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Safety Evaluation to Support High Flux Isotope Reactor Startup Following Unreviewed Safety Question Involving Discovery of Check Valve-Induced Water Hammer

October 17, 2003

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Rev. 0

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Oak Ridge National Laboratory

Research Reactors Division
High Flux Isotope Reactor

Safety Evaluation to Support High Flux Isotope Reactor Startup
Following Unreviewed Safety Question Involving
Discovery of Check Valve-Induced Water Hammer

Rev. 0

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Executive Summary

In May 2002, the High Flux Isotope Reactor (HFIR) experienced an event involving discovery of actuated rupture disks in the primary coolant system, as described in Occurrence Report ORO—ORNL-X10HFIR-2002-0006 (Ref. 1). Investigation of this occurrence identified several possible causes of the rupture disk actuation, and corrective action was taken to address some of the causes. The most likely candidate for that event was thought to be air in the pressure relief system line, which could cause a normal pump transient to result in high pressure at the rupture disk due to acceleration of the water in the line against a compressible air volume. The combined impact of the moving water column and the normal pressure wave was thought to have caused the rupture disks to actuate. The air in the system was linked to the extended time and degree to which the primary coolant system was open during the beryllium outage, and the fact that the original HFIR design did not provide a high point vent at the rupture disk location. A rupture disk vent line was added to the system to correct this potential cause of the event. In addition, instrumentation was added to the pressure relief valve line for the purpose of monitoring the inlet pressure to the rupture disks.

In August of this year, the newly installed instrumentation recorded high pressures following trip of a primary coolant pump as part of the reactor startup process prior to operating cycle 396. Evaluation of the pressures recorded during that transient indicated a check valve-induced water hammer of the primary coolant system had occurred. The reactor was shutdown, and the rupture disks were inspected and found to be unaffected by the transient. Cycle 396 was completed, based on testing which indicated that the check valve-induced closure was a mild water hammer event, and more testing was planned for the Cycle 396 outage. Additional testing during the outage confirmed the existence of the check valve-closure event and provided indications that the characteristics of the check valve-induced water hammer could involve a magnitude and frequency sufficient to result in a possible negative impact on the reactor pressure vessel and primary coolant piping safety bases. The possible effects on the vessel and piping safety bases resulted in declaration of a positive unreviewed safety question (USQ) because of the tendency of the change to decrease the margin of safety. If correct, the magnitude of the indicated pressures at the rupture disk measurement point implied an increased probability of failure of the rupture disk and the possibility of a new mode of malfunction of the rupture disks, which is caused by water hammer following a primary coolant pump trip. The effect on the rupture disk was an additional cause for a positive USQ declaration. The events from August of this year, and declaration of the positive USQ are documented in Occurrence Report number ORO—ORNL-X10HFIR-2003-023 (Ref. 2).

In response to the positive USQ, ORNL management prepared the following plan for resolution of the problem.

1. Obtain an industry expert and internal subject matter expert advice on the Research Reactor Division (RRD) personnel's understanding and response to the problem.

2. Perform preliminary analysis of the event to conservatively estimate the magnitude of the water hammer transient.
3. Prepare an assessment of the impact of the discovery on the vessel and piping safety bases, based on the preliminary analysis.
4. Prepare an assessment of the impact on the rupture disk safety basis and Technical Safety Requirements, based on the preliminary analysis.
5. Develop and test interim corrective measures (new pump shutdown procedures) to mitigate the water hammer.
6. Provide a justification for continued operation based on the short term analyses, testing, and interim corrective measures.
7. Long term, prepare a hydraulic model of the HFIR system to provide a comprehensive safety basis for long term corrective measures.
8. Long term, perform additional HFIR system testing for code validation and final determination of the effect on the vessel and piping as necessary.

This safety evaluation documents the results of work done to carry out the short term portions of the plan described above, and includes a description of the longer term work planned to complete the response to this discovery.

Chapter 1 of this report provides the detailed background and introduction to this discovery. Chapter 2 of the report provides additional detail about water hammer phenomena, and provides discussion of applicable water hammer issues at HFIR. This chapter includes meeting minutes with the water hammer expert who has been subcontracted to provide advice and analysis support on this problem.

In Chapter 3, the plant data that was recorded at the beginning of Cycle 396 is provided to show that this discovery is clearly linked to pump trip transients, involves closure of the swing check valve located at the discharge of the pump, and usually occurs when the first of three normally operating pumps is tripped. Additional data recorded during the 396 outage is provided, using a fast recording of the pressure trace following a pump trip, and shows that the measured pressure transient varies in a damped sinusoidal fashion with a maximum amplitude of approximately 660 psig, a minimum amplitude of 280 psig, and a frequency of about 40 cycles/s. The principal interim measure to mitigate the water hammer at the check valve is to simultaneously trip all three primary coolant pumps, rather than trip them sequentially. Using this procedure, there are no longer 1 or 2 operating primary coolant pumps to provide the back-pressure involved in the check valve-closure. Data recorded following a simultaneous trip of all three pumps is shown in Chapter 3, indicating no water hammer associated with this procedure.

In Chapter 4, results of a preliminary analysis is described that uses a textbook method for estimating the magnitude of the peak pressure developed during a check valve-closure event, which for the HFIR system resulted in a delta-pressure of 306 psi, or a total pressure of 810 psig at the check valve seat. The preliminary analysis also involved a textbook estimate of the transmission factors for wave propagation of the pressure transient as it moves toward the vessel, which indicated that the reactor pressure vessel should see a positive pressure pulse no greater than 58 psi, and a negative pressure pulse no less than -55 psi.

In Chapter 5, the preliminary analysis results of Chapter 4 were provided to the engineers who performed the reactor pressure vessel integrity and life extension analyses (Refs. 3 and 4) for evaluation of this discovery's effect on the safety of the pressure vessel. Since check valve-induced water hammer is assumed to have existed in the HFIR system since initial operation, there was a concern about the long term cumulative effect of the pressure cycling on the vessel integrity. The primary concern in this evaluation is the effect on the flaw growth due to high cycle pressure oscillations. Using a ± 58 psi transient with a 50 Hz frequency as the representative transient for each pump trip, the effect on the vessel was found to be insignificant. All conditional probability of failure criteria are satisfied and TSR vessel-related limitations and surveillance requirements remain valid.

In Chapter 6, the preliminary analysis results of Chapter 4 were provided to engineers who performed the primary coolant system piping analysis for evaluation of this discovery's effect on the piping safety basis. The piping safety analysis (Ref 5) determined that the probability of a large break loss of coolant accident (LOCA) was so small that it should be considered beyond design basis. The concern in this evaluation is that the water hammer event presents a high cycle vibration fatigue effect on the piping over the life of the plant, such that the probability of large break LOCA is no longer beyond the design basis. Using a zero to 900 psi magnitude, with a frequency of 50 Hz as the bounding transient for each pump trip, the effect on the piping failure probability was found to be an increase that leaves the probability of a large break LOCA still very small and well within the realm of being considered beyond design basis.

As described in Chapter 6, the conservative inputs utilized in these preliminary evaluations have resulted in more significant calculated flaw growth in the piping material than in the vessel material. The hydrostatic proof test that is used to prove the integrity of the HFIR vessel is also used to prove the integrities of all the longitudinal welds and about 65% of the circumferential welds in the 3-inch and larger piping until the next hydrostatic proof test is conducted (Ref 6). The remaining 35% of the circumferential welds receive inservice ultrasonic and penetrant testing using the guidance of the ASME Code (Ref 7). A larger flaw growth may show that some welds that have been shown to be proven by the hydrostatic proof test are no longer covered by that test, and perhaps more welds may require ultrasonic and penetrant testing at the next inservice inspection (ISI) of the welds under the ISI Program (Ref 8). The validity of the previous determinations will be evaluated following completion of the detailed plant modeling, as described in Chapter 9.

In Chapter 7, analysis is provided which indicates that the pressures measured at the rupture disk location are believed to be adversely affected by the dynamics of the pressure measurement line. This assumption is supported by initial review from an outside expert (Appendix A), who is providing support to the RRD analysis efforts. A resonance analysis of the measurement line is being prepared to support the assumption that the indicated pressure is at a frequency and magnitude that may be much higher than what actually occurs in the 20-inch line. Thus, using 50 Hz as the transient's frequency in the above analyses is conservative. Discussion is also provided regarding the impact of these high-cycle transients on the functionality of the rupture disks and their ability to continue to adequately protect the reactor vessel from an over-pressurization event.

In Chapter 8, a description of the interim operational and procedure changes is provided. The main change made to mitigate the check valve-induced water hammer is the simultaneous trip of all three primary coolant pumps during normal end of cycle operations to shut down the pumps. In the event of an unexpected or emergency trip of a primary coolant pump during reactor operation, the reactor will be shutdown and the rupture disks will be replaced. Future system modeling and/or testing as described in Chapter 9 will be used to develop proposed changes to the interim measures.

In Chapter 9, plans for long term analysis and testing are outlined. A detailed method of characteristics model of the HFIR system is under development by the consultant, and is planned for use in final analysis of the effects on the vessel and piping. In addition, more plant testing may be required to better resolve the question about rupture-disk-sense-line resonance, determine the magnitude and frequency of the pressure pulses, and provide data for computer model validation.

Chapter 10 provides conclusions that indicate that it is safe to continue operation of the reactor. The three questions that were defined as being "positive" in terms of identifying conditions that were potentially not bounded by the current safety analysis have been analyzed and found to in fact be bounded. The pressure transient does not challenge the conditional probability of failure of the vessel and piping. Rupture disk actuation has been found to be only affected in a conservative manner relative to this transient. Replacement of the rupture disks will continue per current TSRs if the transient is observed again. The margin of safety currently defined in the DOE approved USAR and TSRs is preserved.

Based on this evaluation it is safe to continue operation of the reactor. Upon receipt of DOE's approval of this safety evaluation, the safety basis covering operations will constitute the currently approved USAR, TSRs, and commitments included in this report.

The duration of operations bounded by this evaluation is not set by any of the various analyses. RRD believes that operations under this evaluation can continue for the period necessary to incorporate this new analysis into a formal USAR and TSR update that will include the final analysis discussed in Chapter 9. This update will be submitted to DOE no later than that of the next annual update submittal, currently scheduled for April of 2004.

1.0 Introduction

In January 2003, redundant pressure transducers were installed in the High Flux Isotope Reactor (HFIR) primary coolant system near the rupture disks, which are in series with the primary system relief valves. These monitors were installed in response to an event where rupture disks were found to be ruptured (ORO—ORNL-X10HFIR-2002-0006, 5/24/2002) to more accurately measure rupture disk pressure. Although the definitive reason for the burst disks was not identified, it was concluded that the most likely cause was trapped air that occurs in the branch line to the rupture disks, with no mechanism for venting. This trapped air bubble could magnify the pressure increase in the branch line to the primary relief valve rupture disks while the system pressure indication would not sense the same pressure spike. A design change, DCM HFIR-235M, *"Primary Coolant System Rupture Disk Safety Head Replacement,"* was implemented to allow venting of the space under the rupture discs. The installation of the rupture disc pressure recorder/transducers was also in response to one of the corrective actions from the subject occurrence report.

These pressure transducers were installed as part of the rupture disc pressure monitoring system specified in Design Change Memorandum (DCM) HFIR-186M. The rupture disc pressure monitoring system was designed to provide an accurate indication of rupture disc pressure following a closure of the letdown block valves. The closure of the letdown block valves combined with the "booster" effect of the main pressurizer pumps on the suction of the primary coolant pumps can lead to pressures that can actuate the primary pressure relief system. The transient pressure response of the primary system during this event formed the design basis for the rupture discs pressure monitoring system, which lead to a recommended sampling rate for the rupture disc pressure monitor of 500 msec. The 500 msec represents the minimum sampling frequency (maximum time between samples) to resolve the pressure increase as a result of the "booster" pump effect. The currently installed recorder samples at 8 times per seconds (once per 125 msec) which meets the original design basis.

On August 12, 2003, an operator on mid-shift checks found one of the two pressure recorders in alarm, indicating it had exceeded the set point of 560 psig. The highest recorded pressure was 578 psig. Operating procedures require an evaluation when the rupture disc inlet line pressure exceeds 560 psig. Several action steps were outlined—one of which was to verify that the 578 psig indicated value was the highest pressure reached. Several analytical methods were used, but none could conclusively verify that the peak pressure could not be higher than the deformation value of 588 psig. The 578 psig indication was believed to have been a result of entrapped air under one of the pressure transducers amplifying the pressure during primary coolant pump start/stop transients.

At 12:30 on August 14, the reactor was shut down to physically inspect the rupture disks for deformation. No deformation was noted on the rupture disks. To ensure accurate pressure monitoring a modification to the pressure transducers and sensing lines was performed to reduce the probability of entrapping air in the sensing lines that may have caused amplified pressure readings. The reactor was restarted, with a plan for further evaluation and testing of pumps and transient information during the next outage.

This additional testing was initiated on Saturday, September 13th, during EOC 396B. During this testing a high speed data recorder, used to gather transducer signal information was operated with a "filter" in both "on" and "off" positions. Previous testing utilizing the recorder was done with the filter "on", which was thought to alter the signal and possibly the monitored transducer information.

During the testing both transducer outputs (channels) were verified to respond as they had in the past with the filter "on." With three pumps running, steady state monitoring of both channels indicated pressures of ±5 psig between channels and the baseline of the process monitoring equipment. One channel was then monitored with the filter "off," giving steady state pressure differentials of ±20 psig at approximately 3-msec cycle times with three pumps running.

A subsequent pump shutdown test was run, with one channel in the filter "on" mode, one channel in the filter "off" mode. On the channel with the filter "on," pressure indication during the transient was monitored at 480 psig peak, and 450 psig minimum pressures. The channel with the filter "off" recorded the transient at greater than 600 psig peak (recorder scale set to 600 psig max) and less than 350 psig minimum (350 psig low pressure scale set point). The separate primary vessel pressure indication as well as the other transducer channel indicated pressures of less than 500 psig (observed during test), which was consistent with previous tests described earlier. Throughout the testing the closure of primary coolant pump exit check valves was heard by the test crew, coincident with observed transient pressure readings on the instrumentation. Audible closure of the check valve is a normal response of the system.

The testing was secured, as the indication of pressure greater than 560 psig was a hold point in the work package. Aside from the recorded pressure trace from the unfiltered channel, there were no other indications of abnormally high or low pressures during the event. Checks of the permanently installed rupture disk pressure instrumentation did not indicate an alarm (less than 560 psig), and no rupture disk burst indication was noted (downstream pressure was still at 0 psig). The high primary system pressure cutoff switches for the pressurizer pump did not actuate. These cutoff switches have a setpoint of 540 psig and are located just downstream of the rupture disc inlet line. In addition, no low pressure annunciators alarmed, no letdown block valve closures occurred, and no low pressure scrams were indicated as a result of the unfiltered pressure trace. The pressure transient was not observed on the direct pressure vessel (104) monitoring system or the pressure control (127) system.

The response time of the other plant pressure monitoring instruments mentioned in the paragraphs above (PT-104, PT-127, etc.) are not sufficient to detect transients on the time scale of water hammer events. Their circuitry would/could not respond to a 45 Hz transient. The fact that these instruments did not indicate an event cannot be used as evidence that the water hammer did not occur.

Subsequent to this test a check of possible induced signal noise in the test leads was investigated. With a resistive load applied to the open test leads, no signal noise was seen which indicated the ±20 psig at steady state was not test induced signal noise. This could be attributed to other system noise, such as pump impellor blade pass frequency.

During initial conversations with SENSOTEC, the manufacturer of the transducers used for the primary coolant system rupture disk pressure monitors, it was noted that the transducers in question were not designed to measure pressure waves induced by the starting and stopping of the primary coolant pump AC motors. SENSOTEC noted it would be very difficult to explain transducer behavior when subjected to such rapid transients (i.e., wave traveling at the speed of sound in water: up to ~5000 ft/sec) and that data gathered during the transient testing was not considered to be reliable.

During follow-up conversations with SENSOTEC (and examination of the vendor technical specifications), it was determined that the bandwidth of the 3-wire SENSOTEC transducers installed in the rupture disc pressure monitoring system is 2500 Hz; therefore, they can detect a period of 0.4 msec or a half period of 0.2 msec. Thus, the unfiltered data can not be discounted and it is possible that the rupture discs are exposed to pressures greater than their deformation pressure each time a main coolant pump is stopped when another coolant pump is driven by an AC-motor.

