Attachment
Director of NRR
July 29, 1983

Program for Recirculation Piping Replacement at the Monticello Nuclear Generating Plant

Northern States Power Company

This report was prepared by the Bechtel Power Corporation. Additional information was supplied by the General Electric Company and the Quadrex Corporation.

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I. INTRODUCTION

A. Report Purpose and Scope

This report describes the program developed by Northern States Power (NSP) to replace the recirculation system piping and associated vessel safe ends at the Monticello Nuclear Generating Plant.

Detailed discussions are included concerning materials selected, codes to be applied and design bases evaluations to be performed. The replacement has been reviewed for conformance with applicable regulations and guidance. A preliminary safety evaluation has been performed and conclusions are included, along with information on the quality assurance program to be implemented and the final acceptance criteria.

B. Existing Configuration

The Monticello Nuclear Generating Plant is a single-cycle forced circulation, low power density, BWR-3 reactor licensed to operate at 1670 MWt.

Forced circulation through the reactor core is provided by the recirculation system. This system consists of two external loops together with associated pumps, and twenty internal jet pumps. The design basis of the existing system is:

Number of Loops	2
Pipe Size	28-inch
Design Pressure, Suction	1148 psig at 562°F
Design Pressure, Discharge	1248 psig at 562°F
Original Recirculation System Design Code	ASME B & PV Code Sec. I and USAS B31.1
Recirculation Pump Casing Design Code	ASME B & PV Code Sec. III, Class C

Figure 1 reproduced from the Monticello Updated Safety Analysis Report (USAR) depicts the arrangement of this system while Table 1 provides additional system data.

Piping materials currently installed were purchased in accordance with the NSSS supplier's specification, "Standard Requirements For Recirculation Loop Piping," issued for Monticello on November 11, 1969. Material used in the original design was purchased to one of the following requirements:

Components	Specification
Pipe materials	
Seamless pipe	ASTM A376, Type 304
Welded pipe (with filler marked)	ASTM A358, Type 304
Seamless or welded pipe (w/o filler marked)	ASTM A312, Type 304
Fitting Materials	
Socket welded type fittings or F316	ASTM A182, Grade F304
Forgings	ASTM A182, Grade F304
Welded fittings (with filler marked)	ASTM A403, Type 304
Cast fittings or CF8M	ASTM SA351, Grade CF8

Design, fabrication, testing and inspection were in accordance with ASME Section I with additional requirements of Section III and Section IX. The fabricated piping met the requirements of ASA B31.1 as of the actual date of purchase. Consideration was given to ferrite control although the date of purchase predated the issuance of Safety Guide 1.31. All fabrications were subjected to complete ultrasonic inspection and all welds were completely radiographed.

C. Monticello - Unique IGSCC History

During the course of routine in-service inspection on September 28, 1982, one shallow crack was found in the end cap of one of the recirculation system riser manifolds.

Upon further investigation, very small axial cracks were found at four additional locations. Three cracks were noted in the 12" riser to nozzle safe end welds and one was located in a riser pipe to elbow weld. All indications noted were confined to recirculation loop "A".

While preparing the recirculation safe ends for repair, through wall leakage developed in each of the safe ends. An additional small through wall axial crack was located through hydrostatic testing. The cracks were determined to be due to intergranular stress corrosion cracking (IGSCC). Weld overlay repairs were made (full wall thickness) where cracking was found and the unit was returned to service following NRC review and approval of the repairs.

As a result of these findings, Northern States Power entered into discussions with General Electric Company and Bechtel Power Corporation and developed a repair strategy.

D. Industry Solution Development Program

In 1975, General Electric Company issued NEDO-2100 "Investigation of Cause of Cracking in Austenitic Stainless Steel Piping." This report summarized GE's understanding of the problem up to that time and acknowledged that IGSCC was occurring in the theoretically most susceptible material, type 304, which had been cold worked, sensitized, and installed with high welding residual stresses. This study noted that of the 82 IGSCC cracking incidents up to that time, 62 were located in the heat affected zone of type 304 piping systems. The remaining cracks were located in severely furnace sensitized piping components, primarily safe ends.

During this same period, Bechtel had investigated the IGSCC problem independently and recommended to their clients that piping systems susceptible to IGSCC in nuclear service be fabricated from type 304L (0.03% carbon maximum) where possible, and that solution heat-treatment be used if possible after welding or other sensitization.

In 1981, EPRI NP-1823, "The Final Report on Fabrication-Related Sensitization" expanded the understanding of this phenomenon by including a discussion on low-temperature sensitization and the influence of thermal strain history.

As a result of information developed during these development programs we believe the precursers of IGSCC are known and can be reduced or eliminated. Our piping replacement program will call for:

1) replacement of all original type 304 material with type 316 low carbon (less than 0.02%) material in both loops,

2) modificaton of the piping layout to reduce the number of welds and residual and thermal stresses, and

3) use of last pass heat sink welding for all field welds.

E. Identification of Principal Agents and Contractors

Principal agents involved in the pipe replacement efforts are:

1. Northern States Power Company (NSP) Nuclear Engineering and Construction Department - Monticello, Minnesota

   NSP is the sole owner and licensee of the Monticello Nuclear Plant and is the overall project manager.

2. Bechtel Power Corp.
   San Francisco Power Division
   San Francisco, California

   Bechtel was the original architect/engineer at Monticello and will provide engineering, procurement, and related services (expedition, inspection, surveillance and quality assurance review of vendors) and documentation. As requested by the NSP Project Engineer, Bechtel acts as the agent of NSP in procuring piping material for this project and will generate major portions of the required engineering software.

3. General Electric Company (GE)
   San Jose, California

   GE is a supplier to NSP for the riser inlet safe end replacement and miscellaneous parts. GE will also serve as safe end and piping installer at Monticello.

4. Quadrex Corporation
   Campbell, California

   Quadrex has been selected to supply the engineering and equipment to decontaminate the reactor coolant system and manage waste.

II. REPLACEMENT COMPONENT DESIGN AND FABRICATION

A. Material Selection

Replacement piping for the recirculation system will be nuclear grade type 316 stainless steel with 0.02% maximum carbon and nitrogen added to meet strength requirements of type 316. Grain size will be limited to two.

B. Applicable Codes and Standards

Consistent with 10 CFR 50.55a for a plant whose construction permit was issued prior to Janaury 1, 1971 the existing recirculation system piping for Monticello was designed and fabricated in accordance with ASME Boiler and Pressure Vessel Code Section 1 and USAS B31.1 Code for Pressure Piping. The recirculation pump casings were designed in accordance with ASME Section III Class C while the vessel safe ends meet the requirements of ASME Section III Class A.

Requirements for repairs such as those proposed for Monticello are governed by ASME Code Section XI IWA 4000. As a minimum, replacements must meet the requirements of the edition of the Construction Code to which the original component or part was constructed.

Alternately, replacements may meet all or portions of the requirements of later editions of the Construction Code. Northern States Power has opted to use later additions of the Code to replace the recirculation piping, consistent with IWA 7210(c) of Section XI and NCA-1170 of Section III which allows the use of later editions of the Code. The replacements will be consistent with the guidance continued in Regulatory Guide 1.84, 1.85 and 1.147.

Replacement piping including safe ends to be installed will be fabricated and installed in accordance with the requirements of ASME Section III Class I. The effective code date for material has been selected as Section III 1980 thru Winter 1981 addenda (Winter 1980 for inlet safe ends). The replacement will be consistent with the guidance contained in Regulatory Guide 1.31, 1.43, and 1.44.

Replacement piping will be re-baselined for Inservice Inspection consistent with the requirements of Section XI, 1977 edition.

C. Design Conditions

The design temperature for the recirculation system piping including outlet safe ends is 562°F. The design pressure for the recirculation safe ends and suction piping is 1148 psia and for the discharge piping it is 1248 psia.

Satisfaction of normal condition stress levels based on the above operating conditions will be determined in accordance with ANSI B31.1 1977 through Winter 1978 Addenda. The stress from sustained plus thermal expansion shall not exceed the limits of paragraph 104.8.3B of that code. While OBE stresses will be combined in accordance with paragraph 104.8.2 where k will be given a value of 1.2, and SSE stresses will be similarly evaluated except k will be allowed to take on a value of 2.4.

Piping-to-vessel inlet safe end nozzle stress will be evaluated in accordance with ASME Section III, 1980 thru the Winter 1980 Addenda. Stress allowables will be those given in the piping design specification. Design and conditions for the inlet safe ends are those for the vessel (1391 psig and 575°F per GE design specification for the safe ends).

To validate the overall design, the stress levels developed for B31.1 code, compliance will be reviewed in accordance with NB3600 of ASME Section III, 1980. During the Section III analysis, piping will be evaluated against various operating pressures, temperatures and flow rates representative of time dependent thermal transients which the piping system may encounter during a 40-year lifetime. This analysis will consider normal, upset and emergency conditions.

D. Inlet Safe End Modifications

1. Design Conditions

The replacement safe end and transition piece for the recirculation inlet nozzle will be furnished as ASME Section III, Class 1 material. For the safe end, this is consistent with the classification of the original safe end since it was supplied with the reactor vessel under the vessel ASME code stamp. The transition piece, if used to replace the thermal sleeve reducer, is not a primary pressure boundary component but for convenience has been designed to have the same classification as the safe end.

In the design of the safe end, mechanical pipe reaction loads and thermal sleeve reaction loads have been specified. Due to the fact that the attached piping system is being modified, the pipe reaction loads have been established based on the structural capability of the attaching pipe material. This allows the safe end to be designed within the ASME code design requirements and provides flexibility in completing the piping design. The thermal sleeve reaction loads that will be used are the same as used in the original nozzle design.

The thermal cycles, which have been specified for the safe end thermal and fatigue analysis, represent only the operating cycles that involve a temperature transient in the nozzle. These transients have been categorized into service levels, which are required by the ASME Section III code, based on the frequency of the thermal transient and the function of the nozzle during the transient.

The functional requirements which will be met for the safe end design are that it must interface with the existing nozzle and thermal sleeve and the replacement piping without affecting the performance of the recirculation system and the reactor vessel. Additionally, the safe end replacement is to be accomplished without affecting the fitup and operation of the components which are a part of the jet pump. This will require control of the alignment of the remaining inner thermal sleeve pipe in the nozzle during the repair.

