limited to 60% power. The unit was removed from service on March 7 and entered MODE 2 to effect repairs on the main feedwater flow control valve. During the startup on March 8, the Operations staff discovered all four EFW lines hot. The check valves in the EFW lines were apparently allowing some backflow causing the heat up of the EFW lines. Estimates of the temperatures and conditions of the lines were provided by Operations to Engineering. Follow-up inspections of the EFW lines determined that the temperature had returned to normal following a cool down process which involved flowing a small amount of emergency feedwater through the lines. This action evidently caused the leaking check valves in the lines to reseat since these lines remained cool until May.

On May 27, 1988, during a routine plant walk down, the Shift Operations Technical Advisor (SOTA) discovered the emergency feedwater piping near flow control valve EFV-55 to be warmer than normal, indicating check valve FWV-43 was not preventing backflow into the emergency feedwater line from the steam generator. This condition was discussed with the Shift Supervisor on duty and increased awareness of the condition was maintained by site personnel. By June 19, the temperature of the line from the reactor building penetration to EFV-18 was hot enough to cause some flashing of the water leaking from EFV-18. A Non-Conforming Operations Report (NCOR 88-81) was written on June 19, and upon further investigation a one hour report was initiated on June 21. The report was based on exceeding the design basis temperature for the emergency feedwater line. Prior to this time, the line was thought to have been designed and analyzed for much higher temperatures.

As a result of this heat up, visual inspections and engineering analyses were conducted. These activities included:

- Visual inspections of the penetration and walkdowns of accessible piping and supports by Engineering personnel
- Pressure test of Penetration #109
- Dye penetrant inspection of high stressed weld on Penetration #109
- Thermal stress analysis on penetration steel

- Review of temperature effects on penetration concrete
- Thermal analyses of EFW piping inside and outside of reactor building
- Installation of a temporary modification to reinject leakage from EFV-18 to stop the EFW line heat up.
- Reevaluation of stresses caused by seismic, deadweight and internal pressure forces as a result of the high temperature condition.
- Radiograph of check valve FWV-43
- Repair of pipe restraint FFH-126
- Leak repair of EFV-18

Details of these activities were provided during teleconferences and in a letter to the NRC dated June 28, 1988. Additional information concerning the penetration analyses, the thermal flexibility analyses of the piping, the pipe support evaluations, and the leak repair of EFV-18 is provided below.

Penetration Analyses:

The penetration consists of a 14" sleeve and a 2" thick end plate inside the reactor building. A 6" Schedule 160 process pipe passes through the center of the sleeve and end plate. Welded lugs secure the sleeve to the concrete containment wall. A 3/4" plate is welded on the sleeve to provide a closure plate to facilitate testing of the penetration. The 3/4" plate is not considered part of the containment boundary, and is not credited as a pipe support in the piping analysis.

A two dimensional, steady state heat transfer model was developed to determine the maximum anticipated concrete temperature attained near the penetration inside the Reactor Building (RB). Based on a maximum process pipe temperature (measured) of 480° F, a maximum concrete temperature of 214°F was predicted. The model was validated by comparing its predications with measured values obtained at a location above the penetration outside the RB. Excellent agreement was found (maximum measured was 195°F).

Although the overall concrete design temperature for CR-3 is 150°F(based on ACI 318-1963), FPC considers the criteria established in ACI 349 & 359 to be relevant for this transient situation. ACI 349/359, Appendix A, permits localized concrete temperatures adjacent to penetrations to be 200°F for long term periods (continuous) and 350°F for short term emergency situations. Therefore, FPC concluded and visual inspection confirmed the concrete was not adversely affected by this elevated temperature condition.

The penetration sleeve and end plates were analyzed by conservatively assuming a 300°F differential temperature between the sleeve and process pipe. This exceeded the actual measured differential temperature of 208°F. Under a 300°F differential temperature, it was determined that the 3/4" closure plate would be stressed beyond its elastic limit (28,000 pounds-force). The corresponding stress in the end plate from this load is 6,000 psi. This stress, when added to the stress from the piping loads of 5,000 psi, are well within the allowable stress of 45,000 psi, thereby assuring that containment integrity was never compromised.

The 3/4" closure plate was loaded beyond its elastic range and was performing in the plastic range, however, FPC does not consider this to be a problem since its only function is to facilitate testing of the penetration. It has not been taken credit for in maintaining containment integrity or as a pipe support. An additional inspection and dye penetrant test were conducted to demonstrate no degradation has occurred to the 3/4" plate.

Piping Analysis:

A thermal flexibility analysis was performed on the piping inside and outside of the Reactor Building (RB). The analysis was perfrmed at the elevated temperatures.