Additional testing was performed on October 13th to determine the magnitude of the pressure oscillations following a check valve closure and to investigate an alternate means of shutting down the pumps to avoid the check valve-closure transient. During this testing the high speed data recorder was operated with the "filter" for both channels "off." The results of this testing are provided in Chapter 3 of this report.

2.0 Hazards Associated with Water Hammer

Water hammer is a form of unsteady flow that can occur in water systems when there is a sudden change of velocity (magnitude, direction, or both) of the water. The sudden change in momentum of the fluid creates rapid changes in pressure, which move away from the source of the disturbance at sonic velocity and interact with the system at each point in the piping where a change in direction, expansion, contraction, or flow obstruction exists. Since the acoustic velocity in liquid water systems typically ranges from 4000-5000 ft/s, the pressure transient in a small closed system like the HFIR occurs very fast, and is typically over (the waves reflect, rarefact, and are damped-out by friction and system expansion) in about a second. The pressure waves and their induced velocity changes are superimposed on the existing flow field, whether it is a steady state or transient flow field. The amplitude of the pressure changes in the system is important because the stresses that can be created in structures and components when impacted by the pressure wave can be high enough to cause damage, and the reaction forces on component supports can be sufficient to exceed design margins and/or cause damage. The frequency of the pressure waves is important because it is used in the piping and pressure vessel analysis to assess the long term fatigue growth of cracks due to the pressure cycling over the life of the plant.

The classical causes of water hammer in thermal/hydraulic systems are listed below.

1. Valve Operations. This can involve check valve closure or rapid control valve motion. A good example of this type of water hammer is the letdown block valve closure events for HFIR, which occur whenever the radiation block valves close.

2. Pump Transients. This can involve pump startup, which can cause a surge in the system as a packet of rapidly compressed water immediately downstream of the pump acts as a piston to set up a pressure wave that moves around the system. Operational experience has shown that pump startup transients at HFIR are mild pressure transients involving about a 50 psi pressure change over the steady state pressure distribution. Pump trips are an important transient for systems with parallel pumps because, upon trip of a single pump, the running pumps in the system can cause reverse flow through the decelerating pump or rapidly shut its exit check valve and cause a water hammer event.

3. Formation and Collapse of Vapor Pockets. Good examples of this type of water hammer are (1) liquid column separation caused by fluid momentum downstream of a suddenly closed valve causing local flashing and resultant high pressures as a combination of condensation and reversed water velocity impacting the closed valve (2) pump cavitation and sudden vapor bubble collapse when subcooled water is drawn into the pump, which in some cases can damage or destroy the pump.

4. Pressurization of Entrapped Gas. This can occur (1) downsteam of a pump following startup if a noncondensible gas pocket exists sufficient to allow more

acceleration of the fluid than normal, or (2) in dead-end pipe reaches when transient pressure pulses enter that section of piping. This was thought to be a possible cause of the 2002 HFIR event that involved discovery of broken rupture disks. The design changes to provide air vents at the rupture disks and instrumentation to measure peak pressure at the rupture disks, provided new information that lead to the determination of an unreviewed safety question regarding check valve-induced water hammer that is the subject of this safety analysis.

5. Valve Oscillation. This can occur if a valve changes position in a rapid periodic fashion, sufficient to create a sinusoidal pressure excitation of the system. This was thought to be one possible explanation of the rupture disk actuations in 2002, based on the idea that when a cell block valve is slightly opened with the primary coolant pump running, the block valve gate could oscillate in phase with the pump driving it from upstream, or with the water turbulence driving it on the downstream side of the gate. Check valve fluttering is a subset of this water hammer category, and is considered to be a possible but not likely explanation for the measured response of the PI-151/152 pressure sensors. The flutter could be caused by the creation of a negative pressure pulse on the back side of the check valve, followed by reflection off the spinning pump impellor, followed by reopening of the check valve, then closure of the check valve by system pressure, and repeat of the cycle until the transient dies out.

Theoretically, the pressure oscillations resulting from an instantaneous change in velocity can have a magnitude of,

$$\Delta p = \rho a \Delta v,$$

where

Δp is the magnitude of the pressure rise, ρ is the fluid density, a is the acoustic velocity, and

Δv is the sudden velocity change of the moving stream.

For example, the pressure increase upstream of a letdown block valve as a result of the closure of the block valve can be estimated by substituting known values for density (ρ), the speed of sound (a), and assuming a flow rate of 40 gal/min in a 2-inch SCH 40 pipe yields:

$$\Delta p = (62 lb_m / ft^3)(5000 ft/s)(3.83 ft/s)(ft^2/144 in^2)\frac{lb_f - s^2}{32.2 lb_m - ft} = 256\frac{lb_f}{in^2}$$

The above equation could also be utilized to generate a conservative upper bound on the pressure increase as a result of a closure of a main coolant pump discharge check valve. Assuming an instantaneous decrease in coolant velocity from 20 ft/s to zero through the check valve yields a pressure increase of approximately 1350 psi. This estimate is unrealistically conservative since it assumes an instantaneous cessation of all flow through the check valve. In reality, the actual pressure pulse generated as a result of the check valve closure is a result of the cessation of whatever negative flow though the check valve is created as the pump is coasting down. A more realistic, yet still conservative, estimate of the peak pressure following a primary coolant pump exit check valve closure is documented in RRD Calculation C-HFIR-2003-035 and discussed in greater detail in Chapter 4.0 of this report.

3.0 HFIR Plant Data on Water Hammer

The pressure trace in Figure 3-1 shows the data downloaded from recorder PR-150 following the August 12, 2003 event when an operator on mid-shift checks found one of the two pressure recorders in alarm, since it had exceeded the set point of 560 psig. The highest indicated pressure was 578 psig.

The following section will summarize the data taken on October 13^{th} 2003 as part of Revision 1 of MWP 36342. As discussed earlier this additional testing was performed to determine the magnitude of the pressure oscillations following a check valve closure and to investigate an alternate means of shutting down the pumps to minimize check valve-induced water hammer. The magnitude of the pressure pulse required quantification since the pressure trace during the September 13^{th} testing had gone off-scale high (600 psig) and low (350 psig).

During the testing on October 13^{th} the high speed data recorder was operated with the "filter" for both channels "off."

A scanned image of the recorded pressure trace from PI-152 following the trip of PU-1C with both PU-1A and PU-1B running is shown in Figure 3-2.

The vertical axis is PI-152 pressure from 0 to 1000 psig, each major division is 250 psig and each minor division is 50 psig. The horizontal axis is transient time, the speed of the chart recorder was 200 mm/sec or 5 ms/mm, therefore each minor division (block) represents 25 ms. This information applies to all the figures 3-2 through 3-5.

Upon examination of the data in Figure 3-2 it can be seen that the peak pressure indicated following the trip of PU-1C with both PU-1A and PU-1B running was about 660 psig and the minimum was about 280 psig.

Immediately following the PU-1C trip event discussed above, the pressure transient following the start of PU-1C with PU-1A and PU-1B running was recorded. The recorded trace from this event is shown in Figure 3-3. From Figure 3-3, the peak pressure indicated following the start of PU-1C with both PU-1A and PU-1B running was about 545 psig and the minimum was 470 psig. This figure confirms the plant operating history which indicates that pump start transients introduce very mild pressure transients in the system.

The next test run was the simultaneous trip of all three operating AC-motors utilizing the low-pressure trip circuitry, the trace of which is included as Figure 3-4. As a result of the chart speed it is difficult to see the entire transient on a single sheet of paper. However, it is clear from Figure 3-4 that the indicated pressure is always decreasing indicating that no significant check-valve closure initiated water hammer occurred.

A portion of the testing performed on October 13^{th} also included the determination of the magnitude of the pressure waves generated by a check valve closure while the other two pumps were operating at pony motor speed, as shown in Fig. 3-5. During the event, the AC-motor that is started will close the exit check valves for the other two pumps that are operating at pony motor speed. For this test, pony motors PU-1E, PU-1F, and PU-1G were operating and main AC-motor PU-1A was started. The peak pressure indicated following the start of PU-1A with PU-1E, PU-1F and PU-1G running was about 550 psig and the minimum was 410 psig. This event is not a major concern as a pressure transient.

As with previous tests, the PR-150 recorder event file was utilized to record the PI-151 and PI-152 data for comparison with pressure traces from the high-speed recorder. Figure 3-6 is the PI-150 plot corresponding to the high speed plot of Fig 3-2 for the case of PU-1C trip at full flow. Figure 3-7 is the PI-150 plot corresponding to the high speed plot of Fig 3-3 for the case of PU-1C start against the two other running pumps. Figure 3-8 is the PI-150 plot corresponding to the Fig. 3-4 case of trip of all three pumps from full flow conditions. Figure 3-9 is the PI-150 plot corresponding to the Fig. 3-5 case of startup of a single AC motor-driven coolant pump against two pony motor driven pumps.

All of the PR-150 plots cover much longer time scales (10 seconds or more) than the high speed recordings, and show system dynamics that indicate the slower action of the pressure control system. For example, Fig. 3-6 indicates the trip of the primary coolant pump at about 193 seconds, the water hammer event at 194.3 seconds and the slow return to the original pressure set point from 195 to 200 seconds as the water hammer dies out.

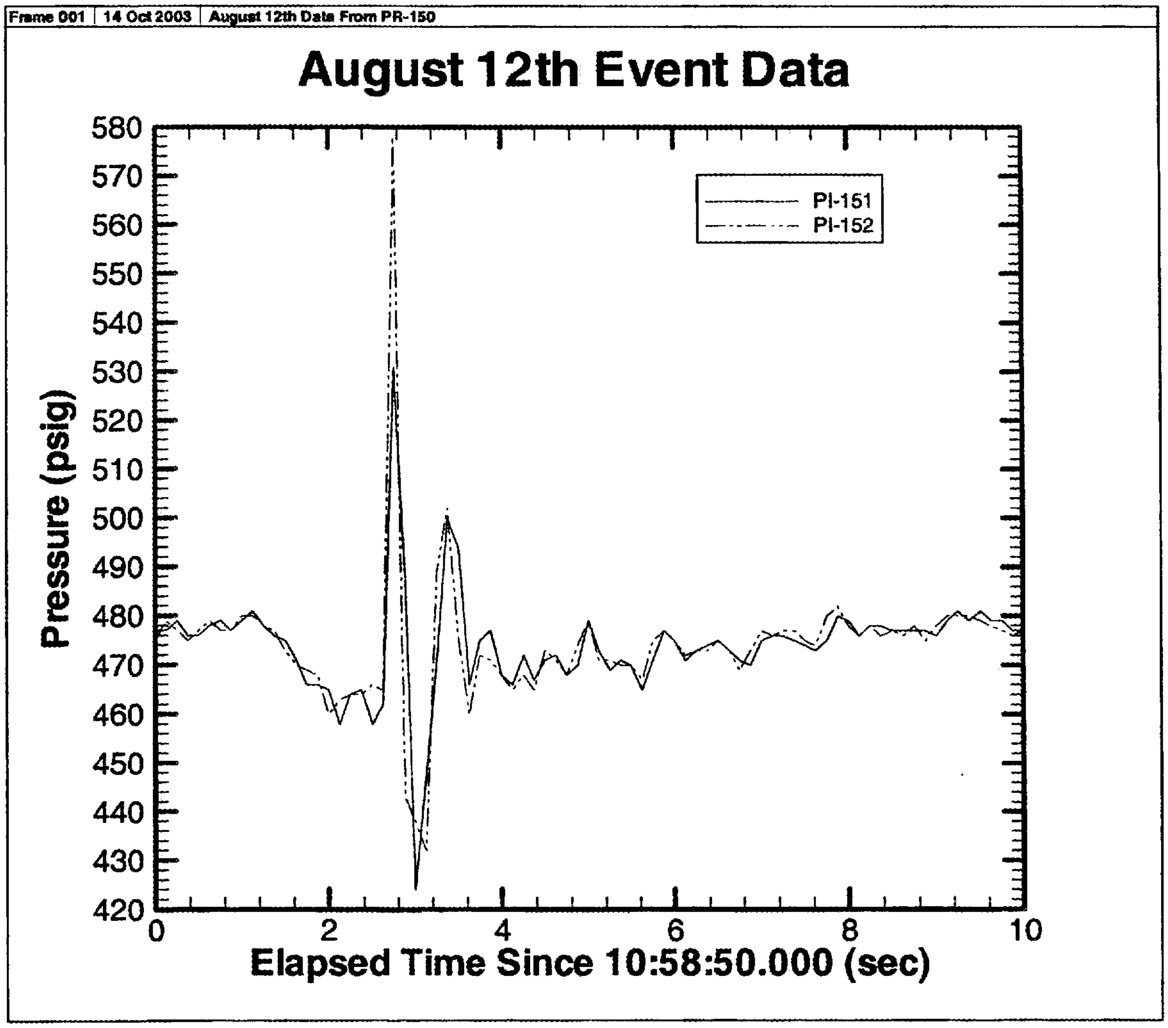

Figure 3-1. August 12th event PR-150 recorder data.

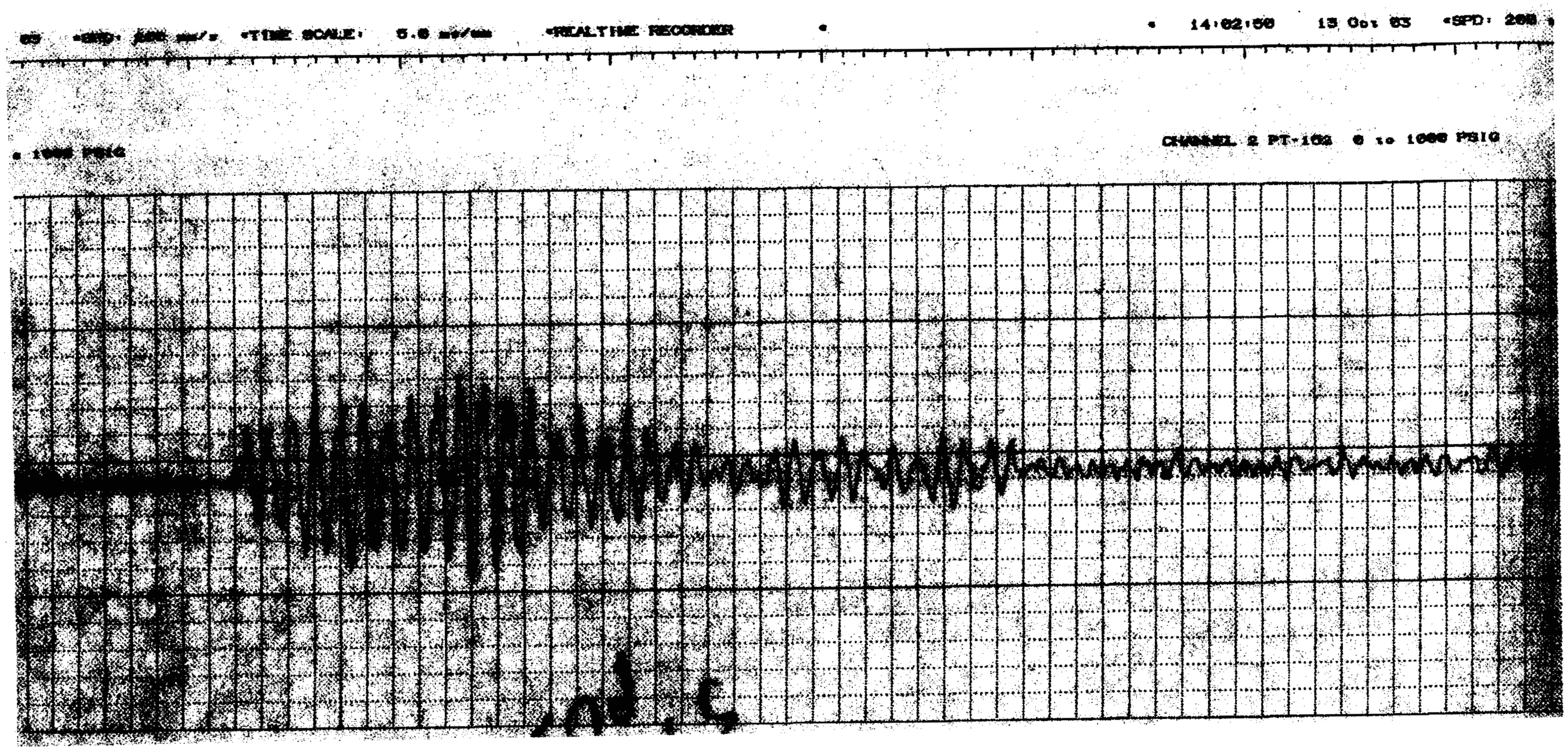

Figure 3-2. High-speed recorder data – PU-1C trip.