The environmental conditions used on the exterior of the nozzle are the same as those used for the original nozzle design. The design life of the replacement safe end and transition piece is forty years.

2. Inlet Safe End Modification

The existing recirculation inlet nozzle at Monticello is shown in Figure 2. The existing nozzle safe end has a tuning fork geometry which attaches to the internal vessel piping which also functions as a thermal sleeve for the nozzle. This piping contains a reducer which provides the dimensional change in diameter to connect to the jet pump elbow and riser pipe. There is also another internal sleeve in the safe end that is held in place by a flat plate spring and ring nut that has a threaded connection in the safe end. A set screw is tack welded in place to prevent loosening of the ring nut. The safe end and all internal components in the nozzle are stainless steel type 304 materials except for the flat plate spring, which is Inconel X-750 material.

The design of the replacement safe end is shown in Figure 3. In this figure, two different replacement configurations are shown. Only the safe end is planned to be replaced, as shown in the upper figure, if the internal pipe reducer welds do not show evidence of intergranular stress corrosion cracking (IGSCC). In the event that the internal welds have IGSCC, the alternate design shown in the lower figure will be installed. Both of these safe end designs will be supplied using nuclear grade stainless steel materials which have been shown to be resistant to IGSCC.

In each of the designs, the following is accomplished:

a. The structural intergrity within the nozzle is re-established such that the function and servicing of the jet pump components is not affected.

b. The inner sleeve is removed resulting in a creviced condition in the original design. In the original design, the purpose of the inner sleeve was to provide thermal protection to the safe end during a sudden start of a cold recirculation pipe loop. Since technical specification requirements prevent this transient condition from occurring, this is a highly unlikely operating condition. Therefore, this transient is classified as a service level C event under ASME code requirements. Under this classification, all ASME design requirements can be met for the replacement safe end without the inner sleeve.

c. Since both the original and replacement safe ends are stainless steel material of similar geometry, the design of the interfacing components is not affected.

d. All safe end attachment welds are full penetration welds which can be performed using automatic welding equipment.

e. Access for ultrasonic examination is provided on the safe end for the primary pressure boundary attachment welds.

f. At the nozzle safe end weld end, the inner sleeve has a uniform wall thickness in order for double wall radiography to be performed.

g. With either of the replacement safe ends, flow through the safe end will have slightly less hydraulic resistance with the removal of the inner thermal sleeve.

E. Design Improvements

Design improvements consist of:

- Use of bent pipe for 12-inch risers. This eliminates 20 welds. See Figures 4 and 5.

- Deletion of end caps by using a bent pipe and reducer on ends of 22-inch headers. This eliminates 4 welds.

- Use of seamless 12-inch pipe. This eliminates inspection of longitudinal welds.

- All spools to be solution annealed after shop welding.

The recirculation piping was redesigned to eliminate as many fittings as possible and utilized bent pipe in order to reduce the number of welds. A material was specified that has a low carbon, high nitrogen chemistry to minimize the possibility of IGSCC. To further minimize this problem, the specification required solution annealing of the shop welds after fabrication.

The crossover piping and valves that presently connect the two 22" manifolds will be eliminated. Operational experience has shown this connection to be without significant utility. Technical Specification preclude operation of the reactor with the crossover line open for extended periods of time. Four-inch equalizing lines will be installed on the RHR suction and discharge lines. Since the loop crossover piping is being eliminated, use of these equalizing lines will facilitate cool down with an inoperable recirculation pump or an isolated loop.

The physical location of the field welds will be improved to allow easier access to automated ISI equipment. This is expected to significantly reduce future inspection time and inspector radiation doses.

F. Weld Procedures

The field welding procedures will use heat sink welding wherever possible and will be qualified to the maximum joint thickness necessitated by the design. Automated equipment will be used to improve weld consistency and minimize welder radiation exposures.

G. Nuclear Class I Component Evaluation Based on ASME Section III

1. Piping System Fatigue Evaluation
The design of this system is committed to the ANSI B31.1 code 1977 through Winter 1978 Addenda. To validate the design, stress analysis shall be performed in accordance with the ANSI B31.1 Code. However, the Nuclear Class 1 requirements will also be verified in accordance with the rules of NB-3600 of the ASME Section III Boiler and Pressure Vessel Code 1980 Edition including Addenda through Summer, 1982. Design, level A, level B, level C and level D service limits shall be considered in the analysis.

The primary loadings considered are weight, earthquake, and other design mechanical loads. All the loadings classified as level A and B service limits, including thermal expansion range, thermal gradients ($|\Delta T_1|, |\Delta T_2|$), thermal gross discontinuity ($|\alpha_a T_a| - |\alpha_b T_b|$), earthquake, anchor movements, and other mechanical loads, are considered in calculating primary plus secondary stress intensity range and peak stress intensity range. Stress indices in Table NB-3681 (a)-1 shall be used in qualification of all the piping products and joints in this system. The following will be evaluated in the analysis of the piping system:

a. Pressure design will be in accordance with rules in NB-3640 to ensure that minimum wall thickness required to sustain the internal design pressure has been achieved.

b. Primary stress intensity limit of Equation (9) in NB-3652 will be met for service levels A, B, C and D.

c. Primary plus secondary stress intensity range limit will be met by satsifying the requirement of Equation (10) in NB-3653.1. If the stress range calculated by Equation (10) exceeds $3S_m$, the simplified Elastic-Plastic Analysis [Equations (12) and (13)] is performed to qualify the piping system.

d. The cumulative damage will be evaluated in accordance with NB-3653.5 based on Equation (11) (Peak Stress Intensity Range) and Equation (14) (Alternate Stress Intensity) in NB-3653.2 and NB-3653.6 (c) respectively.

2. Inlet Safe End ASME Stress Analysis
For the replacement inlet safe end designs, a detailed design report in accordance with ASME Section III NB-3000 is being performed using the ANSYS finite element computer program. In addition to the safe end and transition piece, the computer model also includes the nozzle, inner thermal sleeve piping in the nozzle, and a segment of the attaching pipe and the vessel wall.

In addition to the ASME code analysis, a stress rule index calculation will be performed at the weld location to the existing internal thermal sleeve piping. This is being performed to establish the susceptibility of materials to IGSCC which will remain in service.

H. Piping Stress Analysis

Due to the replacement, the recirculation piping system will be reanalyzed. The recirculation piping and modification of the discharge inlet header including the RHR supply and return lines are to be included in the analysis. Results of the following analyses will be compared with appropriate acceptance criteria listed in Table 2, taken from the ANSI B31.1-1977 Power Piping Code with all the addenda up to and including Winter 1978.

1. Seismic Analysis
The recirculation piping system will be modeled as a lumped mass system with enough details to accurately predict results for piping dynamic response up to 33 Hz. The weight of piping contents plus insulation will be added to the weight of the pipe in the form of a uniformly distributed load (lbs/ft). For the pump motors and valve operators, the extended mass will be modeled as an additional weight at the respective center of gravity.

The stiffness of each support, either from hand calculation or computer analysis, will be included in the modeling.

The OBE response spectra for the analysis is to be the same as used by G.E. in the original analysis; 0.5% damping for OBE. SSE loads are twice the OBE loads, the modal responses will be combined by the square root of the sum of the squares. The vertical direction earthquakes will be considered using a static coefficient of 0.04g for OBE, the two horizontal direction earthquakes are considered to act independently, and each is absolutely summed with the vertical direction earthquake.

2. Thermal Analysis
   Nine different thermal modes that will simulate various thermal loads at normal and upset emergency and faulted conditions will be incorporated in the thermal analysis of the recirculation piping system.

III. SAFETY AND LICENSING EVALUATION

Replacement of the recirculation piping will not result in any unreviewed safety question. It is anticipated that the plant can be operated in accordance with the current operating license with only minor modifications to the existing Technical Specifications. No impact on the accident evaluations contained in Section 14 of the USAR is anticipated as a result of the proposed replacement.

No significant difference in recirculation flow will occur as a result of the replacement since piping size and effective pipe length are essentially unchanged. Hydraulic analyses have determined that the riser flows will change less than 1% and the system resistance change is negligible. Therefore core flow distribution will remain unchanged after the replacement. Nevertheless, during startup, the operating map will reverified and if necessary the jet pump surveillance data base may need to be adjusted as a result of the replacement.

Since support configuration peak stress points are unchanged, LOCA and HELB effects will also remain unchanged. Nevertheless, a reevaluation of pipe whip effects is being undertaken. Any required modifications to the existing restraints will be included in the replacement program. NRC requirements in this area are currently being reviewed to determine if some simplification or reduction in pipe whip restraints is possible.

Piping stress calculations described above will demonstrate that stress levels in relationship to material allowable stresses are not increased over those currently evaluated. This assures that the likelihood of pipe rupture is reduced by the proposed replacement.

Pursuant to 10 CFR 50.71, the Monticello USAR is updated on an annual basis. The appropriate sections of the USAR will be updated in that revision which follows by at least six months the return to full power of the unit following the modification.

No changes in operating parameters are anticipated as a result of these repairs. Replacement of existing 304 material with low carbon 316NG can be expected to measurably enhance plant safety. This is because the possibility of reactor pressure boundary leakage will be significantly reduced as a result of the reduction or elimination of IGSCC that can be expected. While plants using materials similar to that currently being used at Monticello can be safely operated, adequate evidence exists to conclude that the possibility of cracking during the life of the plant will be greatly reduced as a result of the proposed modification.

IV. QUALITY CONTROL AND ASSURANCE PROGRAM

The replacement piping is essential to nuclear safety. It is an integral part of the reactor pressure boundary and a flow path for an emergency core cooling system (ECCS) and normal shutdown cooling. As such, it is subject to full nuclear quality assurance and quality control programs.

The requirements of 10 CFR 21, "Reporting of Defects and Non-Compliance," apply to the owners, designers, material suppliers, fabricators and installers of the subject piping.

V. INSTALLATION AND TESTING

A. Outage Configuration

Prior to initiation of work, the entire core will be off-loaded and stored in the spent fuel racks. Adequate space is available and the spent fuel pool cooling system is adequately sized to ensure that pool temperature will not exceed 120°F.