The piping inside the RB was analyzed at a temperature of 780°F from Penetration #109 to a point midway to the steam generator, and at 590°F for the remaining portion. This temperature profile is a conservative assumption since the actual gradient between the steam generator and the penetration would be more gentle. This portion of the stress analysis demonstrated that the piping inside the RB from Penetration #109 to the steam generator was not stressed above ANSI B31.1,1967 code allowables during the elevated temperature condition. Maximum stress in the piping was calculated to be 17,256 psi which is below the allowable of 22,500 psi.

A thermal analysis was performed on the piping outside the RB at the elevated temperatures measured from EFV-18 to Penetration #109. The system supports were modeled to reflect the as found condition based on the visual inspections. The results of the analysis indicated that the piping outside the RB was not subjected to stresses in excess of code allowables. The maximum calculated thermal pipe stress was 14,427 psi which was less than the 22,500 psi allowable.

A pressure, deadweight, and seismic (SSE) load analysis of the piping outside the RB was performed with EFH-126 modeled in the damaged condition and providing only translational and torsional restraint. Under these load combinations, the maximum calculated pipe stress was 16,445 psi which was less than the allowable 18,000 psi.

Pipe Support Evaluations:

The effects of the loads imposed by the piping on the pipe supports were evaluated. The pipe supports inside the reactor building were not subjected to stresses in excess of B31.1 code allowables during the elevated temperature condition. In addition, only one (1) pipe support on the subject piping inside the RB utilized anchor bolts. Under the worst case combined loading, the anchor bolt safety factor was reduced to 5.6 compared to the minimum 4.0 specified in the NRC Bulletin 79-02. Therefore, pipe supports inside the RB remained fully functional during the elevated temperature condition.

The supports on the piping outside the RB were also evaluated. Base plate stresses in two supports (EFH-129,130) would have exceeded B31.1 allowables during seismic events. The worst case value of 33,100 was below the yield strength of 36,000 psi during this postulated loading condition.

During the postulated SSE loading condition, anchor bolt safety factors for two supports would have been below the minimum of 2.0 as specified in Bulletin 79-02 . The minimum calculated safety factor was 1.7 at EFH-132, which is above 1.0 and does not constitute a failure of the anchorage. It is important to emphasize that these are worst case load combinations based on seismic, deadweight and thermal pipe loads. Actual loading during the elevated temperature conditions, excluding seismic, was within allowables at all times.

EFV-18 Leak Repair:

On June 28, 1988, a temporary modification was made on EFV-18 to seal the body to bonnet leak. A low temperature, flexible sealant was injected by means of a manually operated pressure gun into the spaces on top of the bonnet and inside the lock ring of EFV-18 which totally eliminated the leak. A diagram of EFV-18 is provided in Attachment 2. The sealant was injected into the valve first through the modified cap bolts to seal the interior space and bolt screw holes (area "A"), and second through the two valve vent ports to seal the spaces in the valve lock ring and cap (area "B"). The sealant was injected at pressures below the system pressure to preclude sealant from entering the EFW line except for the final few strokes of the gun. These final strokes must occur at an injection pressure exceeding system pressure in order to compress the sealant and stop the leak. This method minimizes the potential for sealant to enter the valve internals. An EFW system flow test of approximately 50 gpm was used to confirm valve funct[illegible] was unaffected by the use of the sealant. A test of the valve integrity wa[illegible]cted by placing the EFW system into recirculation thereby, generating maxi[illegible] operating pressure. The chemistry of the sealant was certified to be within acceptable values and compatible with valve material.

Previous Experience with EFW Line Heat up:

A review of previous EFW events revealed three (3) License Event Reports (LER's) and two (2) Nonconforming Operations Reports (NCOR) had been written documenting the problems with the flow transducers on the EFW lines due to high temperatures. During the time frame of these events, FWV-43 was suspected of backleakage which resulted in the EFW line heat up. The flow paths for the leakage were not identified in the documents. A summary of previous events is provided in Attachment 3.

Planned inspections of EFW check valves were performed during outages in 1980, 1985 and 1987. The 1985 and 1987 inspections were planned PM's instituted as a result of SOER 84-3. A review of the maintenance history of the EFW system valves was performed and FWV-43 and EFV-33 were identified as having a trend of poor performance from the standpoint of backleakage and leaks to atmosphere. Evaluations have been performed in accordance with the Institute of Nuclear Power Operations (INPO), Significant Operating Event Report (SOER) 86-3, which also identified these valves as candidates for additional inspection and testing.

An announced NRC inspection was conducted at CR-3 during October 5-9 1987, to determine the current status of check valve test programs as a subset of the industry as a whole and to determine the responsiveness of the industry to the INPO recommendations for improving check valve testing programs (SOER 86-3). During the inspection, the NRC identified EFV-18 and FWV-43 as having maintenance problems and as being included in the SOER response program.