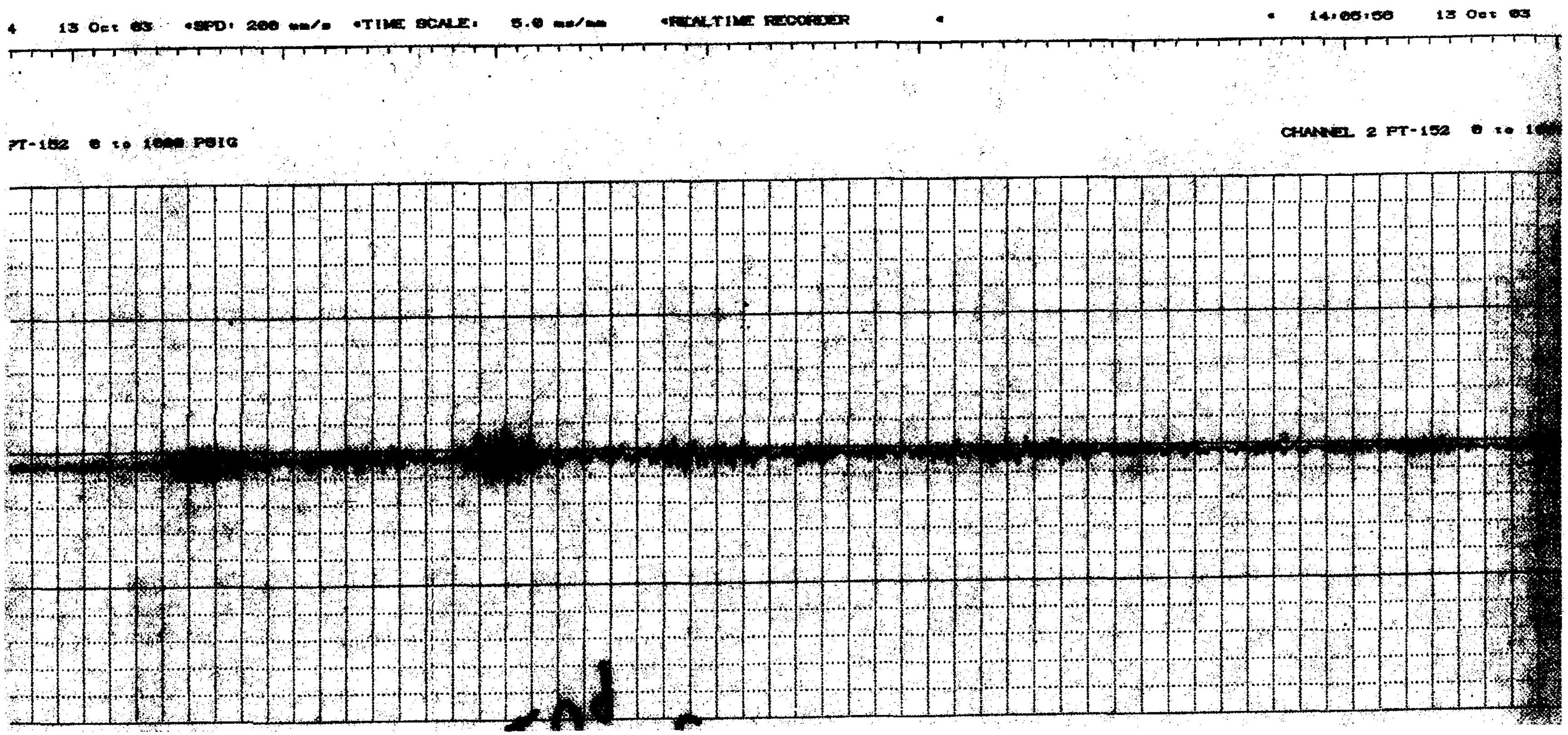

Figure 3-3. High-speed recorder data – PU-1C start.

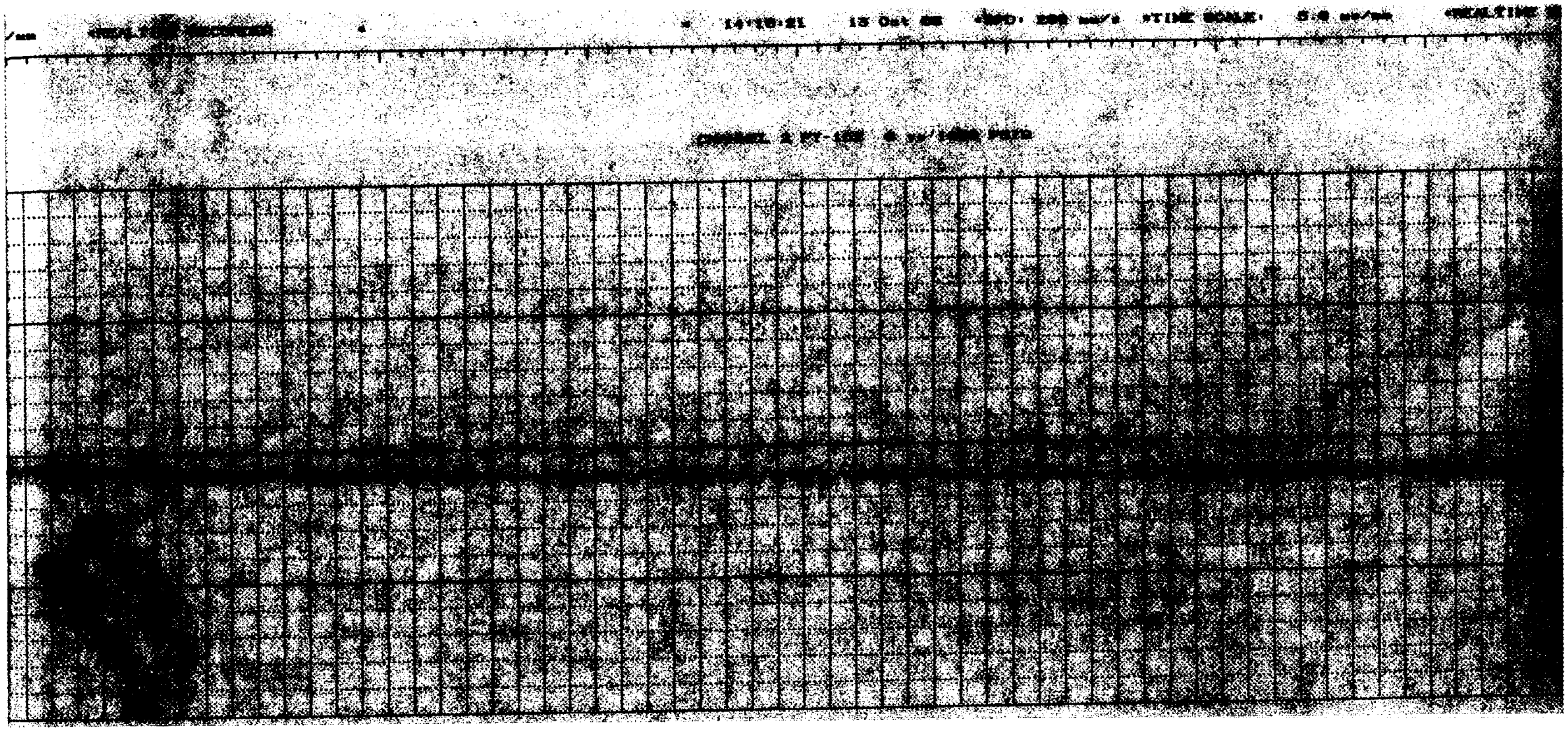

Figure 3-4. High-speed recorder data – PU-1A, 1B, 1C trip.

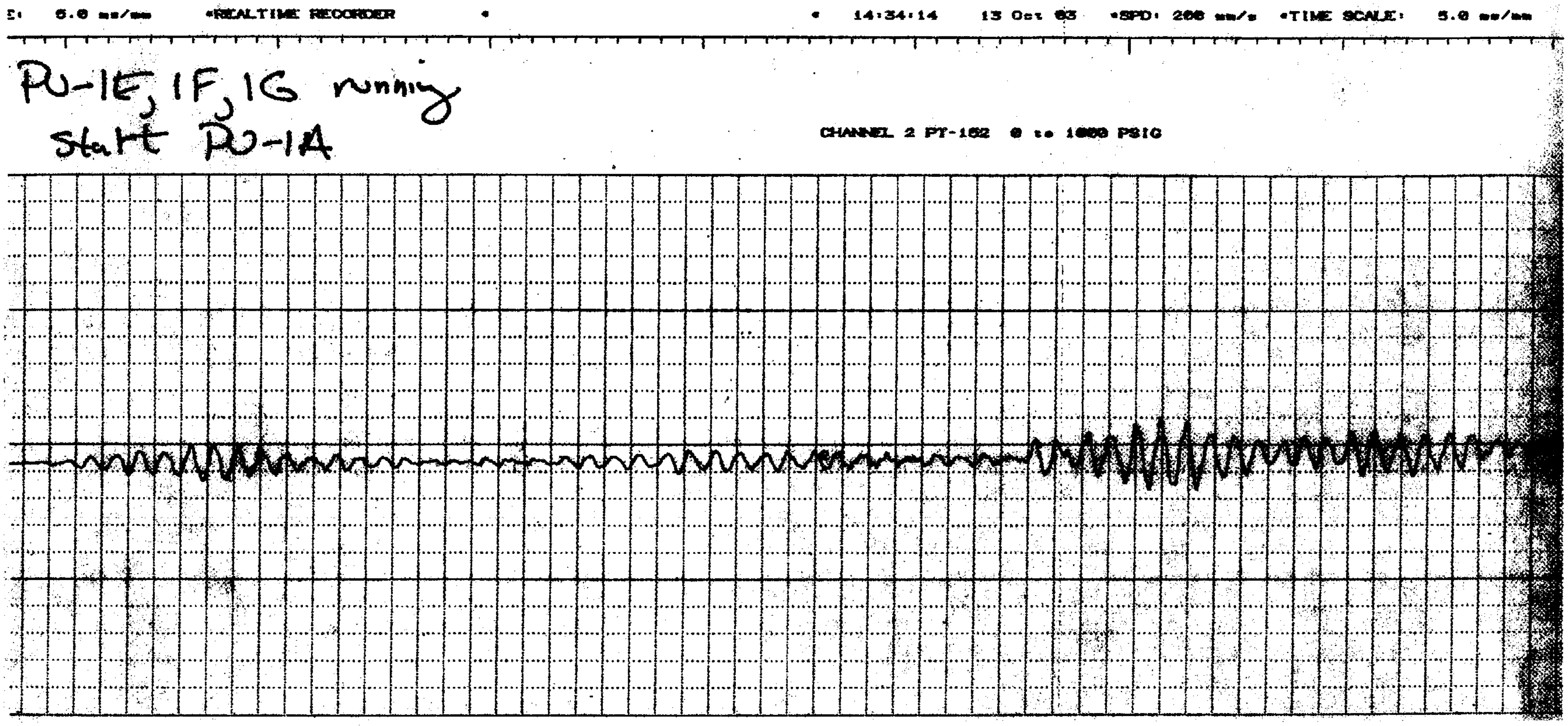

Figure 3-5. High-speed recorder data – PU-1A start.

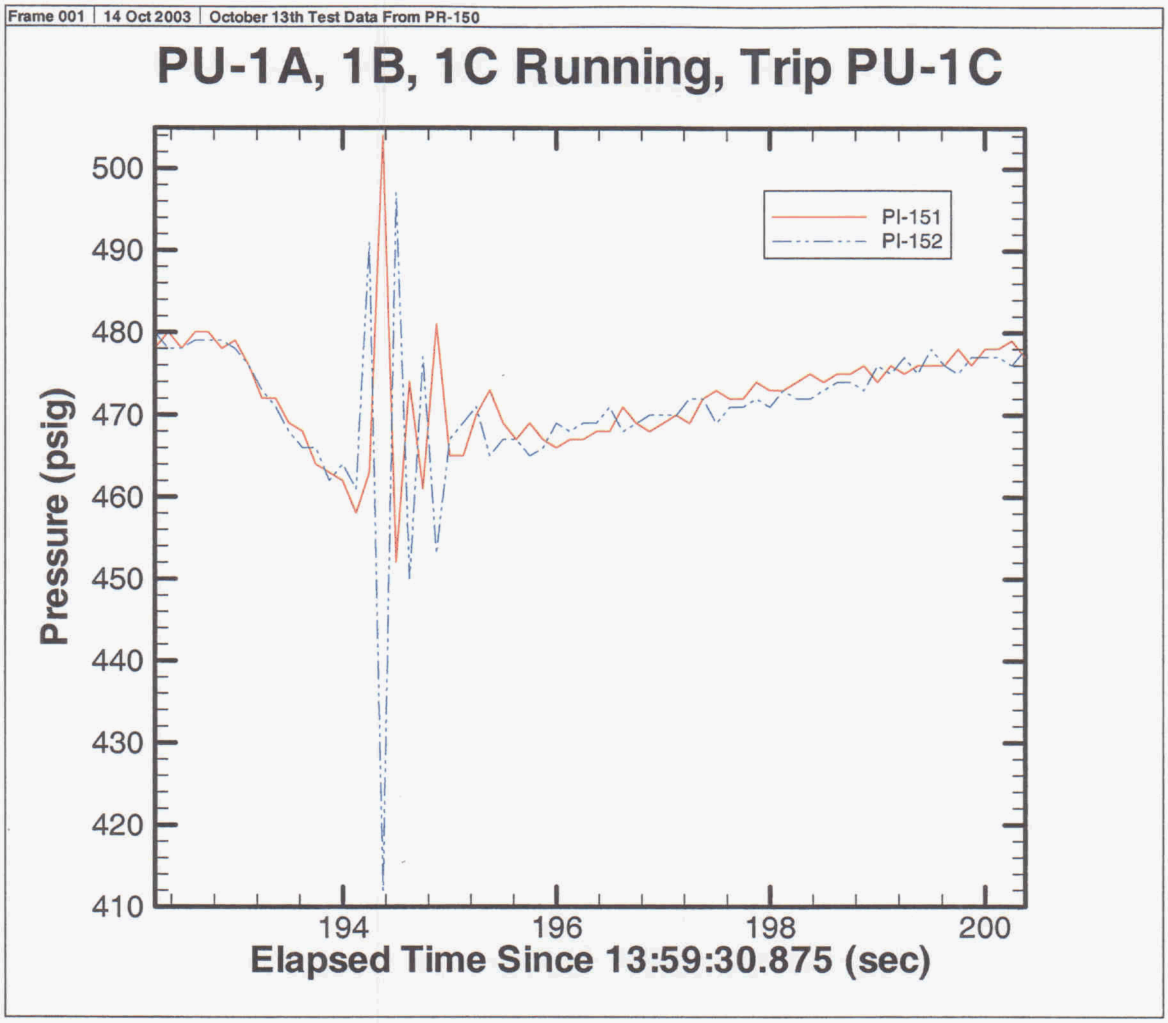

Figure 3-6. October 13th test data (PR-150) – trip of PU-1C.