B. Radiation Management

A major consideration of NSP in planning the replacement program is reducing radiation exposure to worker to as low as reasonably achievable (ALARA) levels. General Electric has reviewed their experiences and has produced an upper bound estimate of 2000 man-rem of occupational exposure for the proposed work. Offsite doses to the public are negligible.

The following elements are considered in the ALARA Program for the replacement effort:

1. Optimally Positioned Control Blades.
Control blades will be positioned in the vessel in such a manner as to minimize worker exposure. Control rod placement as well as shielding design will be based on extensive General Electric studies of in-vessel source terms.

2. Reactor Water Cleanup.
After core off-load, the reactor water cleanup system will be operated to reduce, to the extent practical, the bulk reactor water activity level

3. Repair Replacement Designs.
Designs are being prepared to reduce the time required to remove and install hardware. Designs will reduce to a minimum the number and complexity of field welds. Built-in aids for safe end alignment will reduce installation time.

4. Decontamination.
Safe end thermal sleeve annuli will be in-situ decontaminated prior to work initiation using a process known as "water-lancing". This process can reduce surface dose levels by a factor of 10 (Figure 10).

Loop piping and components will be decontaminated in-situ using a chemical process. This process is described in Appendix A to this report. Quadrex Corporation has been selected as the contractor for this work. A reduction in man-rem exposures of up to a factor of five is possible based on the experience of others who have performed in-situ decontamination.

5. Ventilation
   Ventilation of the dry vessel internals through controlled filtration paths will mitigate airborne activity problems

6. Vessel External Shielding and Local Shielding
   Previous external shield designs were constructed of multiple blocks requiring assembly in the nozzle area which caused accountability problems. General Electric is designing improved shielding.

7. Work Sequence
   Hot spots will be removed first where possible. Work sequence will be optimized to do the most work under the lowest exposure conditions.

8. Work Areas
   Access routes and work areas will be well marked. Auxiliary services will be provided to speed work (e.g. communications, ventilation and lighting systems).

9. Vessel Flooding
   Previous vessel safe end and piping replacement work was performed with the reactor vessel drained. General Electric has designs in progress to provide additional vessel floodable capability (Figures 6 through 9). Optimum placement of control rods combined with annulus and shroud area flooding can result in large reductions in dose rates in the work area.

10. Mockups
    Mockups are being designed by General Electric to assist in tooling design, welding procedure development and welder qualification. Substantial ALARA benefits will be gained through practice with full size mockups. A preliminary design is shown in Figure 11.

    All work will be performed subsequent to the issuance of radiation work permits by the plant Radiation Protection Group. All work will be performed in accordance with NSP approved rules and procedures for maintenance work in controlled areas.

C. Field Work

Consistent with the requirements of ASME Section XI, installation will be performed in accordance with the requirements of ASME Section III, Article NB-4000, as specified in installation specifications developed by General Electric and Bechtel.

All work performed on site will be under the general supervision of NSP plant management. All work will be performed in accordance with procedures developed for NSP by General Electric, Bechtel, or Quadrex and approved by NSP management. Normal plant procedures for the review and approval of design change packages will be followed.

D. Welder and NDE Inspector Qualification

Qualifications will be in accordance with ASME Section III and hence, ASME Section IX and V respectively.

E. NDE and ISI Baseline Program

Fabrication and installation will be in accordance with Section XI-1977, augmented by the ASME Section III - 1980 Quality and NDE Requirements. Seamless and welded without filler metal tubular products and fittings will be 100% ultrasonically inspected after heat treatment. Longitudinal and girth weld in tubular products and fittings shall be 100% radiographed after final heat treatments. All field welds will be 100% radiographed in accordance with ASME Section III, NB-5222 prior to hydrotesting.

Preservice examination of 100% of the pressure retaining welds will be performed in accordance with Section XI, IWB 2000, 1977 Edition through Summer 1978 Addenda upon completion of Section III work but prior to system hydro as provided by IWB 2200.

F. Old Pipe Disposal

Two options have been considered for dealing with removed pipe. The first is to treat it as radioactive waste. If this option is chosen after removal from the drywell, the pipe will be measured for contamination levels than suitably packaged and shipped to a licensed radwaste burial facility. The second option involves similar treatment up to shipment but in this case pipe is transferred to an offsite decontamination facility. There its contamination will be removed to allowable release levels. The clean pipe will then be sold as stainless steel scrap for ultimate remelting and recycling.

As adequate hot storage space is availabe onsite, no immediate decision on ultimate pipe disposal need be made and a through safety and economic evaluation can be made before commitment to any disposal course must be made.

The ultimate choice of disposal method will be based on current burial volume allotments and overall disposal economics. In any case, the current in-place radiation monitoring and radioactive material handling procedures will be followed.

G. Hydrotesting and Restart Testing

Replacements performed in accordance with ASME Section XI, Article IWA-7000, are required to be subjected to a System Hydrostatic Test in accordance with Article IWB-5000.

As discussed, recirculation flow should not be measurably changed as a result of these repairs. Nevertheless, since the repair will require installation of a new flow nozzle, recalibration of the recirculation flow instrumentation system will be performed prior to full power operation.

The hydrotest of the system will be performed in accordance with ASME Section XI, and pre-startup testing will be performed in accordance with GE procedures.

VI. CONCLUSIONS

The Monticello recirculation piping replacement program will meet all applicable codes and standards. Material procurement and installation will be performed in accordance with a comprehensive quality assurance program. An effective ALARA program will be in place to reduce radiation exposures to workers. The modified recirculation system will utilize materials and construction techniques that will effectively eliminate IGSCC as a concern for the life of the plant.

TABLE 1

REACTOR RECIRCULATION SYSTEM
(Original)

RECIRCULATION PUMPS	
Number	2
Type	Vertical, centrifugal, single stage, variable speed
Power rating	4000 hp
Flow rate	32,500 gpm/pump
Design pressure and temperature	1400 psig @ 575°F
Total developed head	400 ft
Design code	ASME B and PV code, Sec. III-C
RECIRCULATION VALVES	
Number	
Recirculation	Four 28-inch
Equalizer	Two 18-inch
Type	Motor operated gate
Design code	USAS B 31.1.0
JET PUMPS	
Number	20
Material	Type 304 stainless steel
Overall height (top of nozzle to diffuser discharge)	20 ft 10 in.
Diffuser diameter	14-3/4 in.

TABLE 2

Loading Combinations

Loading Combination	Service Limit	Stress Limits	ANSI B31.1-1977 Para. References
PD+DW	Sustained	$1.0S_n$	104.8.1 EQ. 11a
PD+DW+OBE	Occasional	1.25_H	104.8.2 EQ. 12a
PD+DW+SSE	Faulted	$2.4S_H$	FSAR
TH+SAM*	Thermal Expansion	S_A	104.8.3A EQ. 13a
PD+DW+TH+SAM*	Sustained plus thermal expansion stress	$S_A S_H$	104.8.3B EQ. 14a

* Either Equation 13a or Equation 14a must be met.