Future Action:

Florida Power Corporation on, or before, October 15, 1988, will permanently repair EFV-18 and EFV-33 body to bonnet leaks, and will inspect and repair, as necessary, FWV-43 and EFV-16 seat leakage. In addition, FWV-44 will be inspected and repaired, as necessary, based on the results of the inspections and repair efforts of FWV-43. FPC is currently evaluating the appropriate post maintenance test and acceptance criteria for FWV-43 and FWV-44.

Hourly temperature readings of the EFW process pipe through Penetration #109 will be maintained until a permanent fix is made. The once per shift check of EFW pumps discharge piping will continue, as well as refueling interval inspections of EFW check valves. A complete evaluation of the maintenance history on these valves will be performed and utilized in conjunction with the INPO SOER 86-3 evaluations to establish recommendations for possible changes to preventative maintenance programs.

A review will be performed to upgrade the EFW piping for actual temperatures expected in the system. This review will include high energy line, stratification and water hammer effects as appropriate.

CONFIRMATION OF ACTION LETTER RESPONSE

Responses to Specific Confirmation of Action Letter Items:

Item:

A discussion of the effect of potential EFW system degradation on the reliability of the system and on long-term plant safety.

Response:

The primary adverse consequence was EFW reliability. Florida Power Corporation maintains that the EFW reliability was unaffected since the piping, penetration and supports were demonstrated through analyses to be capable of performing their intended functions within the design allowables excluding seismic loads. The piping has been returned to within the design temperature and will be maintained within this specification, thereby, alleviating the additional thermal stresses experienced during the abnormal heat up events. The valves and system performance were and continue to be unaffected by the recirculation system or the injection sealant as discussed in earlier correspondence and the presentation summary contained herein.

Item:

A description of the repair of EFV-33 and of the injected sealant repair of EFV-18, results of the repairs, and description and results of testing to verify component/system operability after repairs.

Response:

The on-line repair of EFV-33 is under evaluation. Leakage from EFV-33 has decreased and will continue to be monitored. The injected sealant repair of EFV-13 and subsequent results and tests are discussed in detail in the previous pages as part of the July 8, presentation summary.

Item:

The results of all thermal and stress analyses of all involved piping, penetrations, and supports.

Response:

The results of the thermal and stress analyses of the penetration, piping and supports are discussed in the July 8, presentation summary. Additional analytical information is provided in Attachment 4.

Item:

Evaluation of the safety implications of the valve leakage and EFW system overtemperature and of the EFW system with the interim fix and the temporary repairs in place.

Response:

The EFV system valve leakage did not pose any safety concerns. The leak rate did not exceed 0.9 gpm and had no adverse effects on the EFW flow to the steam generators. The potential for water hammer was considered unaffected by these events due to the following:

- The piping diameter was small
- The piping has short horizontal runs and numerous bends
- The piping has rigid supports at the majority of the pipe bends
- The EFW enters into the steam space in the steam generator
- The majority of the piping remained filled with water because the temperature was below the saturation temperature of 540°F for a majority of the piping.
- The piping generally rises from the EFW pumps to the steam generator.

The potential for steam binding of the EFW pumps was also considered. Monitoring was initially conducted on a once per shift basis and later upgraded to hourly to quickly identify conditions which could lead to steam binding. There was never a problem or threat of steam binding of the EFW pumps as a result of the leaks since other valves in the lines were preventing back flow and heat up of the system. The FWV-43 back-leakage was not considered to be a significant safety concern since the valve was considered operable for EFW injection and minor backleakage was of little consequence as long as the EFW piping and pumps remained cool. The containment integrity issue was not considered significant since the piping and steam generator provided system integrity in the RB (closed system inside containment). For a steam generator tube rupture event, the off site dose consequences remain well within design basis for this accident.

The temporary injection system was carefully designed to consider EFW system isolation, seismic restraint and relief valve needs in order to adequately protect the reliability of the EFW system. The temporary EFW-18 leak repair was discussed in the July 8 presentation summary.

Item:

Verification that EFW components have not been damaged by overtemperature events and can reliably perform their safety functions.

Response:

The only adverse consequences not addressed elsewhere involved EFV-55 and EFV-57. The effects of higher than expected temperatures on the qualification life of EFV-55 and EFV-57 were evaluated, and sufficient margin is retained for continued operation. Early repair/replacement will be evaluated as part of our normal EQ maintenance program.