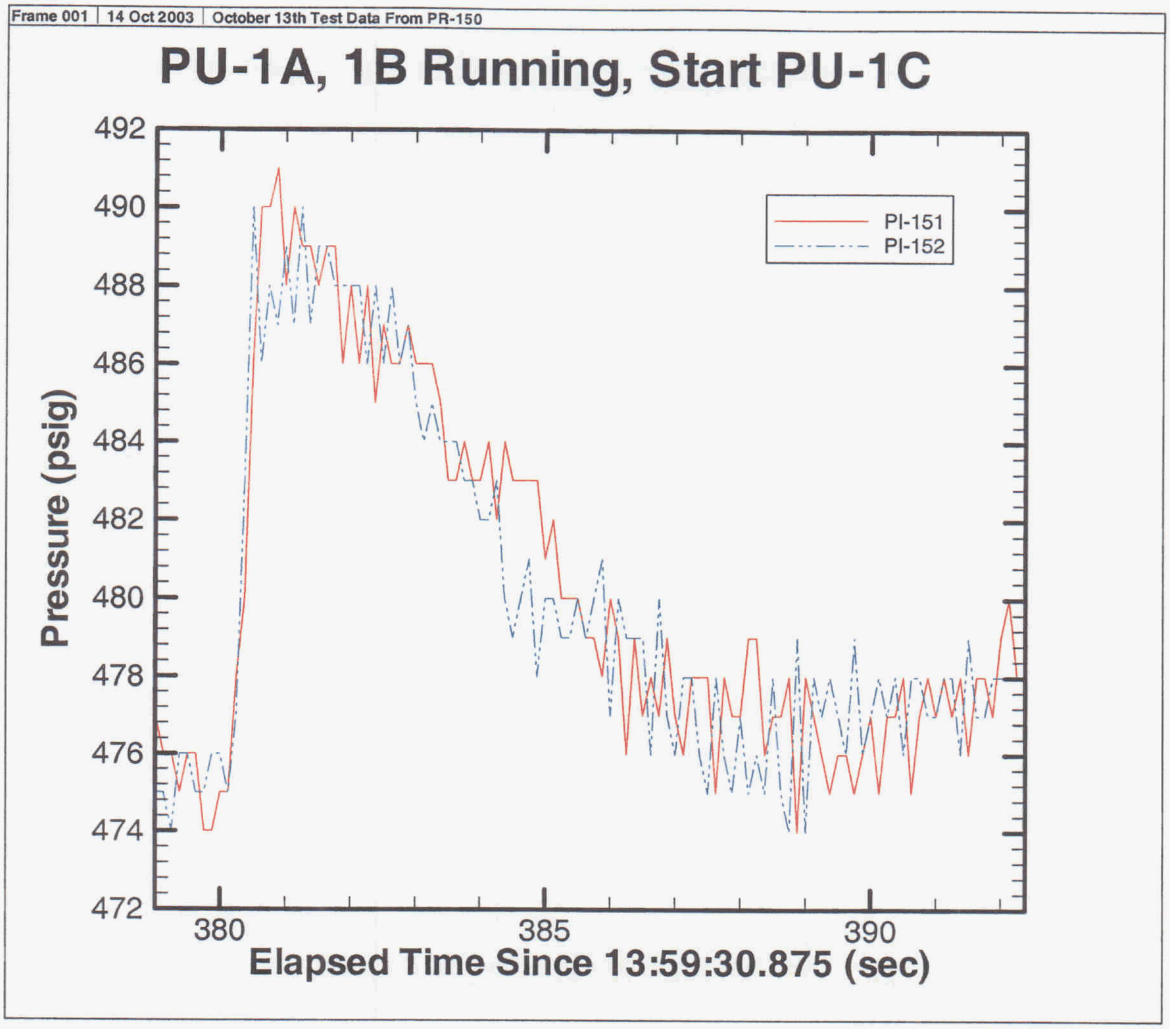

Figure 3-7. October 13th test data (PR-150) – start of PU-1C.

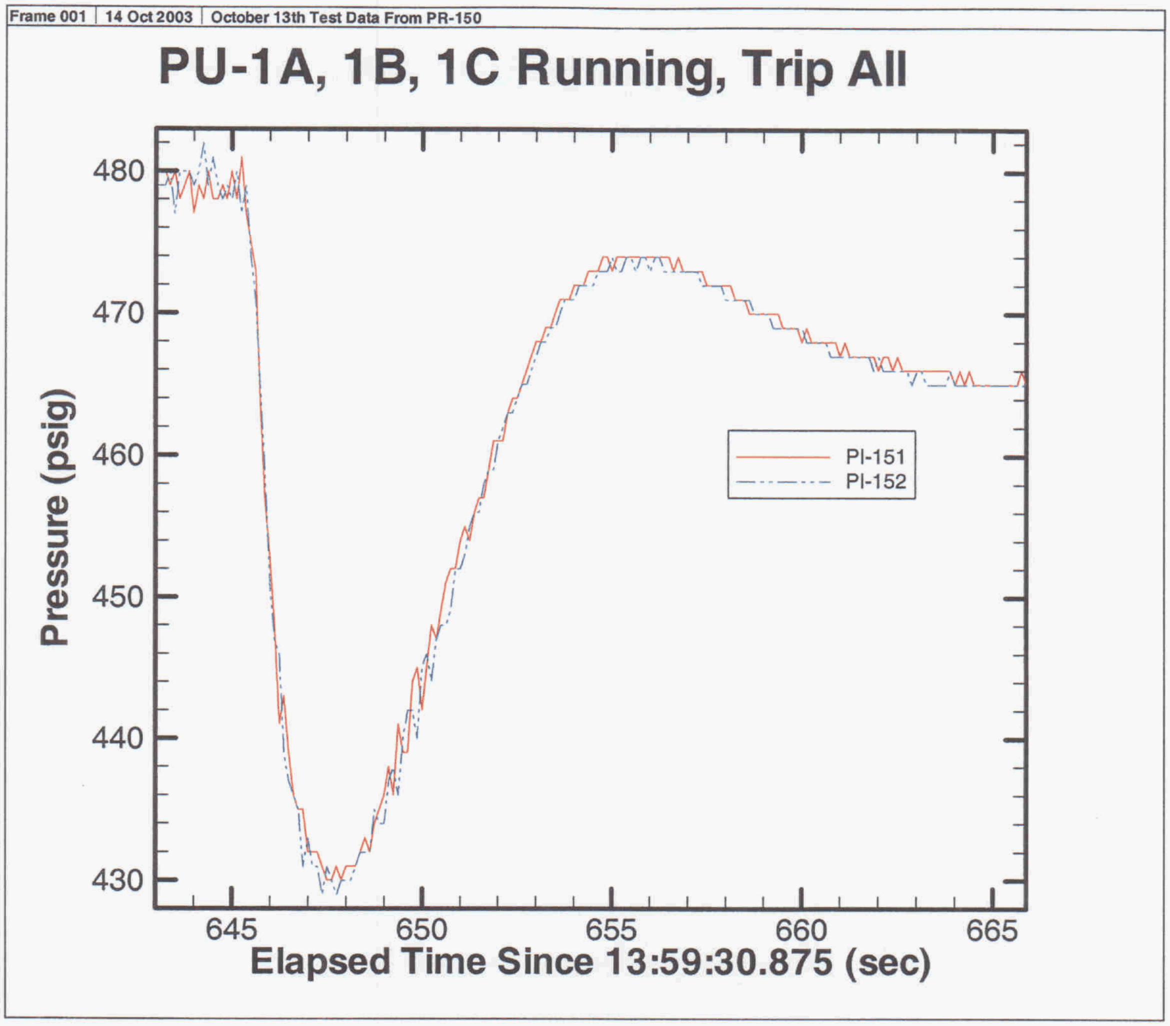

Figure 3-8. October 13th test data (PR-150) – trip of PU-1A, 1B, 1C.

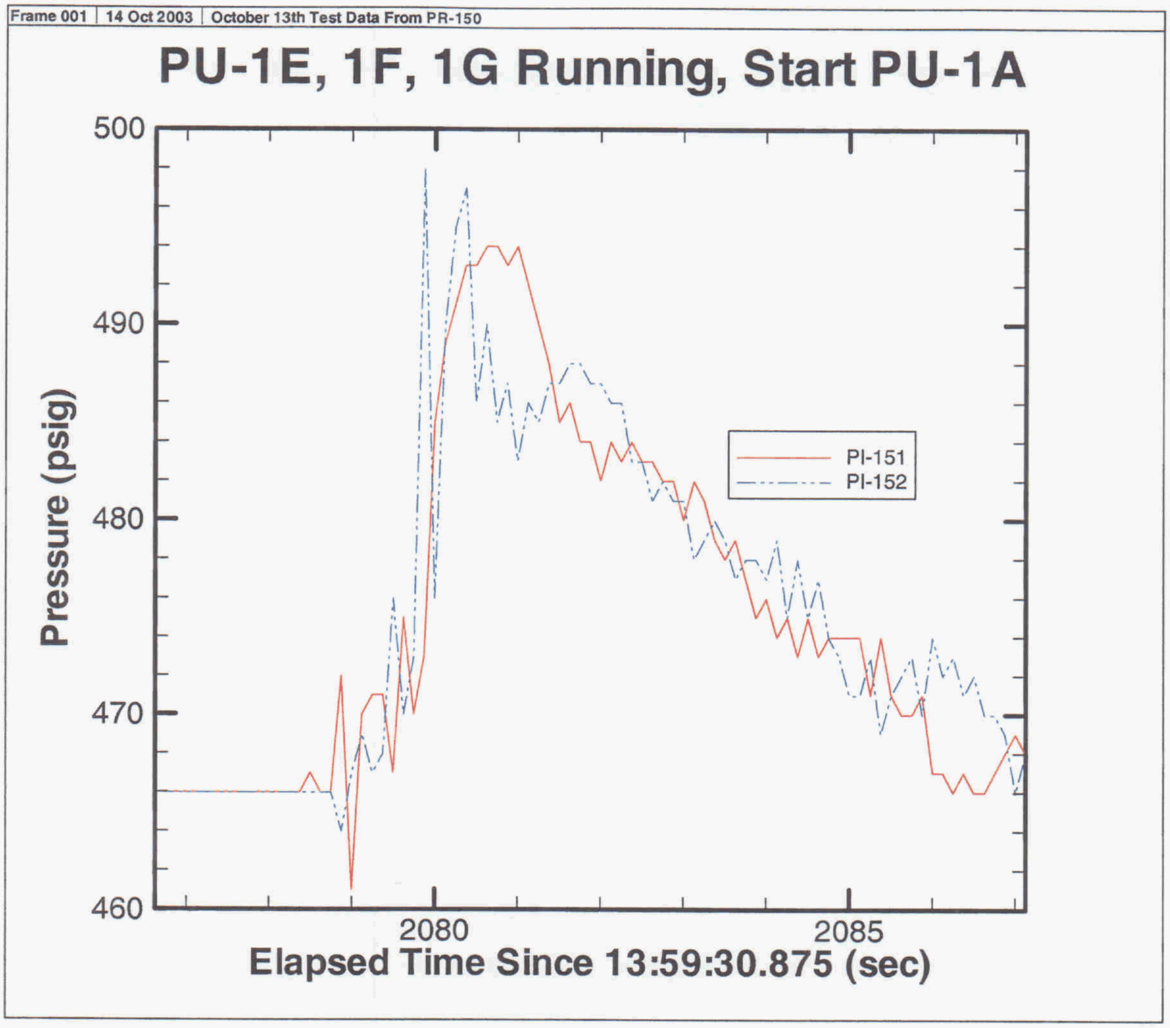

Figure 3-9. October 13th test data (PR-150) – start of PU-1A.

4.0 Preliminary Analysis of the Water Hammer Source and Transport to the Reactor Pressure Vessel

Research Reactors Division calculation C-HFIR-2003-035/R1 (Ref 9) was prepared to estimate the maximum pressure rise, just downstream of the check valve and at the reactor pressure vessel, due to a check valve closure following a main coolant pump trip. This calculation uses a conservative estimate of the pump coastdown in the discharge line of the tripped pump and estimates the pressure at the check valve seat, then calculates the attenuated pressure rise at the reactor pressure vessel using conservative textbook wave transmission factors. This section will discuss the approach utilized in this calculation and how it compares to other commonly-used methods. In addition, the assumptions, input, and results of this calculation will be summarized.

4.1 Analysis background

A textbook entitled *Fluid Transients in Systems*, by E. B. Wylie and V. L. Streeter (W&S), (Ref 10), presents a thorough discussion on the water hammer phenomena. The authors derive the fundamental equations of motion that are required to analyze a water hammer event. Several methods of solution to these equations are discussed and the method of characteristics is the favored method chosen. Several example fluid transient problems are presented and solved including water hammer problems similar to the problem at HFIR.

Wylie and Streeter describe several known methods of solving fluid transient problems. Water hammer problems and their solutions have been well understood for decades. Early methods utilize "arithmetic water hammer" and "graphical water hammer" as solution techniques. In essence the method used in the RRD calculation is of the "arithmetic water hammer" where the pressure rise is estimated from the density, the local speed of sound, and "reverse velocity" component impinging on the check valve. The method utilized by our consultant and also by W&S test is the method of characteristics. More advanced techniques have since been developed that take advantage of the increased computing capability that has arisen in recent years including implicit time integration and finite-element spatial integration.

The method of characteristics has become widely accepted as the preferred choice for solving water hammer problems of the type experienced at HFIR. This is a function of the fact that a water hammer-induced pressure pulse or wave travels at the speed of sound; on the order of 5000 ft/sec for the HFIR normal-operating conditions. It turns out that the eigenvalues of the Jacobian matrix of partial derivatives of the equations of motion with respect to the dependent variables includes a direct entry of the thermodynamic speed of sound ($\pm a$). One can take advantage of this mathematical property to estimate the speed of sound directly from the relationship $a = \sqrt{dP/d\rho}$. The method of characteristics takes advantage of this eigenvalue property to arrive at a very robust scheme for solving, and hence analyzing, problems of this type.

In contrast, the RELAP5 code, which is the customary thermal-hydraulics analysis computer code at RRD for analyzing thermal hydraulic problems at the HFIR (Ref 11), is not well suited for resolving phenomena traveling at the speed of sound. This is because RELAP5 utilizes a fast-running explicit (or semi-implicit as it is sometimes called) time-step scheme to analyze all problems modeled by the code. When attempting to solve this check-valve closure problem with RELAP5, a highly damped, and time-step dependent solution was obtained. Hence, it became obvious that the established method of analysis for system modeling at the HFIR was not adequate for this problem.

The long-term plan for attacking this water hammer problem includes the utilization of the method of characteristics. Several commercial codes are available for this purpose. In order to expedite an accurate solution as quickly as possible, a water hammer analysis expert, Dr. Sam Martin, has been consulted. Dr. Martin's service utilizes a method-of-characteristics-based computer code. This plan is discussed later in Chapter 9 of this report.

In the interim, a conservative and bounding estimate of the pressure changes that occur following the check-valve induced water hammer event at HFIR was prepared to provide a short-term safety basis for startup of the HFIR. The remainder of this chapter discusses the assumption, inputs, and results of this calculation.

4.2 Calculation assumptions, inputs, and results

RRD calculation C-HFIR-2003-035/R1 provides an estimate of the peak pressure that occurs in the primary coolant pump discharge line immediately following closure of the swing check valve located downstream of the pump. The calculation also estimates the transmission of the primary wave to the reactor pressure vessel. The wave transmission factors follow established formulas for shock wave transmission at cross-sectional area changes.

The basic formula utilized for the source term estimation of the pressure rise at the check valve is the Joukowski, or water hammer equation, $\Delta p = \rho a v_r$, where ρ is the water density, a is the local speed of sound, and v_r is the reverse velocity as "seen" by the check valve as it closes. Several assumptions and inputs are needed to arrive at the parameters in this equation. These are discussed in detail in the calculation, but listed here for clarity of the discussion.

The fluid properties are evaluated at the normal operating pressure of 468 psig and temperature of 120° F throughout the calculation. These conditions are identical to the normal system conditions at the end of cycle when one pump is shut off to begin the process of shutting down the primary coolant system or during an inadvertent primary coolant pump (PCP) trip.

It is assumed that no phase change or column separation will occur due to the water hammer which greatly simplifies the analysis. This is justified based on the data obtained from the testing, which is provided in Chapter 3 of this report.

The wave attenuation through transmission factors is obtained in a conservative manner by following the main flow paths of the coolant around the loop. For all area change reductions from the check valve to the reactor, the wave amplification is accounted for. For an area change expansion, the wave reduction is credited if readily obtained. However, if the geometry is too complex as to not allow for a timely solution, the area expansion is not credited. The net result yields a conservative estimate of the pressure change in the reactor pressure vessel.

The deceleration rate of the flow through the tripped main coolant pump is an input quantity that is necessary to estimate the reverse velocity at the check valve. From the deceleration rate, an empirical relationship for a 10-in. check valve is utilized to arrive at the reverse velocity at valve closure. The most conservative of two empirical estimates of the reverse velocity is used in the final result.