PD - Design pressure

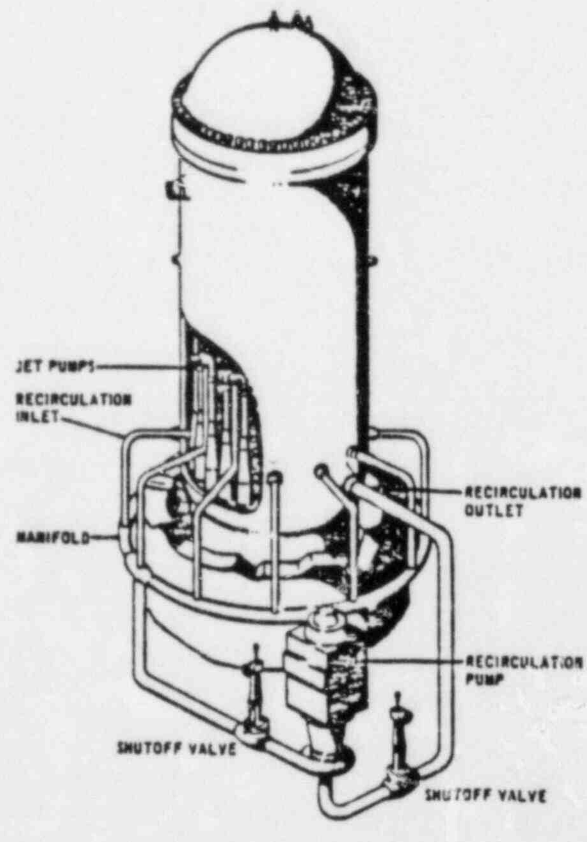
DW - Piping deadweight

OBE - Operational basis earthquake

SSE - Safe shutdown earthquake

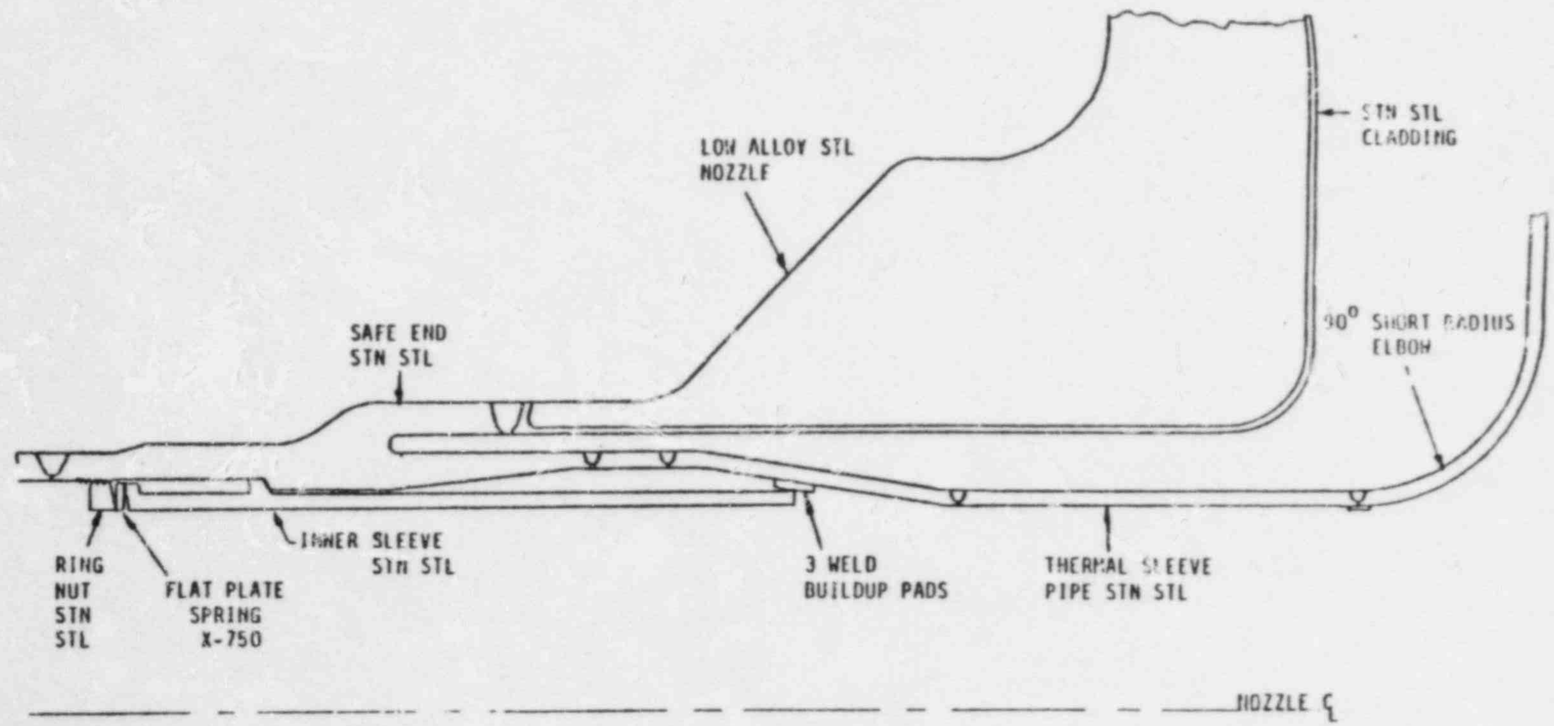
TH - Loads due to thermal expansion of the pipe