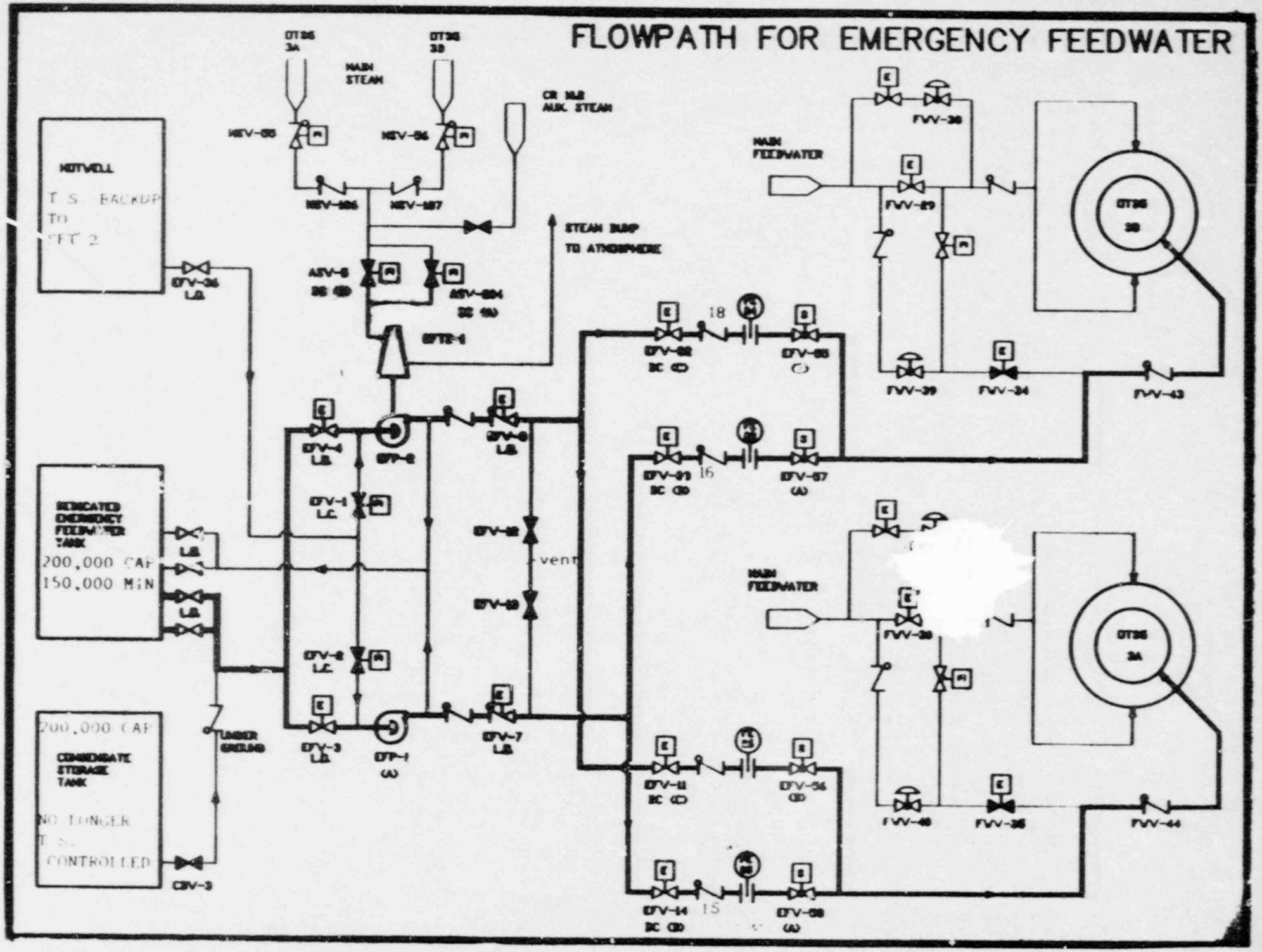
Item:

A discussion of the possible need for a more comprehensive inspection and repair program based on an evaluation of current and past valve leakges.