The thermodynamic sound speed is used as a starting point for the local speed of sound at the check valve and at locations in the attenuation path. Dissolved gas is one factor that will contribute to a reduction in the local speed of sound. However, dissolved gas is not credited in this conservative analysis. Additionally, the main coolant lines and related structures are not perfectly rigid and thus cause a further reduction in the local speed of sound due to small flexing of these structures during the wave propagation. However, the conservative analysis again ignores this effect.

The final results of the calculation indicate a maximum pressure change of ± 306 psi due to a sudden closure of the check valve at the discharge of a HFIR main coolant pump due to a single pump trip. The positive change of the pressure transient corresponds to the check valve discharge side, while the negative change corresponds to the inlet side of the check valve. Upon applying the transmission factors due to area change to the pressure change source term, the maximum expected pressure change at the reactor pressure vessel is +58 psi at the cold leg entry plenum and -55 psi at the hot leg entry plenum. It should be emphasized that this conservative analysis ignores the very large relative flow areas of the heat exchanger internals and the reactor pressure vessel internals. If these complex geometrical areas were credited, it is certain that further reduction in the estimated attenuated pressure perturbations in the reactor pressure vessel would be apparent. Furthermore, it is fully anticipated that when the detailed calculations discussed in Chapter 9 are completed, further reductions in the pressure changes will result.

5.0 Effect of Water Hammer on the Reactor Pressure Vessel Based on Preliminary Analysis

The High Flux Isotope Reactor (HFIR) pressure vessel has been in operation for approximately 35 years and radiation damage in the vessel midplane region has caused the HFIR pressure vessel to be in an embrittled condition (Fig. 5-1). A periodic hydrostatic proof test is used to prove that a critical combination of fracture toughness, flaw size, and stress will not exist before the next required hydrostatic proof test is performed. The potential for crack-like flaws to exist influences the conditional probability of failure of the reactor vessel. Previous studies have shown that the growth of flaws in the HFIR vessel is insignificant and, thus, does not affect the vessel integrity.

The newly identified pressure transient will contribute significantly to the total number of pressure cycles experienced by the vessel materials (if the frequency exhibited in the pressure measurement line is the same as that at the vessel) and thus, may impact the fatigue flaw growth of the vessel materials. For this reason, it is important to determine to what extent the pressure transient has contributed to additional growth of potential flaws in the vessel materials.

This preliminary analysis has determined that the subject pressure transient has contributed less than 0.03 inches growth of potential flaws in the reactor vessel materials, up until now. The amount of flaw growth is considered to be insignificant and, thus, does not result in an increased risk of vessel failure. Therefore, all of the conditional probabilities of failure P(F|E) remain the same as previously analyzed (Refs 3,4), and the hydrostatic proof test conditions were adequate to prove that the vessel may be operated safely [$P(F|E) \leq 1 \times 10^{-6}$ for the LCO pressure/temperature conditions] until the next hydrostatic proof test (Ref 21).

This preliminary analysis includes a very conservative hydraulics analysis. The pressure transient frequency of 50 Hz obtained from pressure traces generated by pressure instruments PT-151 and PT-152 is considered to be very conservative. Dr. Sam Martin, the subcontracted hydraulics expert, suggests that the pressure transient frequency generated by the closure of the pump discharge swing check valve is most likely in the neighborhood of 20 Hz. Additionally, the magnitude of the pressure transient used in this preliminary analysis is believed to be very conservative. The magnitude of the pressure pulse considered is the magnitude of the pressure pulse calculated in the primary coolant inlet nozzle of the vessel and in the primary coolant outlet plenum at the bottom of the vessel (Fig. 5-1). No credit is given in the analysis for the pressure pulse (wave) expansion into the vessel which will further reduce the pressure pulse magnitude (Ref. 9). This being the case, the estimated dynamic effect of this pressure transient on the flaw growth in the reactor pressure vessel materials is likely to be very conservative.

The pressure transient evaluated for impact on the reactor vessel materials has an oscillating magnitude of 58 psig (Ref. 9) with a frequency of 50 Hz for a duration of 0.75 seconds (See Chapter 7).

The total number of pressure cycles considered in the analysis was determined using the primary coolant system pressurization history (Ref. 12) since the beginning of HFIR operation, which is also found in the high-pressure primary piping probability of failure analysis (Ref. 5).

From the pressurization history and pressurizations since the last beryllium outage, it was determined that the check valve closure induced pressure transient has realistically occurred 1,286 times for an approximate total number of 45,000 pressure cycles due to this pressure transient.

The preliminary analysis utilized a pressure oscillation of between 410 psig and 526 psig for 45,000 cycles. The magnitude of pressure oscillation is established by adding 58 psig above and subtracting 58 psig below the reactor vessel core horizontal midplane nominal operating pressure of 468 psig.

Utilizing classical fracture mechanics flaw growth analytical techniques, the growth of a potential flaw in the reactor vessel material is less than 0.03 inches, an insignificant amount of flaw growth (Ref. 21).

Therefore, all of the current HFIR vessel integrity and life extension studies (Refs. 3, 4) remain valid, and all conditional probability of failure criteria are still satisfied. Likewise, all applicable limiting criteria and surveillance requirements remain valid in the HFIR TSR (Ref. 14). The last hydrostatic proof test performed in 2001 during the beryllium outage was adequate to prove that the vessel was safe for operation until the next scheduled hydrostatic proof test (Ref. 21).

The pressure transient, with a maximum pressure of 526 psig at vessel core horizontal midplane, is less than the limiting condition of operation (LCO) pressure (554 psig) corresponding to vessel wall temperature of 45°F in Figure 3.4.2-1 of Ref. 14. Furthermore, the LCO pressures increase as the vessel temperatures increase corresponding to normal operating temperatures resulting in even greater pressure margins at the higher temperatures. Therefore, the TSR LCO P/T curve has not been exceeded as a result of the pressure transient.

This pressure transient will be minimized in the future by changing pump operating procedures that will result in reducing the dynamic effects of the pump discharge check valve closures. This will make the impact of the valve closures even less significant.

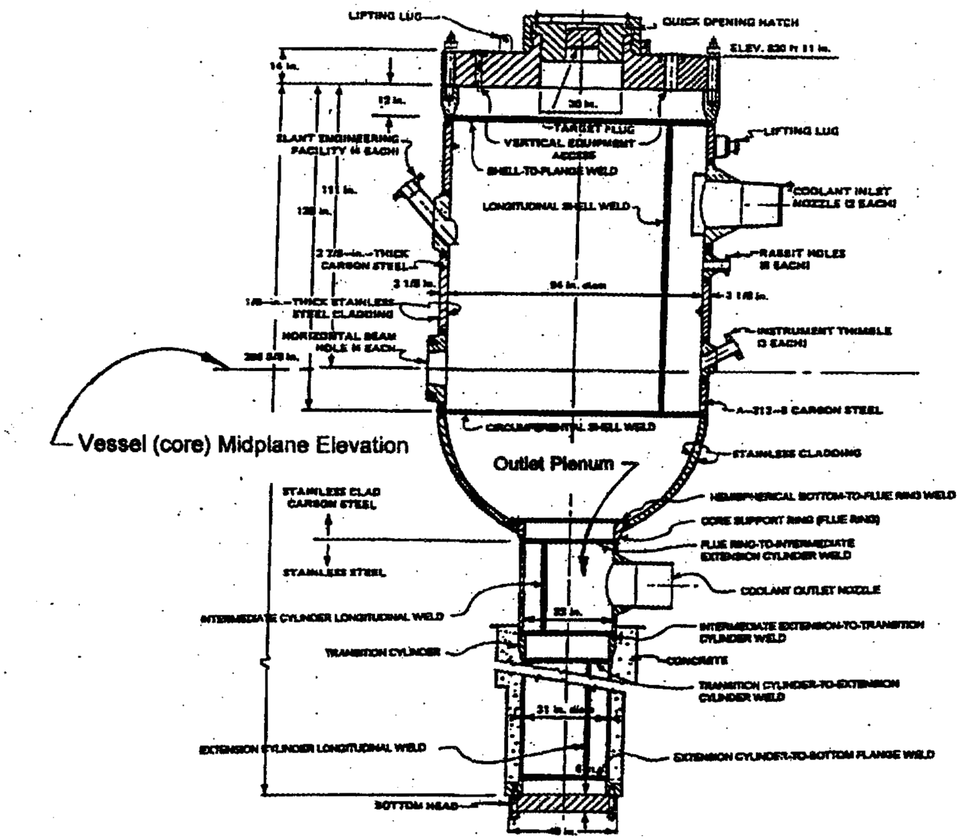

Figure 5-1

Figure 5-1. Reactor pressure vessel section.

6.0 Effect of Water Hammer on Primary Coolant Piping

The pressure transient recorded on September 13, 2003, as a result of the closure of the primary coolant pump discharge swing-check valve, was considered in this preliminary evaluation to determine the potential impact on the integrity of the high-pressure primary coolant piping and components. Pressure cycles, especially high-frequency pressure cycles, can cause an increase in the growth rate of flaws in the piping and component materials.

An evaluation (Ref. 5) was performed in 2001 to show that the conditional probability of a large-break LOCA (failure) was incredible in the high-pressure piping system and components (see Figures 6-1 and 6-2). It was necessary to show that the conditional probability of failure, $P(F|E) \leq 10^{-6}$ for the high pressure primary coolant system so that the process waste drain (PWD) system did not have to be upgraded to a safety system. The evaluation showed that the conditional probability of failure $P(F|E) < 10^{-6}$ and thus, was beyond design bases credibility. However, flaw sizes considered in that evaluation did not accommodate the potentially larger flaw growth rate that might result from the newly identified pressure transient.

This preliminary evaluation was performed to determine if the piping system and components $P(F|E) \leq 10^{-6}$ is still valid.

The recorded pressure transient data (see Fig. 7-3 in Chapter 7) was determined to be suspect and so a hydraulic analysis (Ref. 9) was performed that showed the cyclic pressure to be 198–810 psig at the check valve due to the sudden check valve closure. Therefore, the pressure transient considered in this analysis oscillates between 198 psig and 810 psig at 50 Hz at the check valve.

The hydraulic expert subcontractor, Dr. Sam Martin, suggested that the dynamic effects are likely to be less because the approach in the hydraulic analysis (Ref. 9) assumed a more rapid closure of the check valve than is likely to have occurred and the frequency of 50 Hz is likely to be on the high side as well. Dr. Martin suggested that the frequency of the transient is likely to be in the neighborhood of 20 Hz.

For the above reason, this preliminary evaluation is believed to be very conservative. Additional conservatisms have also been included in the evaluation because they have helped to expedite the evaluation process and they are addressed in the following discussion.

The preliminary calculations (Ref. 20) of the P(F|E) of the piping were performed by the same subcontractor who performed the original P(F|E) calculations in 2001. The approach has been utilized in the commercial nuclear power industry, and is identified in NUREG-1661 as the code for benchmarking any software used for predicting piping reliability in risk-informed inspection programs. The results of these preliminary calculations were utilized to determine the total P(F|E) of the piping system and components. The result was a total $P(F|E) < 1 \times 10^{-6}$ (Ref. 21).

The following is a summary of the preliminary evaluation.

The analysis began before some of the inputs were well quantified. So, input values were chosen that were believed to be bounding, and they were.

The pressure transient considered in the calculation of P(F|E) in the piping was an oscillating pressure of 0 psig to 900 psig with a frequency of 50 Hz. The pressures are conservative compared to the values above and a total of 50,600 cycles was used in the analysis. This number of cycles is larger than the 45,000 cycles used in the Chapter 5 analysis and is thus conservative.

The P(F|E) was calculated for the welds in the piping system representing the greatest potential for flaw growth and failure and were used to calculate the total $P(F|E)_T$. This preliminary evaluation has shown the $P(F|E)_T = 1.3 \times 10^{-8}$, which is less than 1×10^{-6} and, thus, remains beyond design bases credibility.

Therefore, in this very conservative evaluation, the conditional probability of failure of the high-pressure piping system components, $P(F|E)_T \leq 1 \times 10^{-6}$ and remains an incredible event.

It is pointed out that the flaw growth may be sufficiently increased such that some of the welds that were previously shown to be proven by the hydrostatic proof test (Ref. 22) for the inservice inspection (ISI) program (Ref. 8) may no longer be proven by the proof test. The piping welds whose integrities are not proven by the hydrostatic proof test are subjected to the typical ISI ultrasonic and penetrant testing, using the guidance of the ASME Code (Ref. 7).

An evaluation should be performed to determine if any additional welds should be subjected to the typical ISI ultrasonic and liquid penetrant testing. All of the welds that have received the ultrasonic and penetrant testing have been found to be very clean with no evidence of cracks. In 1998 and 2001, an ISI services subcontractor performed ultrasonic testing (UT) of a total of 15 of the highest stressed welds in the piping system. The non-destructive examination (NDE) personnel were certified through the Performance Demonstration Initiative (PDI) program conducted by EPRI and endorsed by the US NRC for the detection and sizing of intergranular-stress-corrosion cracks (IGSCC) and no cracks were identified. In fact, the subcontractor asserted "The welds were acoustically clean (very little geometry and no fabrication discontinuities were observed). Comparatively, based on industry experience, these welds proved to be extremely free of both geometric and metallurgical conditions . . . . Compared to other typical nuclear facilities . . . the welding quality for the HFIR high-pressure system was superior when compared to the industry standard" (Ref. 23).

There is no reason to have any immediate concerns regarding this issue for the following reasons:

1. There have been no failures of HFIR high-pressure system piping due to high-cycle fatigue flaw growth in approximately 35 years of operation (Ref. 24).

2. The UT and penetrant testing of welds have found them to be exceptionally free of crack-like flaws. In fact, none have been found in UT of the welds since the HFIR started up in 1965–1966 (Ref. 24).

3. The leak-before-break concept applies to the HFIR stainless steel piping system and components, and leaks can be readily identified and remedied long before they become safety concerns (Refs. 5, 25).

Nevertheless, the subject evaluation (Ref. 22) should be updated to accommodate this transient in a reasonable time frame following the hydraulic modeling of the piping system and components by Dr. Sam Martin and additional testing performed to more accurately characterize the dynamic effects of the water hammer transient following the changes in the operating procedures.

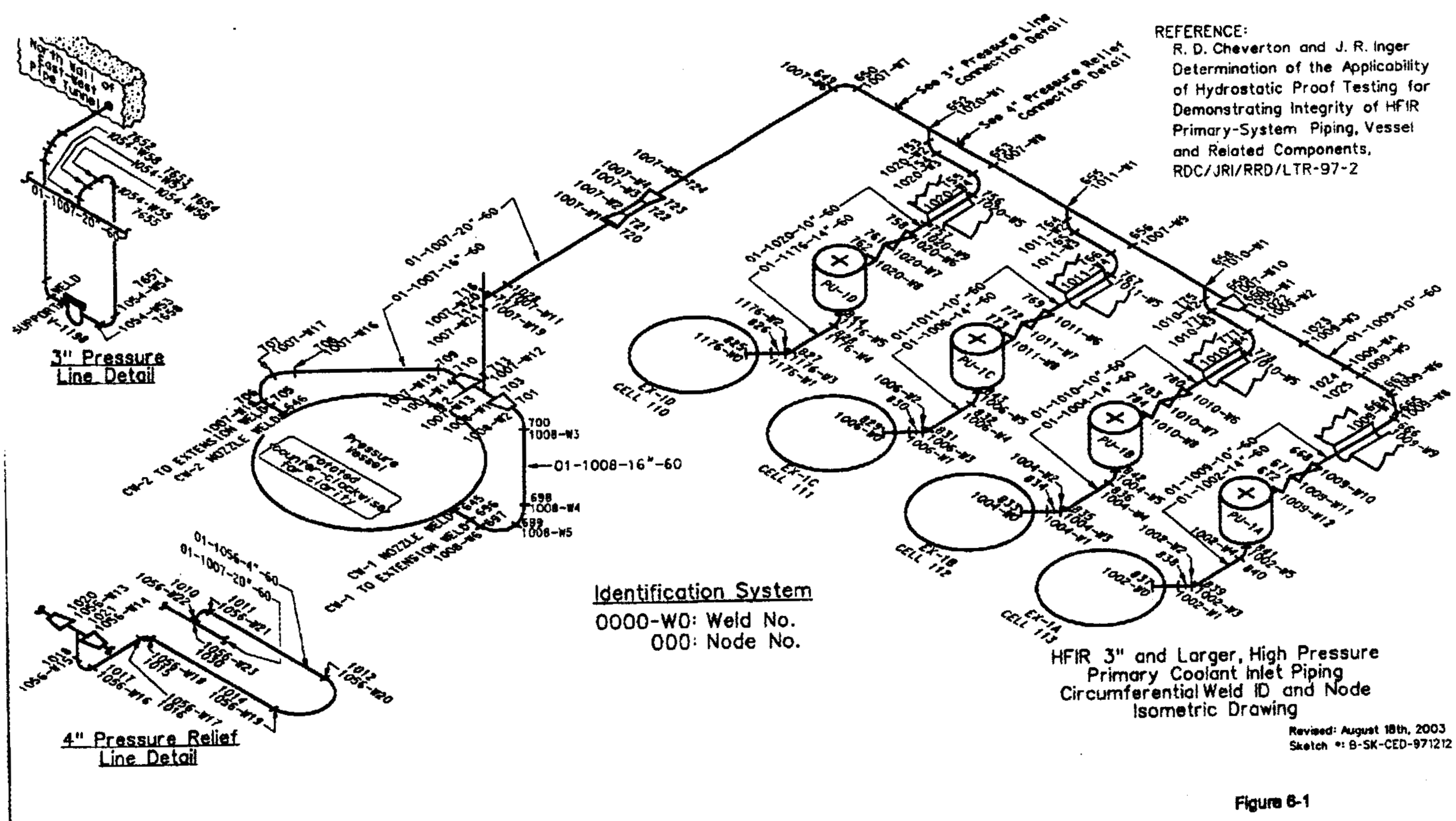

Fig. 6.1. HFIR 3-inch and larger, high pressure primary coolant inlet piping.