SAM - Seismic anchor movement



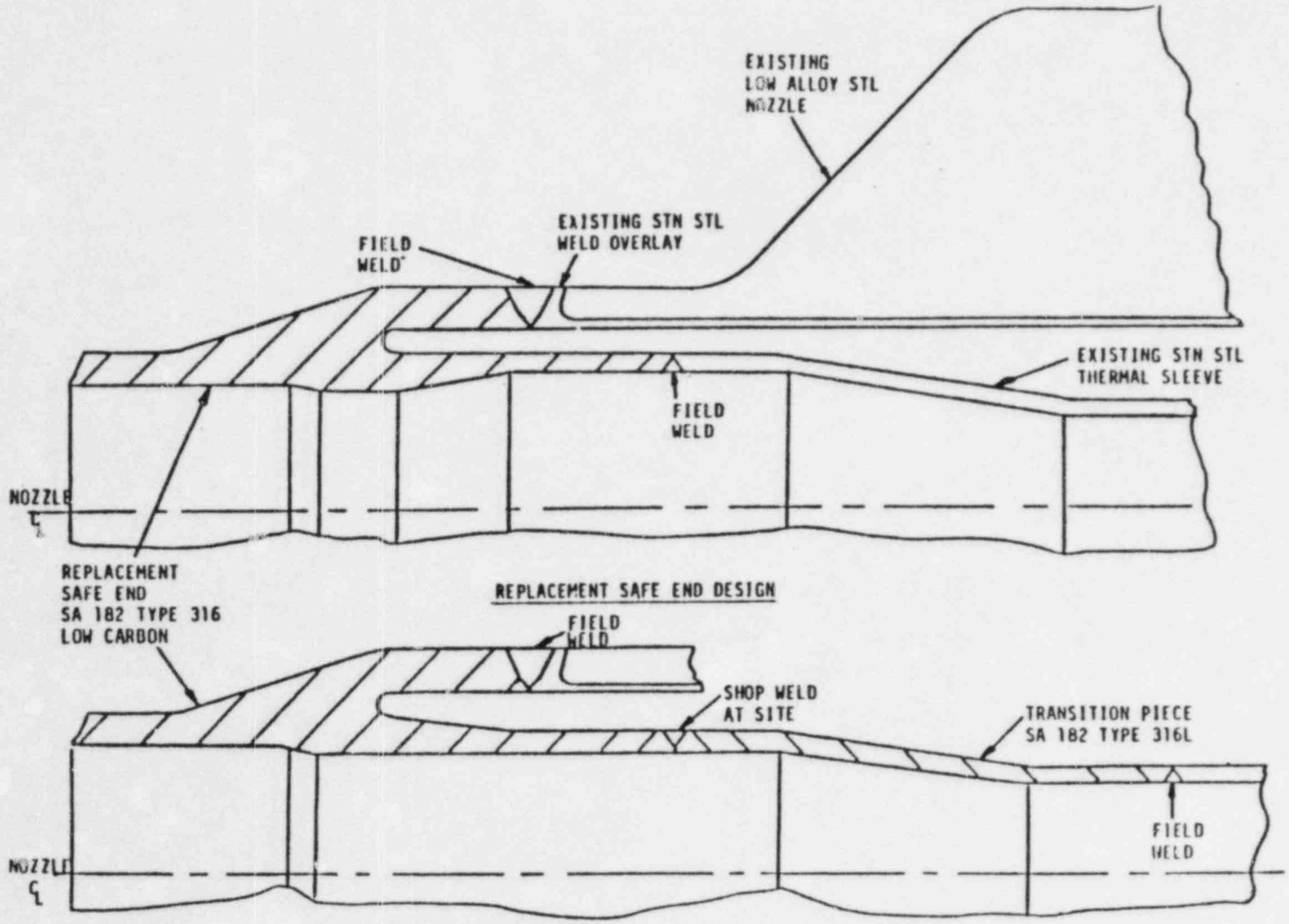
REACTOR COOLANT RECIRCULATION SYSTEM ISOMETRIC

FIGURE 1



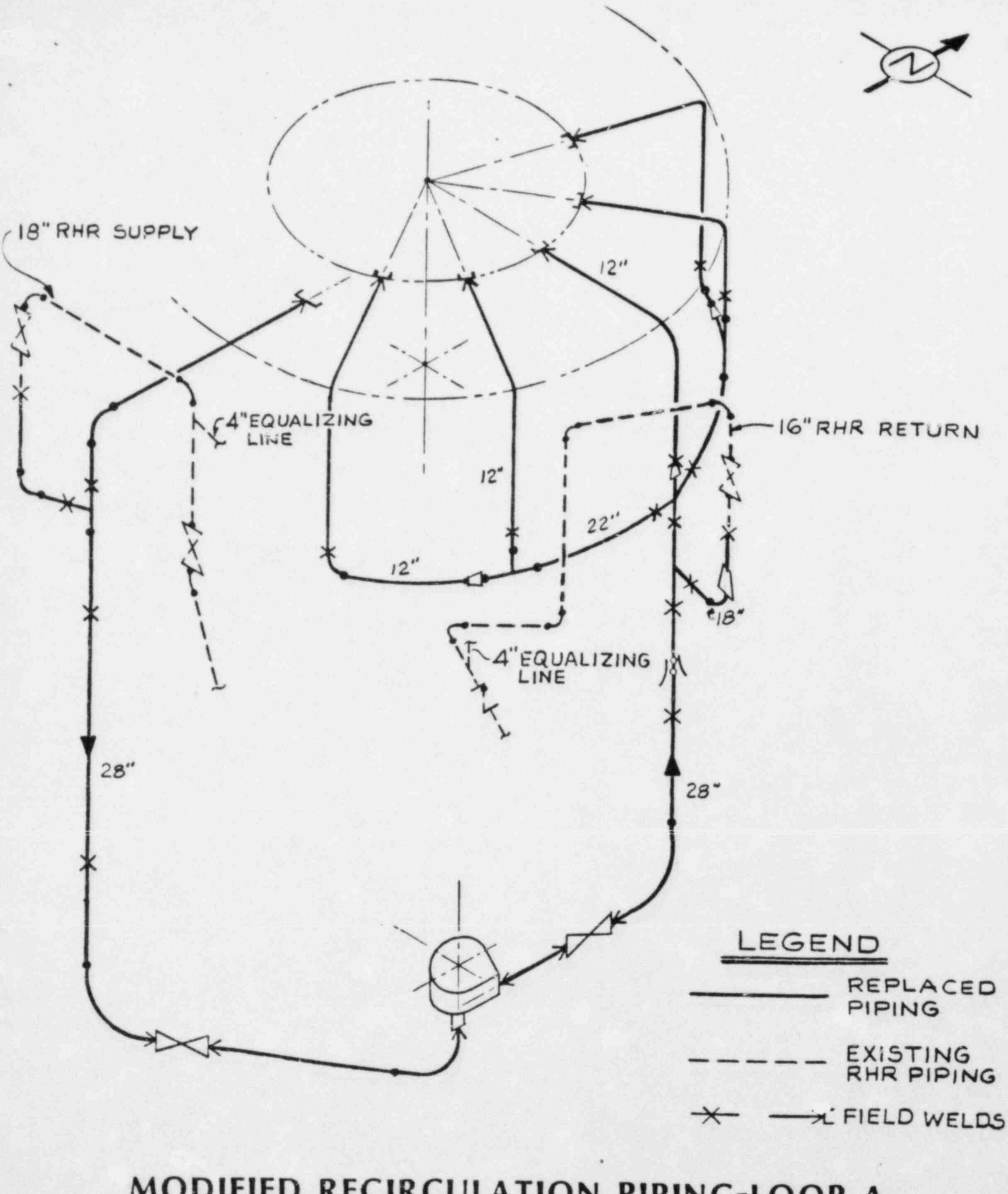
EXISTING RECIRC INLET NOZZLE

FIGURE 2



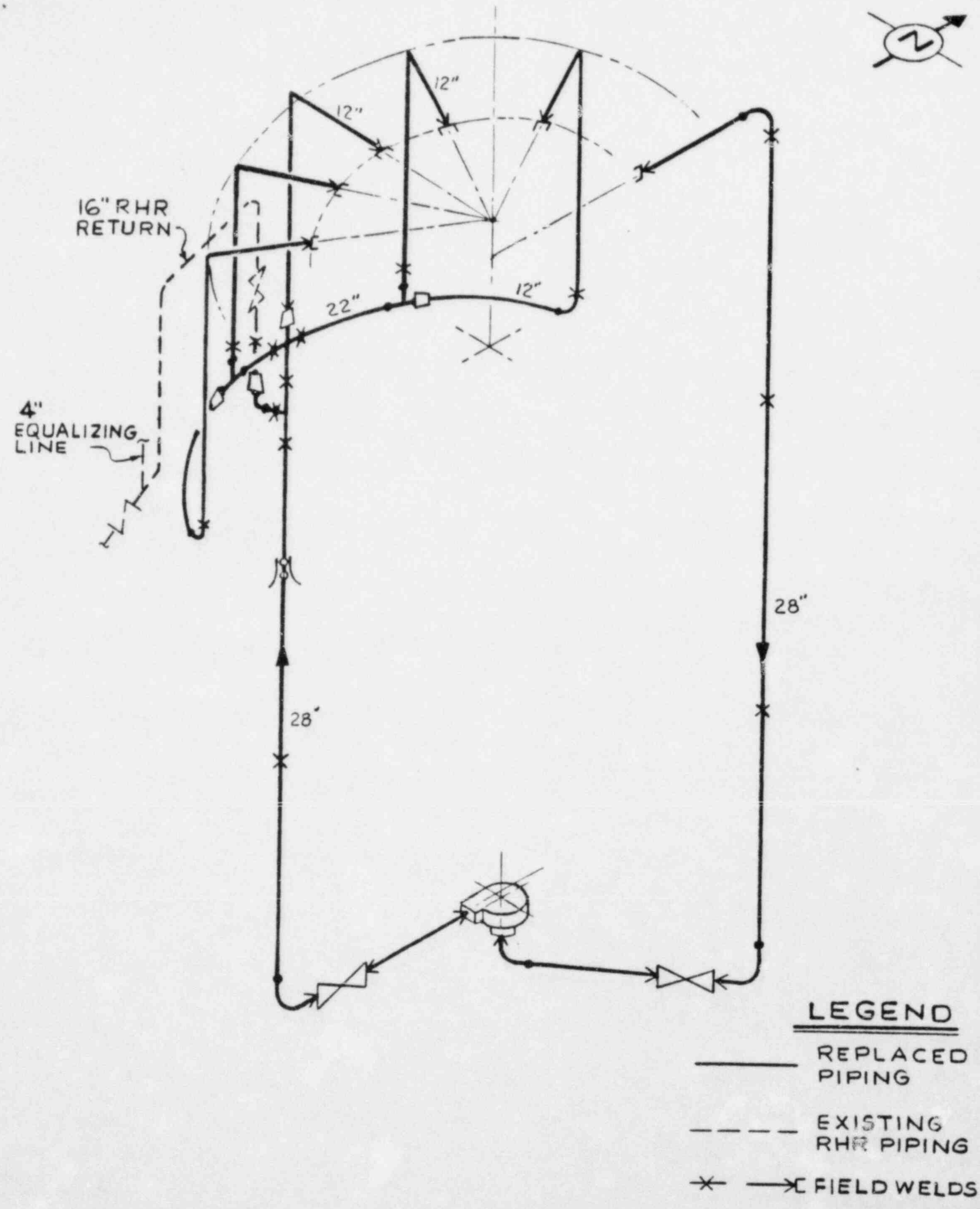
ALTERNATE REPLACEMENT DESIGN

FIGURE 3



MODIFIED RECIRCULATION PIPING-LOOP A

FIGURE 4



MODIFIED RECIRCULATION PIPING-LOOP B

FIGURE 5

PREVIOUS REPLACEMENTS

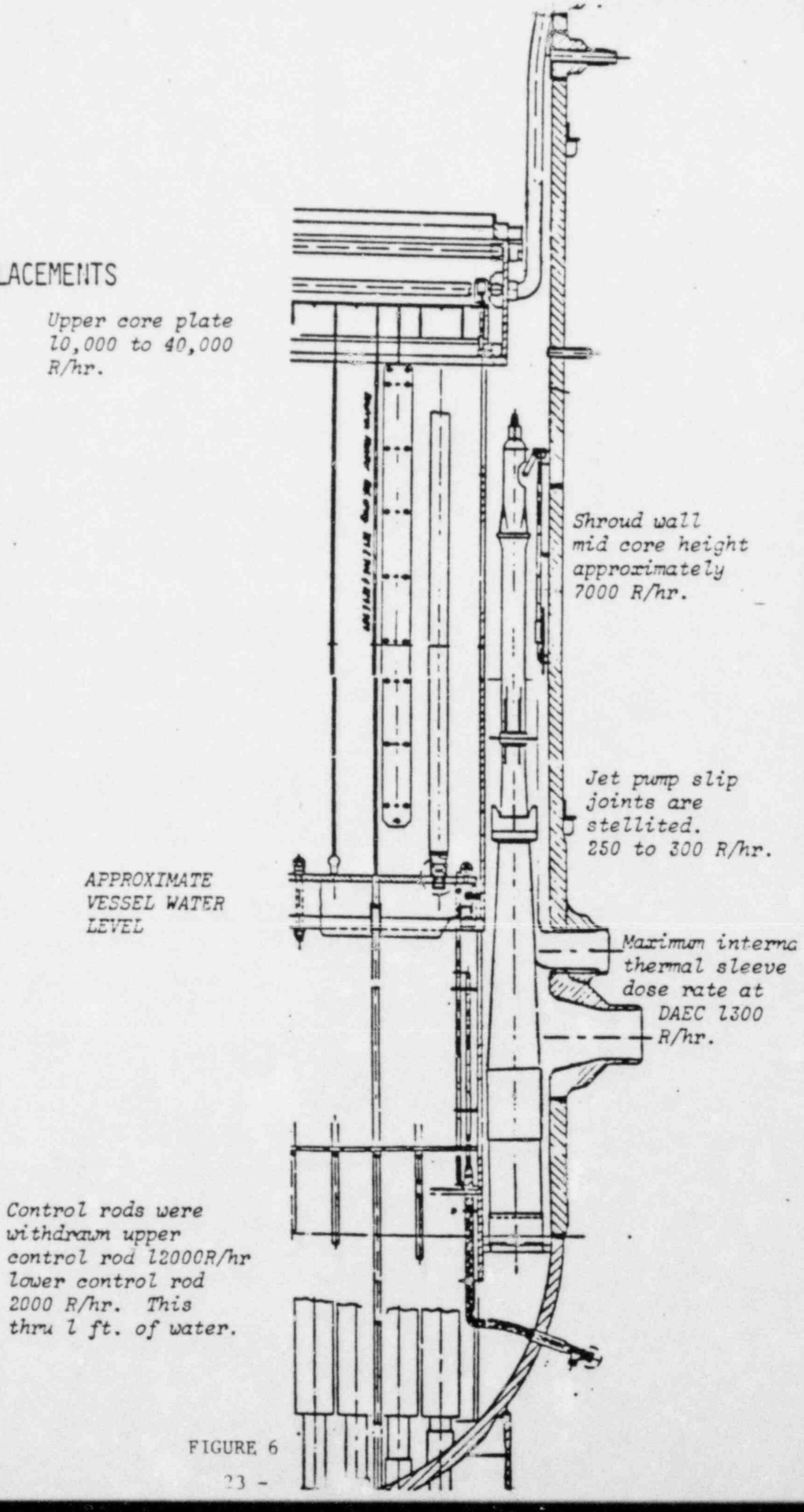


FIGURE 6

PIPING REPLACEMENT

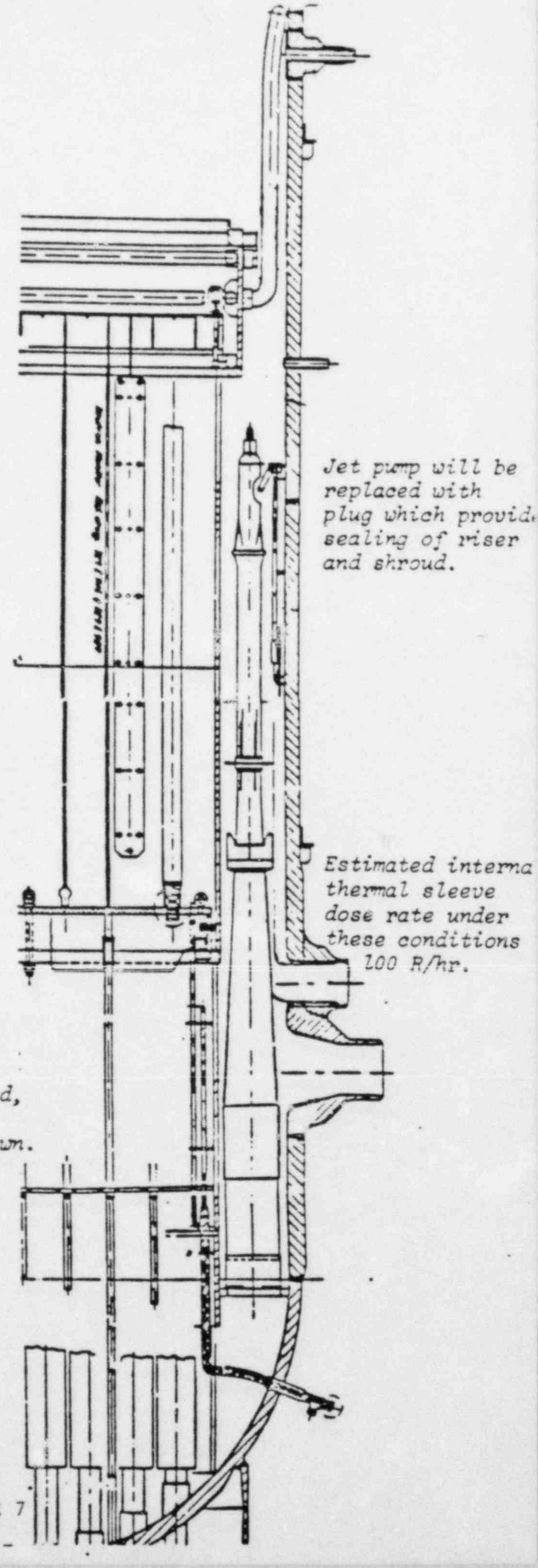


FIGURE 7

SAFE END REPLACEMENT

Annulus is drained shroud remains full. Trickel flow through CRD's will maintain CRD and vessel water quality. Fixed over flow approximately 12 inches above upper grid plus operator control of flow to CRD's will maintain shroud level. Jet Pump defusers will provide shielding since they will be flooded at all times. Special level instrumentation will provide shroud level indication in the control room and in the work area. High and low shroud level alarms in these two areas will be included.