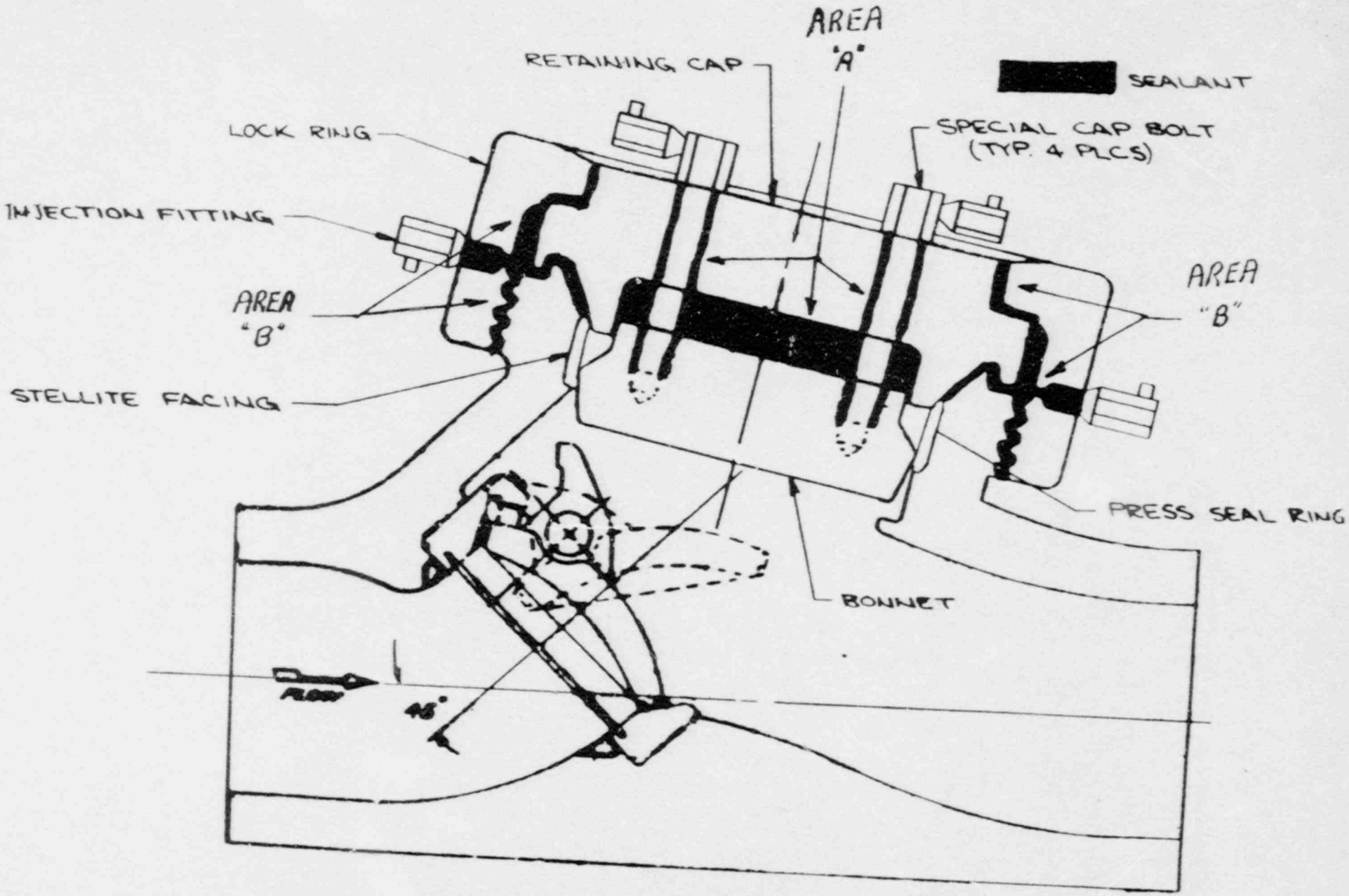
Response:

Florida Power Corporation is continuing to evaluate and develop a more comprehensive inspection and repair program for check valves. FPC will factor the EFW events into that program. These events do not contradict the priorities identified by that program (e.g. these were all high priority valves for enhanced efforts). FPC's detailed response to previous check valve leakage events will be discussed more throughly in separate correspondence.

FLOWPATH FOR EMERGENCY FEEDWATER



ATTACHMENT 2



ATTACHMENT 3

SUMMARY OF PREVIOUS EXPERIENCE

DATE		PROBLEM	CAUSE
	LER's		
1/19/83	82-076	HEAT FAILURE OF FLOW TRANSDUCER	FWV-43 LEAK
8/19/83	83-029	FAILURE OF FLOW TRANSDUCER	HIGH HEAT FAILURE OF SEALANT MOUNTING
8/24/84	83-043	FAILURE OF FLOW TRANSDUCER	FWV-43 LEAK
	NCOR's		
4/24/84	84-101	HEAT FAILURE OF FT	FWV-43 LEAK
6/18/84	84-148	FAILURE OF SPAN ACCURACY TEST (PM-243)	FWV-43 LEAK
	UOER		
10/15/82	82-25	17 MIN FEED TO HIGH NOZZLE FOLLOWING RESTART OF MFP	POST TRIP FAILURE OF FWV-39

ATTACHMENT 4

DETAILED ANALYSES

EMERGENCY FEEDWATER PIPING / SUPPORT REANALYSIS SUMMARY

INSIDE CONTAINMENT

- At Original Design Temperature of 110 Degrees F

All actual stresses on piping and supports were within the allowable limits specified in ANSI B31.1 1967 edition under deadweight, pressure, OBE and SSE loading conditions. Envelope of maximum worst case pipe stress on system due to deadweight, pressure, and SSE loading conditions = 17,365 psi which is less than the 18,000 psi allowable stress permitted in ANSI B31.1.0 1967 Edition.