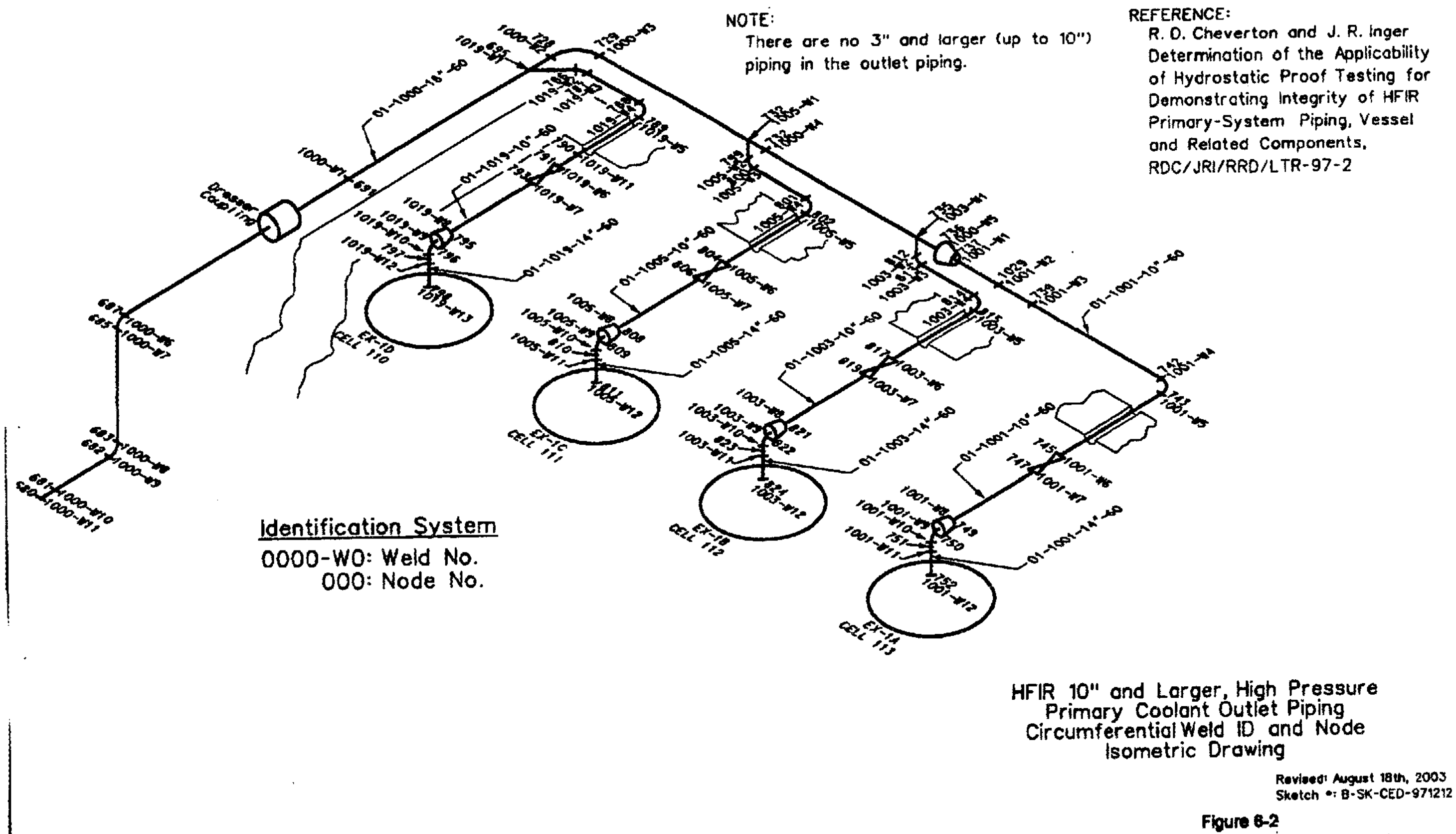

Fig. 6.2. HFIR 10-inch and larger, high pressure primary coolant outlet piping.

7.0 Effect of Water Hammer on the Primary System Rupture Disks

The reactor vessel/high-pressure primary coolant pressure relief system has two parallel relief paths. Each relief path has a rupture disk and a relief valve. The rupture disks are inboard (exposed to continuous primary pressure) of the pressure relief valves. The pressure transient data was obtained from pressure transducers (PT-151 and PT-152 and associated recorders). Their 3/8-inch-diameter tubing sense lines connect to the piping and sense pressures immediately upstream of the rupture disks, as shown in Fig. 7-1. The rupture disks are BS&B, RLS, reverse-buckling type disks with the bulged dome on the pressure side. The bulged dome design utilizes superior characteristics that enhance disk reliability. The manufacturer indicates that its superior life cycle characteristics are such that even after thousands of pressure cycles of 0 – 90% of the disk rating, the RLS disks exhibit no premature failure or fatigue cracks (Ref. 15), as shown in Fig. 7-2.

The initial data obtained on September 13, 2003, show the transient peak pressures to be clipped at approximately 650 psig due to recorder instrument limitations, as shown in Fig. 7-3. An extension of these pressure peaks graphically shows the pressures might reach approximately 850 psig. The frequency of these pressure pulses during the transient is approximately 50 Hz.

The rated rupture pressure of the rupture disks is 650 psig coincident with a disk temperature of 120°F and 679 psig coincident with a disk temperature of 85°F.

The burst tolerance specified by the manufacturer (BS&B) is ± 5% (Refs. 15, 16). This is in accordance with the ASME Code (Ref. 17). The tolerance results in a maximum burst pressure of 682.5 psig and 713 psig coincident with rupture disk temperatures of 120°F and 85°F, respectively.

The manufacturer recommends replacing the rupture disks if the disks experience pressures greater than 90 percent of the rated pressure which corresponds to pressures in excess of 585 psig and 611 psig coincident with rupture disk temperatures of 120°F and 85° F, respectively.

The transient pressures of approximately 850 psig at the rupture disks (see above) exceed:

(1) Ninety percent of the rated pressure, and

(2) The maximum burst pressures of the rupture disks, however, the disks did not burst.

Two potential issues (questions) are suggested by this data that warrant evaluation:

1. Do these pressure transients damage the disks so that they no longer provide adequate over-pressure protection for the HFIR pressure vessel?

2. The subcontracted hydraulics expert, Dr. Sam Martin, suggested that the transient most likely induces system pressures and frequencies less severe than represented by the data obtained on September 13, 2003. Therefore, it is conservative to assume the pressures at the ruptures disks were as high as 850 psig. In this case, it is appropriate to ask, why didn't the rupture disks burst when the pressure at the disks exceeded their rated rupture pressure? Does this indicate that the rupture disks are not adequately protecting the vessel from experiencing over-pressure conditions?

The following discussions resolve both of these issues with no increased risk to the integrity of the HFIR pressure vessel.

1. The manufacturer (BS&B) has indicated that pressure cycles exceeding 90% of the disk's rated pressure may tend to weaken the disk and cause the disk to burst early, i.e., at a lower than rated burst pressure. Exceeding 90% of the rated rupture pressure will not cause the disk (shown in Fig. 7-2) to burst at pressures higher than the rated pressure (Ref. 18).

   Therefore, if these rupture disks are subjected to cyclic pressures exceeding 90% of their rated pressures, the rupture disk may tend to burst at lower pressures, in the conservative direction, while providing adequate over-pressure protection for the HFIR pressure vessel. The tendency to actuate at lower pressures provides a slight increase in the probability of a small-break LOCA due to rupture disk actuation. However, the size of this small-break LOCA (1 ½-inch line) is bounded by the 2-inch vessel break in the USAR. Operational changes to minimize this increase in probability are discussed below.

   The potential bursting of the rupture disks at lower than anticipated burst pressures may negatively impact the availability of the HFIR. The following actions will be taken prior to restart to minimize the potential for this impact.

   - The rupture disks will be replaced with new rupture disks, and
   - Operating procedures will be revised to minimize the severity of the pressure transient.

   It is pointed out that an unanticipated (low instrument indicated pressure) bursting of the rupture disks occurred in 2002. However, analysis determined that failure was not due to a weakened (fatigued) disk, but, rather resulted from an overload event displaying significant plastic deformation with no evidence of fatigue failure, which was caused by a rapidly collapsing air pocket at the disks (see the Introduction for additional information).

   At a later date a design change was implemented that provided venting capabilities on the pressure side of the rupture disks and added the present pressure sensing instrumentation, PT-151 and PT-152, which was used to identify the subject pressure transient (see the Introduction for additional information).

2. The manufacturer (BS&B) has indicated that pressure cycles of less than one-second may not be long enough to burst the rupture disk even if the cyclic pressures exceed the rated rupture pressure of the rupture disk. However, the disk will burst if exposed to the rated pressure for a duration of at least one-second (Ref. 18).

   The manufacturer of the rupture disks did not have rupture disk data for disks exposed to high-cycle pressure transients (Ref. 18). The simplified reason for why short-lived pressure pulses may not burst the rupture disk is that sufficient strain must be experienced by the disk material before rupture (failure of the disk material) can occur. If the pressure is not maintained for a long enough duration to result in excessive strain (deformation), rupture (failure of the disk material) will not occur. Although the BS&B RLS reverse buckling rupture disk burst mechanism is somewhat more involved, this, nevertheless, is consistent with sound strength of material relationships.

   However, this is not a concern because the pressure-relief system is not designed to respond to high-cycle pressure transients. The design of the pressure-relief system is adequate even if it does not respond to this high-cycle pressure transient because the high-cycle pressure transient (identified as the most severe high-cycle transient in the high-pressure primary coolant piping system) has been shown to have a net impact on the conditional probability of failure P(F|E) of the HFIR pressure vessel and piping system components such that the P(F|E) remains less than 1×10^{-6}, and thus, remains incredible (Chapters 5 and 6).

   It is concluded that these pressure transients do not affect the rupture disks in a nonconservative direction and that the pressure-relief system continues to provide adequate over-pressure protection for the HFIR pressure vessel.

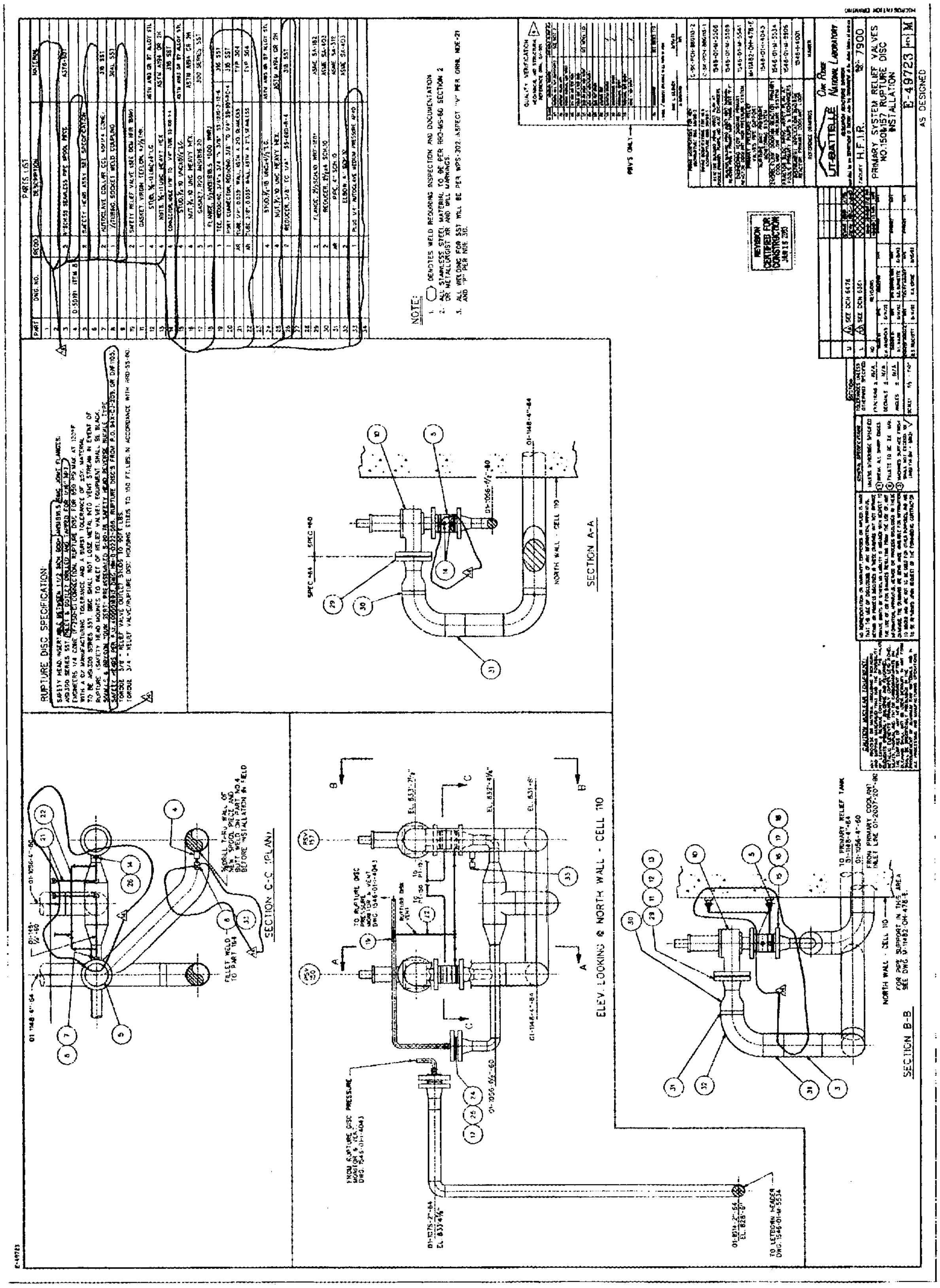

Fig. 7-1. Pressure relief system and pressure sense line geometry.

RLS™ Precision Circular-Scored Reverse Buckling Rupture Disk

The RLS is a Precision Circular-Scored Reverse Buckling Rupture Disk. When over-pressurized, this domed, solid metal, reverse buckling disk reverses and opens along a pre-weakened circular score line on the down-stream side of the disk. A patented hinge welded to the disk facilitates relief opening along the score line and retains the disk's central petal preventing fragmentation even at high burst pressures.