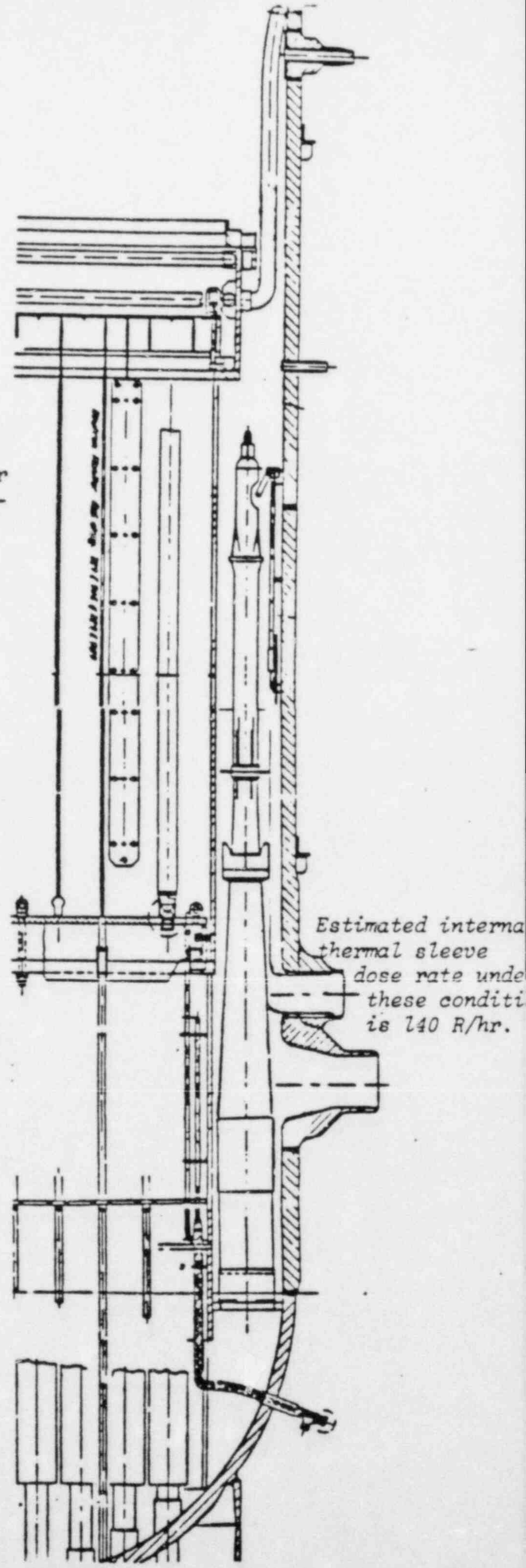
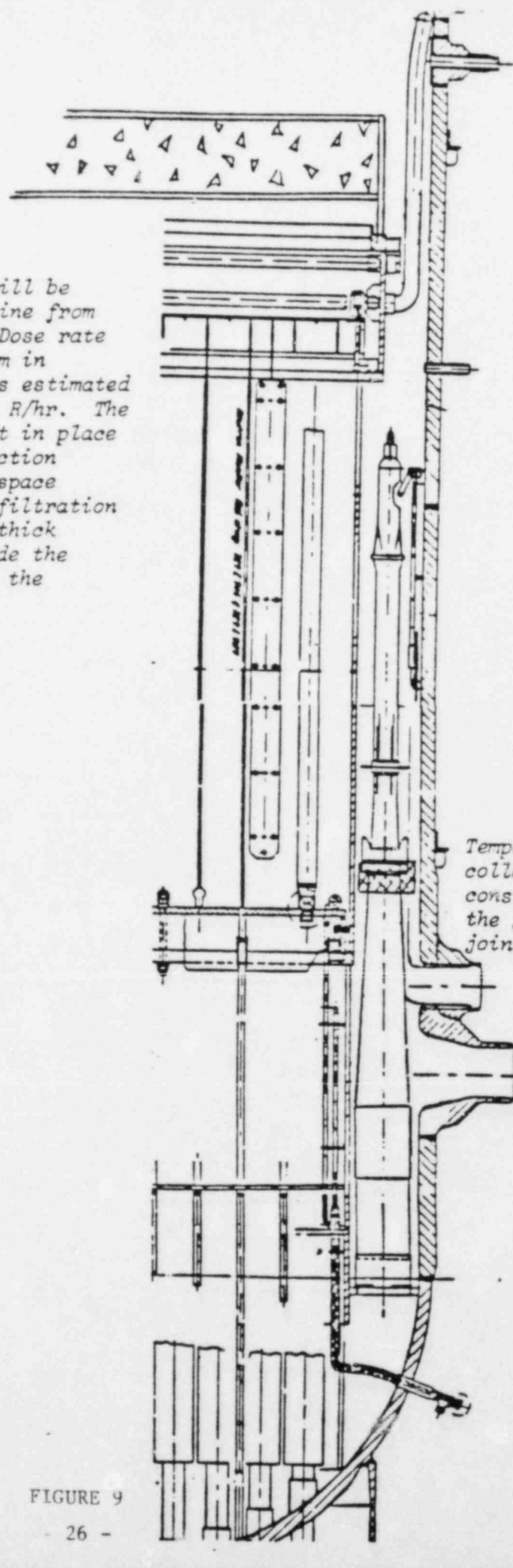


FIGURE 8

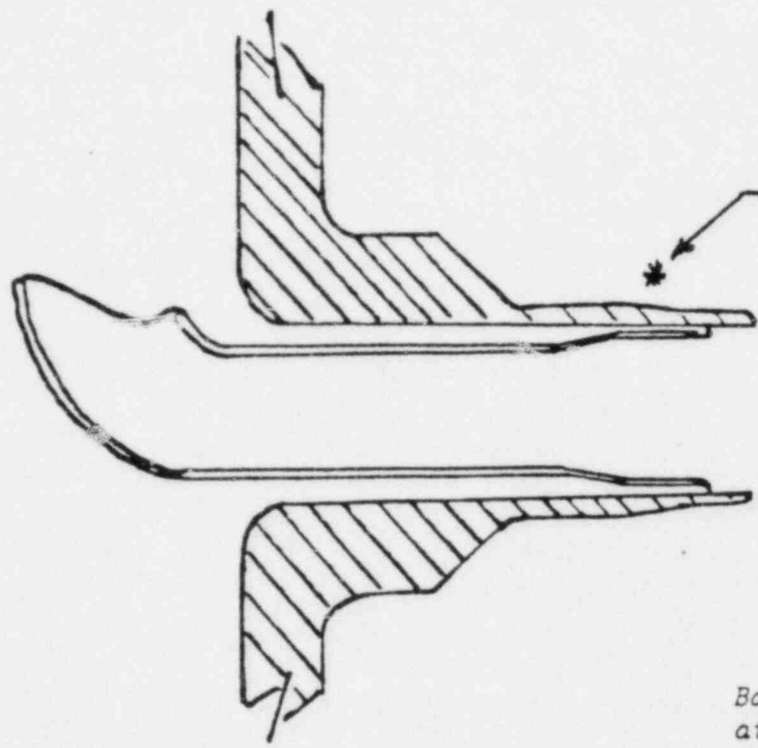
ADDITIONAL INTERNAL SHIELDING

Shielded platform will be placed to reduce shine from RPV when drained. Dose rate through the platform in drained condition is estimated to be less than two R/hr. The RPV head will be set in place with ventialtion suction taken from RPV air space exhausting through filtration units. The 6 inch thick RPV head will provide the final shielding for the above vessel area.



Temporary shield collars are being considered for the jet pump slip joints.

FIGURE 9



Based on actual experience at Monticello in November 1982.

BEFORE WATER LANCING	AFTER WATER LANCING
25 TO 35 R/HR.	2 TO 3.5 R/HR.