All pipe support anchor bolt safety factors were within the requirements of NRC Bulletin 79-02. (i.e. Safety Factor greater than 4 for wedge type anchors under deadweight, pressure and SSE loading conditions.) Anchor Bolt Safety Factor for EFH-025A (only support attached to concrete in the effected piping) under deadweight, pressure, and SSE loading conditions = 19 .

- During Elevated Temperatures

All actual stresses on piping and supports were within the allowable limits specified in ANSI B31.1 1967 edition under deadweight, thermal , pressure, OBE and SSE loading conditions.

Maximum worst combination pipe stress due to deadweight, pressure, and SSE loading conditions remained unchanged.

Maximum thermal stress in piping = 17, 256 psi which is less than the allowable stress of 22,500 psi permitted in ANSI B31.1 1967 Edition.

Anchor Bolt Safety Factor for worst case baseplate for EFH-025A under deadweight, thermal, pressure, and SSE loading conditions = 5.6 .

OVERVIEW OF PIPING ANALYSIS
(Inside Containment)

6" Diameter Emergency Feedwater Piping from Penetration No. 109 to Steam Generator 3B was analyzed for thermal effects.

Temperatures Used in the Thermal Analysis Included:

- 480 Degrees F from Penetration No. 109 to the approximate " half-way " point (i.e. Elbow between supports EFH-8 and EFH-9).

- 590 Degrees F from the " half-way point to and including the Ring Header.

Maximum Thermal Stress & Nozzle Loads

- Maximum thermal stress from the reanalysis was 17,256 psi. This stress was located on the Ring Header and is acceptable from an allowable stress of 22,500 psi provided in ANSI B31.1.0 1967 Edition and B&W Dw'g. No. 108031, Rev. 0.

Maximum Primary Stress

- Maximum Deadweight & Pressure Stress= 5,665 psi @ node 32
  Maximum Seismic (SSE) = 11,700 psi @ node 561

  Total = **17,365** psi

This stress envelopes the maximum stresses acting on the EFW piping system inside containment. Worst case primary stress is unaffected by thermal transient.

EMERGENCY FEEDWATER PIPING / SUPPORT REANALYSIS SUMMARY

OUTSIDE CONTAINMENT

- At Original Design Temperature of 110 Degrees F

All actual stresses on piping and supports were within the allowable limits specified in ANSI B31.1 1967 edition under deadweight, pressure, OBE and SSE loading conditions. Envelope of maximum worst case pipe stress on system due to deadweight, pressure, and SSE loading conditions = **7,906** psi.

All pipe support anchor bolt safety factors were within the requirements of NRC Bulletin 79-02. (i.e. Safety Factor greater than 4 for wedge type anchors under deadweight, pressure and SSE loading conditions.

Anchor Bolt Safety Factors (SSE) for EFW pipe supports :

Support	Safety Factor	Support	Safety Factor
EFH-125	Infinite	EFH-133	10.2
EFH-126	16.0	EFH-532 (wejits)	4.8
EFH-127	12.0	EFH-542	16.5
EFH-128	27.0	EFH-543	11.3
EFH-129	17.0	EFH-544	9.9
EFH-130	8.7	EFH-545	33.0
EFH-131	14.0	EFH-546	31.0
EFH-132	4.3		

- During Elevated Temperature Condition

All actual stresses on piping were within the allowable limits specified in ANSI B31.1 1967 edition under deadweight,thermal,pressure, OBE and SSE loading conditions.

Maximum thermal stress in piping = **14, 427** psi which is less than the allowable stress of 22,500 psi permitted in ANSI B31.1 1967 Edition.

Maximum deadweight, pressure, and SSE stress in piping = **16,445** psi which is less than the allowable stress of 18,000 psi permitted in ANSI B31.1 1967 Edition. EFH-126 was modeled as an X,Z & My restraint only (4 bolts are considered for factor of safety).

Anchor Bolt Safety Factors (SSE) for EFW pipe supports:

Support	Safety Factor	Support	Safety Factor
EFH-125	5.5	EFH-133	5.6
EFH-126	7.4	EFH-532	14.5 (maxi)
EFH-127	3.5	EFH-542	4.3
EFH-128	Removed from Analysis	EFH-543	2.1
EFH-129	1.8	EFH-544	6.5
EFH-130	2.9	EFH-545	5.0
EFH-131	6.0	EFH-546	2.9
EFH-132	1.7		

EFW PIPING / SUPPORT REANALYSIS SUMMARY (cont'd.)

OUTSIDE CONTAINMENT (cont'd.)

Systems may be classed as operable on an interim basis if the factors of safety compared to ultimate strengths is less than the original design but equal to or greater than two. (Ref. NRC Bulletin 79-02 Supplement No. 1). Only two supports (EFH-129 and EFH-132) exhibited safety factors less than two during the elevated temperatures.