- **Gas and full liquid service**
- **Designed for non-fragmentation**
- **Withstands full vacuum**
- **Suitable for operating pressure to 90% of the marked burst pressure and 95% of the minimum burst pressure (CEN ISO 4126-2 Standard pending)**
- **Damage safety ratio 1.5. A damaged RLS disk will burst at or below 1.5 times its marked burst pressure**
- **Optimum fatigue resistance in pressure pulsating or cycling conditions**
- **Recommended for safety relief valve isolation**
- **Optional TEF liner on the process and/or down-stream side of the disk. Order as "TEF Liner" or identify special requirements**

Disk Material

Disk Size		Tantalum				Nickel Alloy 200				Hastelloy® Alloy C-276 & 316 Stainless				Inconel® Alloy 600				Monel® Alloy 400			
		Min.		Max.		Min.		Max.		Min.		Max.		Min.		Max.		Min.		Max.	
in	mm	psig	barg	psig	barg	psig	barg	psig	barg	psig	barg	psig	barg	psig	barg	psig	barg	psig	barg	psig	barg
1	25	125	8.62	2000	137.9	125	8.62	2000	137.9	175	12.07	2000	137.9	150	10.3	2000	137.9	150	10.3	2000	137.9
1½	40	85	5.86	1800	124.1	85	5.86	1800	124.1	145	10	1800	124.1	105	7.24	1800	124.1	105	7.24	1800	124.1
2	50	70	4.83	1800	124.1	70	4.83	1800	124.1	115	7.93	1800	124.1	85	5.86	1800	124.1	85	5.86	1800	124.1
3	80	55	3.79	1600	110.3	55	3.79	1600	110.3	75	5.17	1600	110.3	65	4.48	1600	110.3	65	4.48	1600	110.3
4	100	45	3.10	1050	72.4	45	3.10	1050	72.4	65	4.48	1050	72.4	55	3.79	1050	72.4	55	3.79	1050	72.40
6	150	35	2.41	650	44.8	35	2.41	650	44.8	45	3.10	650	44.8	40	2.76	650	44.8	40	2.76	650	44.8
8	200	n.a.	n.a.	n.a.	n.a.	35	2.41	400	27.6	45	3.10	400	27.6	40	2.76	400	27.6	40	2.76	400	27.6
10	250	n.a.	n.a.	n.a.	n.a.	35	2.41	250	17.2	45	3.10	250	17.2	40	2.76	250	17.2	40	2.76	250	17.2
12	300	n.a.	n.a.	n.a.	n.a.	35	2.41	150	10.34	45	3.10	150	10.34	40	2.76	150	10.34	40	2.76	150	10.3
14	350	n.a.	n.a.	n.a.	n.a.	35	2.41	130	8.96	45	3.10	130	8.96	40	2.76	130	8.96	40	2.76	130	8.96
16	400	n.a.	n.a.	n.a.	n.a.	25	1.72	110	7.58	35	2.41	110	7.58	30	2.07	110	7.58	30	2.07	110	7.58
18	450	n.a.	n.a.	n.a.	n.a.	25	1.72	90	6.21	35	2.41	90	6.21	30	2.07	90	6.21	30	2.07	90	6.21

US Patent number: 4,404,982 and other international patents. The hinge attached to the disk is 316SS. TEF liners available at all burst pressures. For burst pressures below RLS minimums use JRS, HRS, SK_R, Sigma or S-90 type disks. Hastelloy is a trademark of Haynes International Inc., Monel and Inconel are trademarks of Inco Alloys International, Inc.

Figure 7-2

7

Figure 7-2. Rupture disk information from vendor.

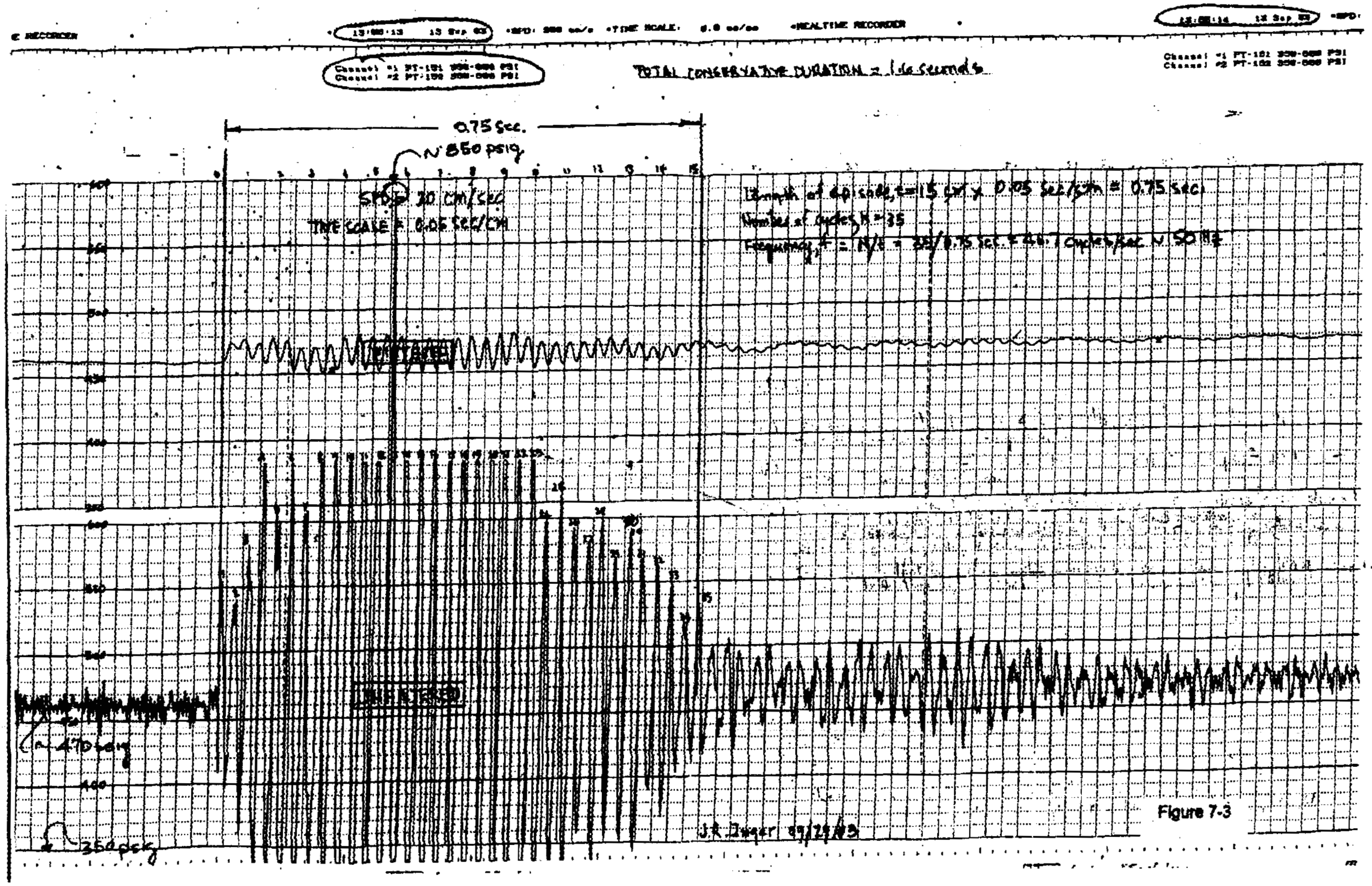

Fig. 7-3. Initial transient data from PI-151/152 showing clipped peaks.

8.0 Operational Changes to Support Reactor Startup

Changes to plant operating procedures are needed to ensure the following two objectives are met.

1. Prevent the occurrence of check valve closure-induced water hammer during normal stopping of the AC-powered main coolant pumps (MCP).

2. Due to the continued uncertainty in the pressures at the primary system rupture disks during the water hammer transient, provide for the evaluation of rupture disk operability and replacement when the AC-powered main coolant pumps are stopped by any means that doesn't prevent check valve closure-induced water hammer.

To address the first objective, it was proposed that simultaneous shutdown of all operating AC-powered MCPs would maintain similar differential pressures across each pump during coast-down, thereby preventing the high-energy closure of any one pump's check valve. The method chosen to initiate the simultaneous shutdown of all operating AC-powered MCPs was the deenergizing of relays in the 2-of-3 matrix providing a trip of the MCPs on low primary system pressure. The actuation of these relays performs the same function in the MCP control circuit as placing the pump control switch in stop, but accomplishes this for all MCPs at once.

This method for performing a simultaneous shutdown of the MCPs was validated during the testing of October 13, 2003. The testing showed no evidence of check valve closure-induced water hammer when using this method. Further, tests were run to evaluate the two other potential sequences of pump starting and stopping that could result in rapid closure of the check valves: 1) stopping a MCP driven by a pony motor with the other two MCPs being driven by their pony motors, and 2) starting an AC-powered MCP with the other two MCPs being driven by their pony motors. Similarly, these sequences did not produce a check valve closure-induced pressure transient of concern.

To ensure that the tested method for performing a simultaneous shutdown is utilized during normal stopping of the AC-powered MCPs, the procedure for "Operation of the Primary Coolant Pumps" (NOP-2106) has been revised to require deenergizing of the relays whenever stopping the AC-powered MCPs. While this will satisfy the first objective in the interim, the logistics of using this method are more cumbersome than use of a stop switch. It is anticipated that the final analysis of this problem will show that the interim measure of tripping the pumps simultaneously is not necessary, as described in Chapter 9. Until the final analysis is complete, the interim measures will stay in place. As a contingency, a request for modification will be submitted to install a common stop switch to achieve simultaneous shutdown of the AC-powered MCPs, so that a permanent change can be made if simultaneous shutdown of the pumps is required by the final analysis.

To address the second objective, it was necessary to identify procedures in which identification is made of instances when the AC-powered MCPs are not stopped simultaneously. During reactor operation, this will occur as part of either: 1) swapping of primary heat exchanger and MCP combinations, or 2) unexpected trip of a MCP. The procedures which direct the swapping of heat exchanger and MCP combinations ensure that the MCP discharge isolation valve is shut before the pump is stopped. With the discharge valve shut, the check valve will not close rapidly when the AC-powered MCP is stopped. An unexpected trip of a MCP during reactor operation will always produce the entry conditions for the "Primary Coolant Flow Problems" abnormal operating procedure (AOP-9002). This is further reinforced by the fact that all of the annunciator procedures that would accompany a MCP trip reference AOP-9002.

Technical Safety Requirement (TSR) surveillance 4.4.7.4 requires replacement of the primary system rupture disks whenever the pressure at the inlet to the disks reaches their deformation pressure. Because the instruments that monitor rupture disk pressures have shown that deformation pressures are reached when stopping a single AC-powered MCP (with its discharge valve open and not simultaneous with the stopping of two other operating AC-powered MCPs), yet the installed pressure recorder does not reliably capture the peak pressure, it must be assumed that deformation pressures are reached during each of these check valve closure-induced water hammer transients. Assuming the deformation pressure has been reached would also necessitate declaring the rupture disks inoperable. The TSR limiting condition of operation (LCO) 3.4.7 requires the reactor to be shutdown with inoperable rupture disks. Therefore, the necessary course of action upon identification of a pump trip would be: (1) shutdown the reactor and (2) replace the primary system rupture disks.

Since AOP-9002 is entered upon a trip of a MCP during reactor operation, it has been revised to require shutdown of the reactor when a MCP trip has occurred. A precaution and limitation has been added to NOP-2106 to declare the rupture disks inoperable and require their replacement when any AC-powered MCP is not stopped simultaneously with other operating AC-powered MCPs. Taken together, these revisions provide an interim measure to ensure the TSRs, as written, are satisfied by performing the necessary actions described above. As a longer term solution, a revision of the TSRs will be evaluated, using as its basis the position that the pressure transient's effect on the rupture disks does not change their characteristics in a way that is adverse to safety.

9.0 Plans for Long-Term Analysis and Testing

This chapter describes the plans for long term analysis and testing to resolve the water hammer issue at the HFIR. A detailed method of characteristics model of the HFIR system is under development by our consultant, Prof. Sam Martin. It is anticipated that results from this model will be applicable for use in the final analysis of the effects on the vessel and piping. Additional plant testing is planned, with pressure sensors placed directly on the 20-inch cold leg, the reactor pressure vessel, and perhaps elsewhere in the system. This additional testing and resulting data will enable RRD staff to better resolve questions about rupture-disk-sense-line resonance, magnitude and frequency of the pressure pulses, and validation of the afore-mentioned method of characteristics model.

In Chapter 4, a discussion is presented of the various methods available for analyzing water hammer events. As a first order and conservative estimate, the Joukowski equation was utilized to compute the maximum amplitude of the pressure change at the main coolant pump discharge due to the sudden closure of the check valve. In addition, the transmission of a pressure change to the reactor vessel was estimated based on area changes. Dr. Martin has confirmed with RRD that this approach is a valid method of analysis toward a conservative estimate of the wave amplitudes. While this method is certainly adequate to provide a conservative estimate, it is important to establish a more accurate understanding. Therefore, RRD is engaging to develop a complete method of characteristics (MOC) model of the entire primary coolant loop and to analyze results from this model.

Prior to the MOC model development, it is important to establish the fundamental frequency of the instrumentation subsystem attached to the main coolant piping which was used to acquire the data presented in Chapter 3. The data indicates a fundamental frequency of approximately 40 Hz. A first order approximation of the main coolant loop fundamental frequency is much lower than this (~10-20 Hz). Therefore, it is believed that the frequency measured may primarily be a function of the method in which the data is acquired. There was also a concern expressed that the frequency may be caused by a chatter of the check valve itself after the rapid closure. A confirmation of the cause of the higher than expected frequency will then justify the need to place additional pressure instruments in a different location such that a more complete measurement of the water hammer event may be obtained. Dr. Martin planned to obtain the instrumentation fundamental frequency, utilizing an established frequency domain analysis and code methods, as his first priority. As soon as this information is available, it will be incorporated and reported to DOE.

Having established a maximum pressure amplitude and a firm understanding of what the existing instrumentation is providing, the next step in the long-term plan is to build a MOC model of the HFIR system. This model will incorporate many items which are well established for the HFIR system such as piping geometry, vessel geometry, valve characteristics, etc. Indeed, much of this information is already included in the present RELAP5 model of the HFIR system which will help in the efficient development of the MOC model. However, some items need to be newly developed in order to complete the

MOC model to the detail required. The items to be newly developed include a detailed model of the check valve, generation of additional pump performance curves in reverse flow quadrants through the use of homologous pump curve or dimensionless similarity transformations, and information about the pump speed and moment of inertia. The current RELAP5 model uses actual pump data for conditions of positive flow and speed, and built-in curves for negative flow and speed. Further, the present RELAP5 model utilizes a check valve model which is not as detailed as the model Dr. Martin plans to develop for RRD to accurately simulate the source term of the pressure disturbance to the HFIR main coolant loop.

Dr. Martin emphasized the importance of an accurate representation of the check valve dynamic flow characteristics and structural motion. The entire event is on the order of a single second in duration. The check valve closure time is even smaller than this, but it is not instantaneous. Based on his past experience with check valve closure simulation, Dr. Martin expected the HFIR check valve to gradually close to a certain fraction of the fully-open flow area, then the closure would accelerate and finally close at a much faster speed then the initial closure motion. Dr. Martin felt that the resistive torque exhibited by the check valve rotational axis would contribute in a major way to the force generated on the fluid at the closure, and hence, the magnitude of the resulting pressure change as the source term in the water hammer event. Direct measurements of the HFIR check valve torque during closure are not available. However, Dr. Martin anticipated that he could arrive at that torque given the detailed flow versus pressure drop data that was available on the check valve from testing performed during the time of initial installation of the check valve. This aspect of the issue is key in assessing the accuracy (and perhaps ultimate reduction) of the pressure change presently estimated by RRD calculation C-HFIR-2003-035/R1.

In addition to the check valve, new model development is necessary for the main coolant pump performance curves. This information is necessary to properly describe the performance of the pump during the check valve closure event on the order of a few seconds up to, including, and after the check valve closure. Most centrifugal pump manufacturers only provide the performance of the pump during normal operations; i.e., for positive flow, positive head, and positive torque applied by the rotating shaft under power. Indeed, this is the case for the HFIR main coolant pump and this data is well established. Up until now, during off-normal events analyzed for HFIR with RELAP5, additional data such as pump coastdown curves, and pump speed versus time after trip, are available from direct plant data to describe the pump during these events. However, in order to simulate the water hammer off-normal event, it will be necessary to describe the pump performance during reverse flow, and positive, but decelerating pump speed (un-powered). A complete description of the centrifugal pump performance is available utilizing dimensionless-homologous pump characteristics. This technique is described in detail in the text by Wylie and Streeter discussed earlier in Chapter 4. Because the homologous curves are dimensionless, information available from other pumps similar to HFIR's main coolant pump may be applied to arrive at pump performance in the other quadrants (negative flow, negative head, negative speed). Dr. Martin pointed out that it would not be necessary to model all possible quadrant combinations, but only those regions that would be exercised by the flow developed during the water hammer event. As a side benefit to this part of the effort, because RELAP5 also includes the ability to model pump homologous curves directly, this information could be ported over into the HFIR system RELAP5 model.