DECONTAMINATION
OF
NOZZLE TO THERMAL SLEEVE ANNULUS

FIGURE 10

MOCK UP

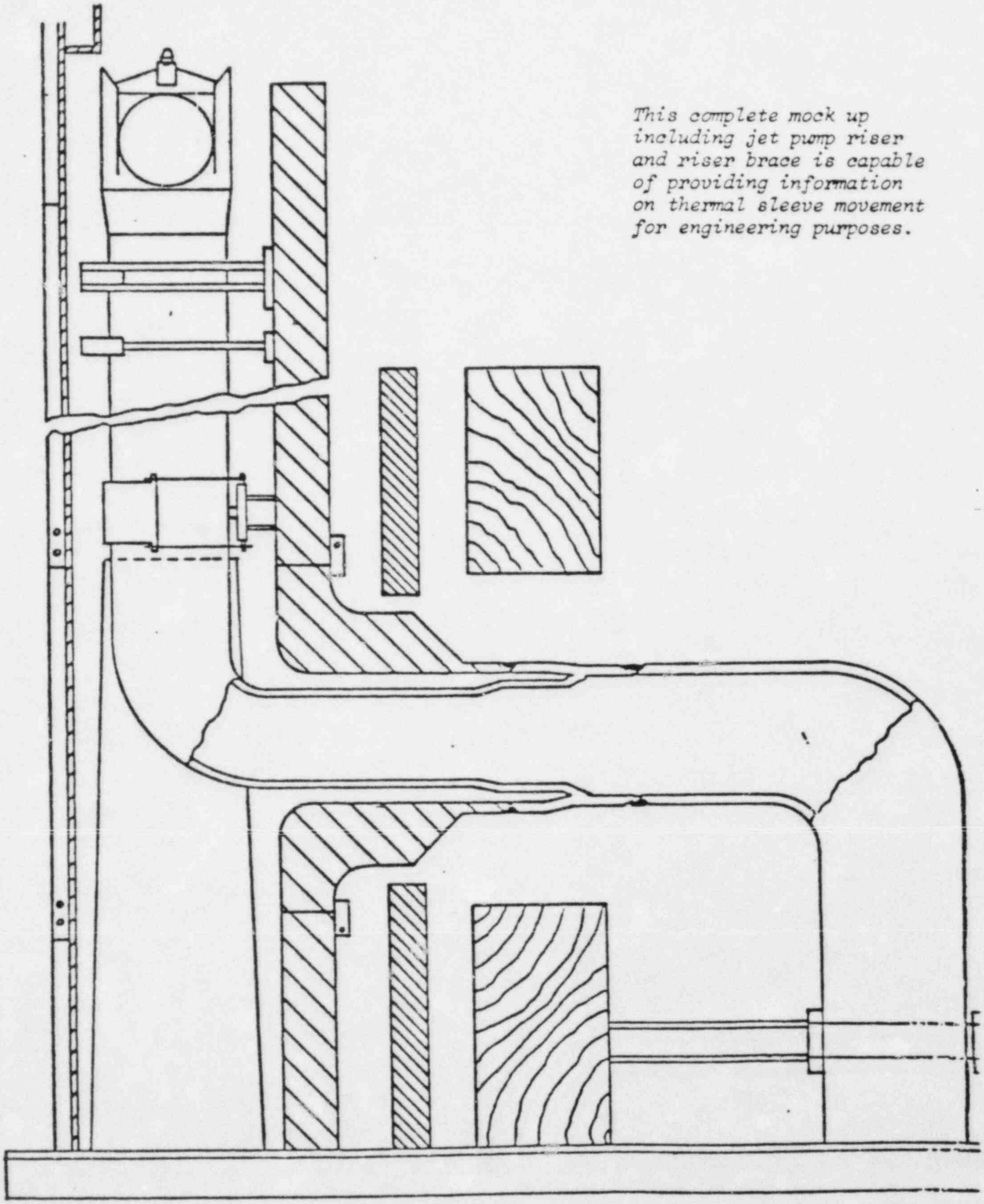


FIGURE 11

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APPENDIX A

DECONTAMINATION AND WASTE
DISPOSAL FOR
MONTICELLO RECIRCULATION PIPING
REPLACEMENT LICENSING REPORT

Prepared for:

NORTHERN STATES POWER COMPANY
Minneapolis, Minnesota

Prepared by:

QUADREX CORPORATION
1700 Dell Avenue
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July 1983

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APPENDIX A

RECIRCULATION SYSTEM LOOP DECONTAMINATION AND WASTE DISPOSAL

A1.0 DECONTAMINATION CAMPAIGN OVERVIEW

A1.1 General

In order to significantly reduce the overall occupational radiation exposure (ORE) to the workers involved in the recirculation piping replacement, NSP has contracted Quadrex Corporation to decontaminate both loops of the nuclear boiler recirculation system from near the reactor vessel suction nozzle to near the external riser vessel nozzles. In addition, RHR lines REW10-18" and TW30-16" to isolation valves MO18-N14DU and MO16-NW123D for loop A and similar lines and valves for loop B, and the recirculation loop equalizer lines will be decontaminated. Recirculation loop valves MO2-43A, MO2-53A, MO2-43B, and MO2-53B, and the recirculation pumps will be treated directly through available drain lines to obtain more effective decontamination of the multiple interior surfaces of these components.

A1.2 Radioactivity To Be Encountered at Monticello During the February 1984 Outage

Out-of-core activity deposited in BWR coolant circuits consists of several nuclides, the most important of which is cobalt 60, with a 5.2-year half-life. This isotope, produced through the activation of ordinary cobalt and released from reactor component alloys, is responsible for 80 percent of the radiation received by reactor personnel during routine maintenance tasks. Activity buildup takes place on the pipe walls through the process of corrosion film buildup, which incorporates activated isotopes captured from the coolant stream.

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Successful decontamination processes involve the removal of these oxides from the pipe walls without imparting harmful effects to the structural alloy substrate.

Radionuclide activity in the Monticello BWR plant was studied in 1974 and reported in NEDC-13361-03, tables 2.4 and 2.5. On the basis of these and data from other plants, an analytical model was developed to predict out-of-core activity accumulation in BWRs based upon cobalt 60. Figure 3-B in NEDC-21550, issued in June 1977, compares model predictions with one experimental point obtained from Nine Mile Point BWR. In the absence of specific data from Monticello, Rad-11 computer code predictions were read directly from this report, taking into account total effective full power hours (EFPH) that Monticello will have accumulated by February 1, 1984. The estimated EFPH at the next outage will be 84,686. Using this estimated EFPH, Rad-11 code gives a reading of 40 $\mu Ci/cm^2$ for a projected activity reading when the next plant outage occurs.

Prior to decontamination activities a three-week period is scheduled which will allow testing to determine the actual oxide thickness, character, and radioactivity accumulation. Flanges now attached to the recirculation piping can be removed and deposits on them tested within the three-week period from plant shutdown to decontamination initiation.

Excess chemicals and ion exchange cleanup resins will be on hand at the site to cover possible situations where the existing radioactivity burden is, in fact, greater than that estimated.

A1.3 Oxide Characteristics

Information available which is specific to Monticello was found in NEDC-13361-03 and NEDC-21550. Analysis of decontamination flanges will permit a more precise assessment of the existing oxide burden. As in the case of surface radioactivity, excess chemical reagents will be on

hand to respond to information which is more specific to the plant. The assumed character of the deposited oxides is shown in table A-1.

It is anticipated that the decontamination solution can be applied directly to Monticello, without a preoxidation step. This assumption is based upon the nature of oxides assumed to be present in this plant. It is possible, however, that the oxide analysis of the decontamination flange may show high chromium content and/or chromium with unexpected stoichiometry. Either of these findings may require the addition of an osidizing step prior to the decontamination solution. The oxidation will be accomplished by using dilute solutions of $KM\mu 0_4$ and $NH0_3$ for approximately 24 hours.

A1.4 Flow Paths

The specific flowpath for decontamination is still under consideration by utility personnel. It is probable that solutions will be restricted to the recirculation pump suction and discharge lines, with no introduction of chemicals to the reactor annulus or vessel proper. In order to achieve the agitation necessary to enhance mass transfer of chemicals to the pipe surfaces, alternate fill and drain between the two recirculation loops will probably be utilized. Liquid flow wil be limited to the piping system by using a controlled volume of solution, plugs on the pipe ends, and level instrumentation.

If it is decided that the benefits of potentially higher decontamination factors (DFs) warrant the use of a recirculating flowpath, one of several alternatives could be implemented. One alternative would consist of recirculating decontamination solution through the reactor vessel annulus by using the recirculation pumps at their lowest speed control (approximately 24 percent of design flow rate). Another alternative would involve decontaminating the recirculation pump suction and discharge lines separately. For that method, temporary lines would be

TABLE A-1. Elemental Concentration of Deposits on BWR Out-of-Core Surfaces*

Oxide Layer	Fe	Cr	Ni	Co	Total
	(mg/dm^2 of Metal Per Oxide Layer)				
NMP Outer	48	1.1	1.3	0.05	50
NMP Inner					
Species 1	30	0.4	5.4	0.17	36
Species 2	9	1.2	2.9	0.08	13
QC1 Outer	11	0.6	0.6	0.03	12
QC1 Inner					
Species 1	49	0.7	5.8	0.24	56
Species 2	12	3.1	2.3	0.09	17

Oxide Layer	Fe	Cr	Ni	Co
	(Percent of Oxide Layer)			
NMP Outer	95	2	3	0.1
NMP Inner				
Species 1	83	1	15	0.5
Species 2	70	9	20	0.6
QC1 Outer	90	5	5	0.2
QC1 Inner				
Species 1	88	1	10	0.4
Species 2	68	18	14	0.5

*Source: EPRI report NP-522.

installed off the discharge and suction piping. Decontamination solution would then be recirculated through either pipe section between the permanent decontamination connection and the new temporary line.

It is anticipated that the branch RHR pipe from the recirculation loops will be decontaminated by flowing solution between the recirculation loops and vent or drain lines adjacent to solution valves M018-N14D and M016-NW123D on one loop and similar valves on the other loop.

A1.5 Estimated Time

The time required for decontamination will be dependent on the flowpath selected. Estimates for the alternatives discussed in section A1.4 range from 7 to 16 days, including critical path time required for equipment set up, decontamination, clean up, and equipment dismantling.

A1.6 Expected Results

Access to the system and circulation of chemicals throughout all of its parts will determine the degree of success. A target overall decontamination factor of 10 for stainless pipe surfaces has been established and has been demonstrated to be achievable in a previous project performed by Quadrex. Nitrogen gas injections will be used to periodically provide mixing through the system. Separate, localized injections of decontamination chemicals will be made at all convenient entries into the plant loop (especially valves) to enhance accessibility of chemicals to all plant system surfaces.

A1.7 Corrosion Surveillance

Surveillance specimens will be placed into the chemical stream, to provide verification that no corrosion damage was imparted during the decontamination campaign. Selection of alloys, geometry, and heat treatment conditions will be those considered most prototypical of plant materials.

Corrosometer probes will be placed into the chemical stream to provide on-line corrosion rate information. Thus, an independent monitoring method will be available for information regarding the plant piping.

A1.8 Documentation

A description of all processes to be used during the decontamination will be written in advance for review by the NSP appropriate organizations. Readiness reviews will be conducted prior to start of the decontamination operations. Schedules and safety will be maintained within the allowable limits agreed to in advance.

Safety issues, including radiation exposure, associated with the chemical process will be addressed as a part of the operating procedures.

A final report will be prepared, summarizing the findings, data, and results obtained during the decontamination projects.