Actual stresses on pipe support components (baseplates) for EFH-129 and EFH-130 exceeded OBE stress allowable of **12,600** psi per ANSI B31.1 1967 Edition . Maximum OBE stress of **26,900** psi occurred on support EFH-129 . Maximum OBE stress of **18,100** psi occurred on support EFH-130. All other pipe supports effected by thermal transient maintained component (baseplate) OBE stresses within the limits of ANSI B31.1 1967 Edition which are very conservative compared to the stress allowables of **27,000** psi referenced in the AISC Code through direction of ASME Section III Subsection NF.

Maximum SSE stress of **33,100** psi occurred on support EFH-129. Although the stress is greater than the allowable 0.9 Sy (Sy = 36KSI) the stress would have been below yield. Impairment of system function or pipe rupture not achieved (i.e. yield of piping not reached.) All other supports experienced a maximum SSE stress of **20,600** psi.

OVERVIEW OF PIPING ANALYSIS EFFORT
(Outside Containment)

Piping analysis packages CR-44 and CR-46A were combined (modeled through anchor restraint EFH-132) for the deadweight and thermal analyses. The seismic analysis was defined at the separation points per the original Piping Analysis packages CR-44 and CR-46A.

The degraded pipe support anchor EFH-126 was modeled as a three directional restraint (forces Fx, Fz, and moment My) due to the shear capacity of the anchor bolts existing in the degraded condition. In addition, rigid rod support EFH-128 was removed from the piping analysis due to it's inability to resist the resulting compressive loads generated at this point.

To approximate the thermal transition that occurs within the piping, the following temperatures were modeled into the thermal analysis:

480 Degrees F - From Pen. #109 to support EFH-126 to valve FWV-34, and from the tee at analysis node 24 to the elbow above support EFH-543.

375 Degrees F - From the tee below EFH-126 to support EFH-132 continuing to and including the flange located east of EFH-133.

255 Degrees F - From the flange located east of EFH-133 to valve EFV-18.

150 Degrees F - From the elbow above support EFH-543 to support EFH-532 and from EFV-18 to support EFH-83.

70 Degrees F - All other affected piping outside containment.

Piping Analysis Results (Stress Summary)

Primary Stresses

Maximum Deadweight and Pressure Stress	= 7271 psi	@ Node 1181
Maximum Seismic (SSE) Stress	= 9174 psi	@ Node 190
	16,445 psi	

Secondary Stresses

Maximum Thermal Stress	= 14,427 psi	@ Node 1183

EMERGENCY FEEDWATER PENETRATION ANALYSIS SUMMARY

Initial Design Basis

The containment penetration for the emergency feedwater piping consists of a 14-inch sleeve, 2-inch thick containment side closure plate, 3/4 inch exterior closure plate and a 6 inch schedule 160 process pipe. Containment boundary is formed by the 2-inch closure plate and full penetration welds at the process pipe and penetration sleeve interface to the closure plate. The exterior closure plate was provided to facilitate initial penetration functional testing at construction.

Initial design of the penetration assembly and anchorage will withstand the developed full plastic moment capacity of the process pipe. Axial and shear loads were taken as a force equal to the system pressure times the full flow area of the process pipe. For penetration No. 109 these values are:

Mu = 1,002,000 in-lbs
Vu = 32,600 lbs
Pu = 32,600 lbs
Tu = 1,090,000 in-lbs

Temperature Transient Evaluations (End Plates)