In addition to the detailed check valve model, and additional pump characteristics, the entire loop geometry will be incorporated into the MOC model. This task is a more or less routine part of the model development and will possibly take advantage of the existing data from the HFIR RELAP5 model to generate this data in an efficient manner. The geometric data needed for the MOC model is less detailed than that required for the similar data in the RELAP5 model, so that it is anticipated that the loop geometry may be generated in a relatively short amount of time. In addition, Dr. Martin has been provided isometric, physical, and P&ID line drawings of the loop which will be his primary source for the data input.

After providing all this necessary data describing the HFIR loop as a MOC computer model, Dr. Martin will then provide numerical solutions to the check valve closure event at HFIR. The MOC model will produce pressure changes in any point in the loop, including the reactor vessel, simultaneously such that a complete assessment of the impact of the water hammer event may be determined. As mentioned earlier, the entire event is expected to last only 1-2 seconds in duration (loop length on the order of 100-200 ft).

Depending on the fundamental frequency determination and MOC model results, further testing may be required to validate the model. If it is determined that the present instrument measurements are complicated by local excitation of the instrument lines themselves, it may also be necessary to insert additional instruments directly onto the main coolant piping. In this manner, the wave amplification effect would be removed, and the proper fundamental frequency would be measured. Dr. Martin has suggested direct pressure instruments might be placed at the check valve discharge, reactor pressure vessel, main 20-inch coolant line, and pump suction piping (see Appendix A). In addition, Dr. Martin has suggested that a direct measurement of pump speed versus time, and a separate flow measurement be available at the pump discharge.

10.0 Conclusions

RRD has followed a careful and technically sound approach in analyzing this issue. Plant data, including pressure monitoring and physical inspections of two sets of rupture disks involved in the testing, has been collected and analyzed, and has undergone both independent review as well as input and feedback from a recognized industry expert.

All of this has been considered in the overall analyses and described in this safety evaluation. As noted in the associated USQD, three questions were defined as being "positive" in terms of identifying conditions that were potentially not bounded by the current safety analysis. RRD believes that these questions have been sufficiently analyzed with justification for further operations based on this documented evaluation and as noted below.

(1) Could the change increase the probability of occurrence of an accident previously evaluated in the documented safety analyses?

Two separate issues were discussed relative to this probability question. The first was the probability of failure of the rupture disks. As described in chapter 7 of this evaluation, the manufacturer recommends replacing the rupture disks if the disks experiences pressures greater than 90 percent of the rated pressure. Previously it was thought that exposure to these high cycle pressure transients would possibly cause the disks to rupture at higher than rated pressures. This issue has been clarified with the manufacturer, with it now being known that the disks will become weaker, and rupture at lower than rated pressures if challenged in this way.

It is concluded that these pressure transients do not affect the rupture disks in a non-conservative direction and that the pressure-relief system continues to provide adequate over-pressure protection for the HFIR pressure vessel.

The second issue relative to the probability question dealt with the conditional probability of failure of the primary piping or the reactor vessel. Previous analysis had not fully considered high cycle fatigue issues that this water hammer transient identified. Analysis described in Chapters of 5 and 6 of this evaluation has shown that although probability of failure may have increased slightly, it is still bounded by the 1×10^{-6} conditions delineated in the approved USAR. Therefore, all of the current HFIR vessel integrity and life extension studies remain valid, and all conditional probability of failure criteria are still satisfied. Likewise, all applicable limiting criteria and surveillance requirements remain valid in the HFIR TSR. Piping weld ISI inspections, which are required by the Administrative Controls Section of the TSRs, will be updated as necessary when the final analysis is complete. The last hydrostatic proof test performed in 2001 during the beryllium outage was adequate to prove that the vessel is safe for operation until the next scheduled hydrostatic proof test, and the TSR LCO P/T curve has not been exceeded as a result of the pressure transient.

Based on this information, the probability issue has been addressed, and found to be bounded by the current DOE approved safety basis.

(6) Could the change create the possibility of a different type malfunction of equipment important to safety than any previously evaluated in the documented safety analyses?

Technical Safety Requirement (TSR) surveillance 4.4.7.4 requires replacement of the primary system rupture disks whenever the pressure at the inlet to the disks reaches their deformation pressure. Installed instrumentation used to monitor pressure at the rupture disks were not specified to capture information surrounding a water hammer in the primary system as described in Chapters 7 and 8 of this evaluation, though they have shown that deformation pressures are reached when stopping a single AC-powered MCP (with its discharge valve open and not simultaneous with the stopping of two other operating AC-powered MCPs). Because the installed pressure recorder does not reliably capture the peak pressure, it must be assumed that deformation pressures are reached during each of these check valve closure-induced water hammer transients. Although the data recording instruments do not capture the same information as a high speed recorder relative to the transient, they provide significant information relative to maintaining compliance with current TSRs.

Assuming the deformation pressure has been reached would also necessitate declaring the rupture disks inoperable. The TSR limiting condition of operation (LCO) 3.4.7 requires the reactor to be shutdown with inoperable rupture disks. Therefore, the necessary course of action upon identification of a pump trip would be: 1) shut down the reactor, and 2) replace the primary system rupture disks.

Since AOP-9002 is entered upon a trip of a MCP during reactor operation, it has been revised to require shutdown of the reactor when a MCP trip has occurred. A precaution and limitation has been added to NOP-2106 to declare the rupture disks inoperable when any AC-powered MCP is not stopped simultaneously with other operating AC-powered MCPs. Taken together, these revisions provide necessary measures to ensure the TSR, as written, is satisfied by performing the necessary actions described above. RRD has committed to replacing the rupture disks if system monitoring indicates the deformation pressure has been reached in the system because of inadvertent pump trip, until such time that an approved change to the TSR is made.

(7) Does the change reduce the margin of safety?

The possibility that the margin of safety could be reduced was considered in the USQD following the discovery of the transient and the lack of a full analysis as part of the vessel life extension. As noted in this safety evaluation regarding USQD Question 1, the conditional probability of failure of the vessel and primary piping is still bounded. As well, the plan to replace rupture disks upon indication of a pressure above that of the deformation pressure continues to keep the facility within the bounds of the current

TSRs. Based on this evaluation, the facility maintains the bounded margin of safety as described in the USAR.

Based on the above, it is safe to continue operation of the reactor with the described procedure changes to simultaneously trip the primary coolant pumps during any planned trip of the pumps, and to replace the rupture disks in the event of a single primary coolant pump trip in accordance with current TSRs.

Upon receipt of DOE's approval of this safety evaluation, the safety basis covering operations will constitute the currently approved USAR, TSRs, and commitments included in this report.

The duration of operations bounded by this evaluation is not set by any of the various analyses. RRD believes that operations under this evaluation can continue for the period necessary to incorporate this new analysis into a formal USAR and TSR update that will include the final analysis discussed in Chapter 9. This update will be submitted to DOE no later than that of the next annual update submittal, currently scheduled for April of 2004.

11.0 References

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5. R. D. Cheverton, Delta 21 Resources, Inc., "Probability of Failure of the HFIR Primary Piping," RDC/ORNL/RRD/LTR-01-1, September 2001.

6. "Determination of the Applicability of the Hydrostatic Proof Testing for Demonstrating Integrity of HFIR Primary System Piping, Vessel, and Related Components," RDC/JRI/RRD/LTR-97-2/R1, September, 1998 (Rev. 1 update in review process).

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12. Memorandum, J. R. Inger to R. D. Cheverton and C. T. Ramsey, "Pressure Cycle History for the HFIR High-Pressure Primary System," July 16, 2001.

13. "Impact of Pump-Shutdown/Check Valve Closure-Related High Load/High Frequency Transients on HFIR Primary System Integrity," Memorandum, R.D. Cheverton (Delta 21 Resources, Inc.) to J. R. Inger, October 17, 2003.

14. *High Flux Isotope Reactor (HFIR) Technical Safety Requirements (TSR),* ORNL/TM-12841, R5.

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16. RRD Calculation ID: C-HFIR-2002-025, Determination of the Adequacy of BS&B Rupture Disks, Lot No. 95090001-1 in the HFIR High Pressure Primary Coolant System, August 30, 2002

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20. D. O. Harris, Engineering Mechanics Technology, Inc., *Summary of Win PRAISE Analyses of HFIR Piping Considering High Cycle Pressure Transients*, Report No. TR-03-10-1, October 17, 2003.

21. Memorandum, R. D. Cheverton to J. R. Inger, *Impact of Pump-Shutdown/Check-Valve-Closure-Related High-Load/High-Frequency Transients on HFIR Primary-System Integrity*, October 17, 2003.

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23. ORNL High Flux Isotope Reactor Inservice Inspection Report, Research Reactors Division, March 23, 1998.

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Appendix A

Meeting Notes With Dr. C. Sam Martin, Expert in Water Hammer Analysis

- Based on what we have seen from the plant instrumentation, the event we are looking at is check valve closure-induced water hammer.
- Based on review of the filtered/unfiltered plot of the PI-151 response, he is "almost positive" that the signal we recorded is not what the plant is seeing in the main 20-inch pipe. What we have downstream of the 4-inch pressure relief line is a geometry that is conducive to creating a high frequency resonance in the measuring line. In order to verify that diagnosis, Sam offered (and we accepted) to model the 4-inch line with a frequency domain code that he has, which will identify the possibility of measuring line resonances as a large part of what we see in the high frequency, high amplitude plot. We provided him drawings so he could do this work, and he thought he could do it in a few days, starting Monday.
- The hand calc that we did to estimate the magnitude of the initial pressure at the check valve seat is a good initial approach and should yield a conservative estimate of the pressure (the line deceleration should be slower than we assumed, and the real flapper closing dynamics may yield pressure increases less than the Joukowski equation). When the complete system model is finished, including a detailed model of the check valve, we will have a much better understanding of what the system pressure vs. time plots do. An accurate model of the check valve is key to understanding the pressure source term.
- The hand calc that used transmission factors to translate the positive pulse from the check valve to the vessel was a correct approach, and should yield a conservatively high pressure at the vessel, since the expansion into the vessel was not modeled. Again, the complete model will do this task correctly, include all the reflections, and give transient results.
- He is confident that "the vessel did not see anywhere near the magnitude of the pulse generated at the check valve."
- Simultaneous trip of all the pumps is a good idea, and is likely to not excite the same behavior in the sense lines. System assymetries and actual pump trip times may cause smaller water hammer events than we have seen thus far.
- Another idea on interim operational workarounds is to close the pump discharge block valve, and then trip the pump. After discussion that indicates this would cause a letdown block valve closure for us, he indicated that a partial closure of the block valve (as much as possible) will mitigate the backflow through the check valve and help reduce the water hammer.
- When check valve closure occurs, a positive wave moves down the cold leg toward the reactor, and at the same time, a negative pressure wave is generated on the backside of the check valve (of the same magnitude) that moves down the hot leg toward the vessel. There may be the possibility of check valve fluttering as the negative wave reflects off the spinning pump impeller and then hits the backside of the check valve, but after more discussion, this was characterized as not being a likely cause of the PI-151 unfiltered response. If check valve flutter occurs, its frequency would probably be less than the measured PI-151 response. This aspect of

the system performance needs to be investigated with the detailed check valve model that he plans to include in the system model.

- A hand calc similar to the one done for the cold leg will provide an estimate of the initial negative pressure wave that enters the vessel. Again, the complete model will do this task properly.
- The frequency of positive and negative pulses on the main 20-inch line and vessel should be lower than what we see on the unfiltered plot, and should be consistent with a/2L, which is in the 10-20 Hz range, rather than the 50-Hz range. The only real way to know a more complete answer to the question on pipe/vessel frequency response is to build and run the model, and/or instrument and take more data.
- He suggested the following instrumentation points when/if we do detailed system testing:
  - Check valve discharge
  - Reactor pressure vessel
  - Main 20-inch line
  - Pump speed vs. time
  - Venturi
  - Ultrasonic flow meter on pump discharge line
  - Pump suction piping
- Installation of sensors for the detailed testing should be within a few inches of the line/vessel where the pressure is being measured.

Dr. C. Sam Martin Background

WATER HAMMER. Professor Martin has been involved in many aspects of water hammer during the past 25 years. In addition to teaching graduate courses in the subject, he has offered a week-long short course to engineers for over 20 years. He has published numerous papers on the subject and has been active at many water hammer conferences. Moreover, Professor Martin has consulted with many firms on the subject of water hammer.

TWO-PHASE FLOW. In addition to the study of cavitation, Professor Martin has been active in the field of two-phase flow. He has conducted research, taught courses, and published papers on this topic, primarily regarding gas-liquid flows. Special topics of his expertise include slug flow, wave propagation, and pressure drop in two-phase conduit flows.

LIQUID CAVITATION. Professor Martin has had research and field experience regarding valve cavitation. His consulting work has led to his involvement with the effects and solutions of pump cavitation.

PIPE FLOW. Professor Martin has had considerable experience with the hydraulics of pipe flow. In addition to having had the normal exposure presented by teaching hydraulics to seniors and graduate students, he has worked on numerous pipe flow projects as a consultant.

OPEN CHANNEL FLOW. Professor Martin's experience with open-channel flow has come from teaching two graduate-level courses on the topic. He has also conducted laboratory experiments and hydraulic model studies for major clients of the Institute and in his consulting work.

EDUCATION

Year	*Degree*	*Subject*	*Institution*
1964	PhD	Civil Engineering	Georgia Institute of Technology
1961	MS	Civil Engineering	Georgia Institute of Technology
1958	BS	Civil Engineering	Virginia Polytechnic Institute and State University

WORK HISTORY

Years	*Employer*	*Title*
1974 to	Georgia Institute of Technology	Professor
1967 to 1974	Georgia Institute of Technology	Associate Professor
1963 to 1967	Georgia Institute of Technology	Assistant Professor
1962 to 1963	Georgia Institute of Technology	Graduate Research and Teaching Assistant
1961 to 1962	Georgia Institute of Technology	Ford Foundation Teaching Assistant
1960 to 1961	Georgia Institute of Technology	Graduate Research Assistant

CAREER ACCOMPLISHMENTS

Associations/Societies
Professor Martin is a member of the American Society of Civil Engineers, the American Nuclear Society, the International Association for Hydraulic Research and he is a fellow of the ASME.

Licenses/Certifications
He is a Registered Professional Engineer in the state of Georgia.

Awards/Recognition
He was the recipient of the Alexander von Humbolt U.S. Senior Scientist Award (1984-85) and the American Society of Civil Engineers J.C. Stevens Award (1977). Professor Martin was named the ASME's John R. Freeman Scholar (1970-71).

Publications and Patents
He is the author of numerous technical papers and book chapters. He is the editor or co-editor of many technical publications.

PUBLICATIONS

Author	*Year*	*Title*	*Publisher*
Martin, C.S. and Rao, P.V.	1984	Application of signal analysis to cavitation.	J. Fluids Engineering.
Martin, C.S., et. al	1981	Cavitation inception in spool valves.	
Martin, C.S. and Padmanabhan, M.	1979	Pressure pulse propagation in two-component slug flow.	J. Fluids Engineering. March, 101(1):44-52.
Martin, C.S., co-author	1978	Numerical analysis of pressure transients in bubbly two-phase mixtures	

Attachment 1

Inger, Joseph R.

From: Frank Mcquilkin [bsbtom@msn.com]
Sent: Wednesday, October 08, 2003 8:04 AM
To: Inger, Joseph R.
Subject: Fw: HFIR Disks

----- Original Message -----
From: Beair, Charles
Sent: Tuesday, October 07, 2003 5:18 PM
To: bsbtom@msn.com
Subject: HFIR Disks

Steven
Revised response to Joe's questions.
1) Exceeding the 90% rated/stamped pressure will not cause a disk to go high. Basic strength of materials support this fact.

2) Frequency of pressure cycles have been run at 1 sec to as long as minutes with no appreciable difference in cycle life on a disk.

3) The disk will burst if it is exposed to the rated maximum pressure in a one second period.
4) The RLS has a long histoy of performance. However, pressure pulse data ~~is~~ not readily available. We could produce the cycle test data if required as along as the frequency is in the 1 second plus range.

Regards
Charlie

10/14/2003

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