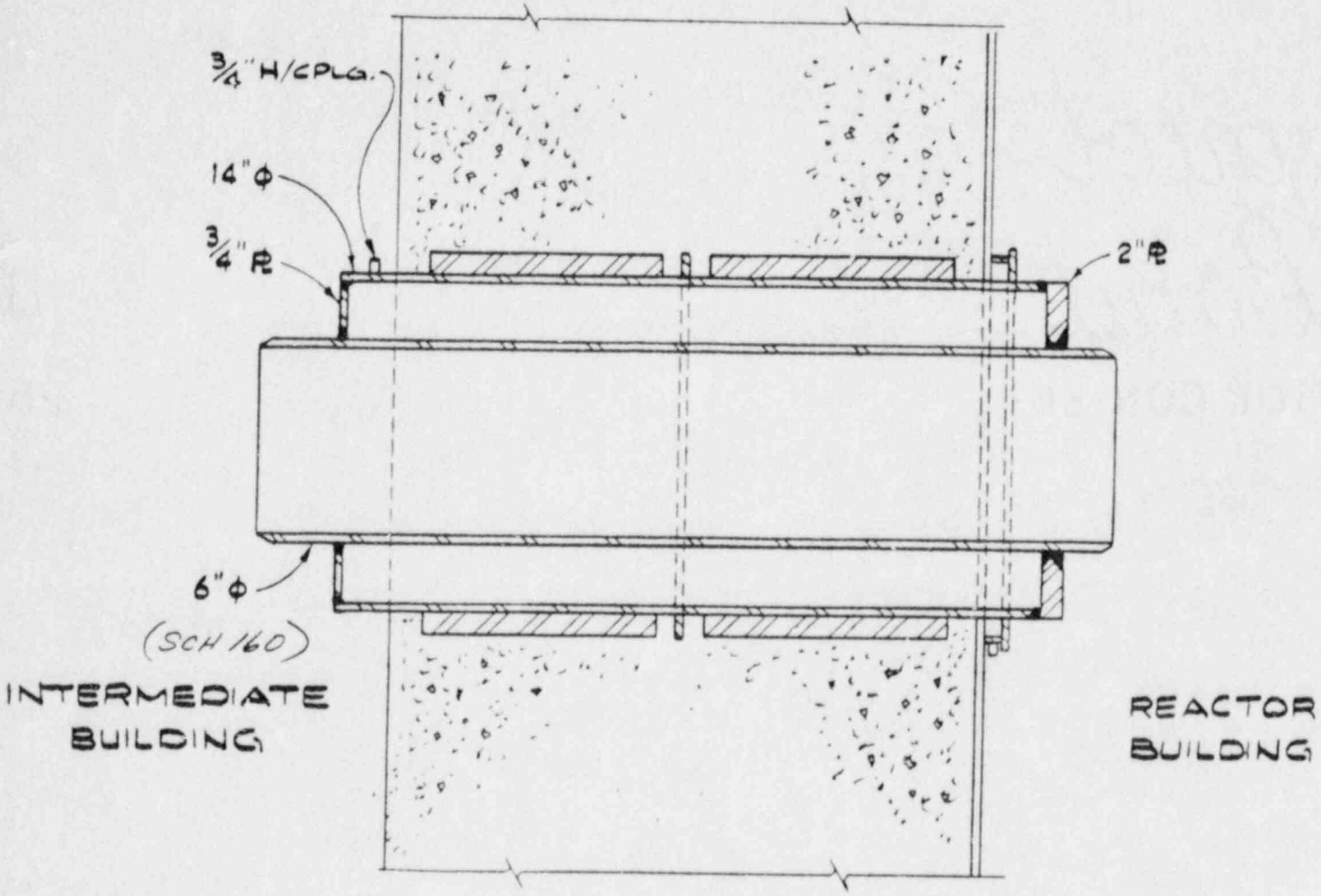
Effects of the thermal transient on the penetration were evaluated by considering an effective differential temperature between the containment sleeve and process pipe of 300 degrees Fahrenheit. The actual differential temperature was measured at 208 degrees Fahrenheit during the thermal transient event.

The closure plate flexibility was calculated to define the axial restraint applied to the process pipe within the boundary of the penetration assembly. Initial elastic analysis results indicate development of an axial load of 130,700 lbs. Corresponding stresses within the 2-inch closure plate were determined to be within code limits. The resultant stresses in the 3/4 inch closure plate indicated that elastic capacity was exceeded and plastic response of the 3/4 inch closure plate would be experienced. Axial forces developed in the process pipe when plastic response of the closure plate starts is approximately 28,000 lbs. The corresponding stress in the 2 inch closure plate at this load is 6,000 psi. External piping system loads induce an additional stress of 5,000 psi in the closure plate. Therefore, a total stress of 11,000 psi was therefore induced on the 2 inch closure plate.

OVERALL PENETRATION ASSEMBLY PERFORMANCE

With the exception of the exterior closure plate on the Type III (cold) penetration, there are no physical differences between the Type II (hot) penetration and the Type III penetration. The penetration sleeve details, concrete anchorage methods, and the containment liner penetration joint arrangement are identical for the two classes of penetrations. Maximum process pipe temperatures for the Type II penetrations are 600 degrees Fahrenheit which are higher than measured thermal transients for the emergency feedwater system (measured at approximately 195 degrees Fahrenheit) . **Based upon these similarities, the Type III Penetration No. 109 has not been subjected to any loading beyond the original design basis.**

A detailed steady-state thermal analysis of the containment structure to penetration interface was performed. Based upon a process pipe temperature of 480 degrees Fahrenheit, maximum concrete temperatures are 214 degrees Fahrenheit. The concrete temperatures were below 200 degrees Fahrenheit within 3 inches of the penetration sleeve to containment structure interface. These concrete temperatures are in agreement with Appendix A of ACI 349/359 for localized concrete temperatures adjacent to penetration areas which allow for 200 degrees Fahrenheit for long term operation and 350 degrees Fahrenheit for short term emergency considerations.


3/4" H/CPLG.
14"Φ
3/4" PL
2" PL
6"Φ
(SCH 160)
INTERMEDIATE BUILDING
REACTOR BUILDING