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A2.0 DECONTAMINATION PROCESS TO BE USED

A2.1 Process Description

Quadrex Corporation will use the low oxidation-state metal ion (LOMI) dilute chemical process for the Monticello recirculation system decontamination. This process was developed in the United Kingdom by the Central Electric Generating Board (CEGB), and is ideally suited for BWR oxides, as its active reagent, vanadus formate, reacts stoichiometrically with ferric ions of the deposited oxide to reduce these to a ferrous state. This reaction takes place quickly destabilizing the oxide structure and allowing the chelant, picolinic acid, to tie up the ferrous ion and thus keep it in solution. No solid particulate by-products are involved. Once the oxides are dissolved, cobalt and other radioactive species are also complexed by the chelant and thus remain in solution. Working temperature for the LOMI process is 90°C.

Because the process involves dilute chemicals (typical concentrations are vanadus formate $4x10^{-3}$ molar, picolinic acid $2.5x10^{-2}$m), the liquids can be processed through anion or cation resin beds, and the activated cations removed. The general chemical reactions of interest have been summarized in figure A-1.

Quadrex Corporation is licensed by the CEGB to use the LOMI process.

The LOMI process is known to the NRC from two previous activities:

- o The process was used to decontaminate the Surry steam generator at Battelle PNL. This is part of an investigation program under funding and technical direction of the NRC. Report is now being prepared by Battelle (PNL), Robert Clark project manager.
- o A detailed presentation was made to the NRC in the fall of 1980, by UK nationals describing the LOMI process in detail.

Dilute Chemicals Decontamination

1 One Electron Transfer Reduction

$M^{n+} + e^- \xrightarrow{\text{Reduce}} M^{(n-1)+}$

Fe^{3+} oxide + $V^{2+} \longrightarrow Fe^{2+}$ oxide + V^{3+}

2 Fe^{2+} oxide $\longrightarrow Fe^{2+}$ solution

Fe^{2+} solution + Picolinic Acid $\longrightarrow$ [Fe Pc] Complex

3 Radionuclide Removal

$(M\ Lx)^{n+} + nRs^- \rightleftharpoons (Rs)_n - M + xL$

L = Chelant Molecule (Picolinic)
M = Metal Ions
Rs = Ion Exchange Resin

Need to Favor Equilibrium to the right
Holds Metal Ion in Rs
Releases Chelant for Reuse

4 When Chromium is Present in Cr^{3+} in PWR Oxides

$Cr_2O_3 + 2\ MnO_4^- + 2OH^- \longrightarrow 2\ CrO_4^{2-} + 2\ MnO_2 + H_2O$

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FIGURE A-1

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A2.2 Corrosion Considerations

As figure A-2 illustrates, the LOMI process is oxide-specific and avoids substrate metal removal characteristic of other decontamination processes.

General corrosion rates for alloys of interest are shown in table A-2 from studies performed in the UK. They indicate very low initial loss. No pressure boundary damage is anticipated from this type of corrosion. The corrosion rates of surveillance coupons in the solution were found to be much less than 0.5 mils for stainless steel and Inconel 600 for the entire decontamination, which consisted of two cycles. Corrosometer probes, which measure rates of corrosion during the presence of chemicals in the component being decontaminated, indicated corrosion rates of 0.03 mils for 304 stainless steel, 0.1 mil for Inconel 600, and 1.8 mils for carbon steel. Examination of U-bend (stressed) specimens under 16X magnification revealed no stress or crevice corrosion. Dye penetrant was employed for increased sensitivity in detecting any surface imperfections on U-bend specimens, but none were found. Several double Ubend specimens were exposed in the decontamination solutions, to study the behavior of alloys when a crevice is present. The double Ubend specimens were separated for examination, again using liquid penetrant to enhance sensitity in finding surface cracks. None were found.

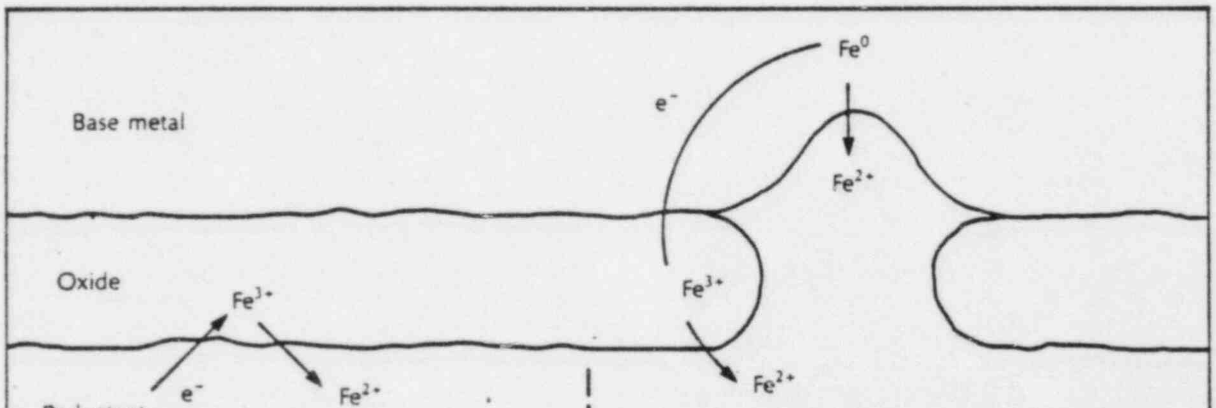
There has been a concern for the effect of traces of chemicals that may escape during the rinsing operation.* Tests at UK were performed with type 304 creviced specimens, which after LOMI processing were autoclaved in BWR conditions for 100 hours. No crevice, stress, or integranular attack was found.

---

*The predominant chemicals in LOMI are formic acid or picolinic acid in in concentrations of approximately $4x10^{3}$ molar. Most likely the residues that may remain in reactor components will decompose upon reactor startup to form CO_2, NH_3 or H_20.

Dilute Chemicals Decontamination

LOMI Chemical Reactions
Comparison of Oxide Dissolution with Acid Dissolution



REDUCTIVE DISSOLUTION

Fe^{3+} oxide + reductant $\longrightarrow$ Fe^{2+} oxide

Fe^{2+} oxide $\longrightarrow$ Fe^{2+} solution

ACID DISSOLUTION

Fe^0 metal $\xrightarrow{\text{corrosion}}$ Fe^{2+} solution + $2e^-$

Fe^{3+} oxide + e^- $\longrightarrow$ Fe^{2+} oxide

Fe^{2+} oxide $\longrightarrow$ Fe^{2+} solution

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FIGURE A-2

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TABLE A-2. General Corrosion Rates of Alloys in LOMI Chemicals*

Alloy	Cycle I Preautoclaved for 28 Days		Cycle II Preautoclaved for 28 Days		Nonautoclaved	
	Mg/cm^2	Mils/Day	Mg/cm^2	Mils/Day	Mg/cm^2	Mils/Day
347 Inconel 600 weld	0.48	0.037	0.45	0.035		
Incoloy 800/Inconel 600 weld	0.42	0.037	0.39	0.030		
Inconel 600 tubing	0.37	0.028	0.42	0.031	0.46	0.035
Incoloy 800 tubing	0.050	0.004	0.061	0.005		
Incoloy 800, grit blasted	0.063	0.005	0.065	0.005		
304 stainless steel	0.082	0.006	0.14	0.011	0.073	0.006
Stellite 6	0.54	0.041	0.57	0.043		
Zircalloy 2	0.006	0.0006	0	0		
410 stainless steel					0.32	0.026
321/321 stainless steel weld					0.053	0.004

*Rates based on weight loss, data from three samples for each measurement shown.

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The experiments performed by CEGB and shown in table A-2 indicate that neither crevice, nor stress, nor pitting occurred in LOMI solutions for types 304 or 316 stainless steel, or Stellite 6 (valve seat material).

A comprehensive program is now in progress, designed to yield additional corrosion information for the alloys of interest.

A2.3 Previous Experience with LOMI Decontamination

A2.3.1 Winfrith Reactor Decontamination

Primary circuits of the 100 MWe Winfrith reactor in UK, a BWR, were decontaminated in 1980 and 1981 with LOMI solutions identical to those to be used at Monticello. Materials used in Winfrith are predominantly type 321 stainless, type 410 stainless, and carbon steel; valve seats are Stellite 6B; and some components are made of type 304 stainless. The reactor has operated with no problems related to LOMI chemicals.

A2.3.2 Surry Steam Generator Decontamination Demonstration

The salient points, as shown in this demonstration, for application of the Quadrex process to reactor components are summarized below:

- o Decontamination factors as high as 60 for stainless and about 10 for Inconel 600 surfaces are obtained with only two applications of chemicals.
- o Application of electropolishing can increase DFs even higher, if needed, in specific areas (as demonstrated at Surry/Battelle following LOMI application).
- o Capture of radioactivity in ion exchange columns is compatible with LOMI, as demonstrated at Surry/Battelle.
- o The time involved is two to three days of critical path schedule or less (highly dependent on component involved).

- Corrosion of substrate metal is much less than $1x10^{-3}$ inches of stainless or Inconel 600.
- Chemicals used do not cause stress or crevice corrosion to alloys usually present in reactor systems.
- Continuous monitoring of the process is easily accomplished through simple, standard techniques.
- The mechanical/hydraulic system, designed with redundancy to provide reliability, functioned as required during the Surry/ Battelle job.
- Confinement, control of chemicals, and rinsing were demonstrated.
- No solid reaction products are generated in this process.

A2.3.3 Recent Corrosion Test Data

In addition to the corrosiion test data presented in table A-2, additional corrosion tests of LOMI-treated specimens have been completed to date.

- Metallographic examination of the surveillance specimens used in the Surry steam generator decontamination work using LOMI and POD* processes. Alloys include types 304, 316, and 347 stainless and welds, type 410 stainless, Inconel 600, and Inconel/347 welds. Samples include double Ubends (with crevices). This work was completed at CEGB. Results are summarized in table A-3.
- Metallographic examination of type 304, solution annealed, and sensitized. The samples were prepared by Carolina Power & Light, exposed to LOMI solutions in the United Kingdom, and returned to the CP&L for examination. Duplicate specimens were examined in the United Kingdom, also. Results are summarized in table A-4.

---

* POD = Pressurized Water Reactor Oxide Decontamination

TABLE A-3

SURRY SURVEILLANCE CORROSION COUPONS

Alloy	Geometry	Metal Loss* (mils)	Metallographic Examination Results (Range of penetration in microns)
304	Flat	<<0.1	Less than 2 microns, mostly smooth
347	Flat	<0.2	3 microns max, mostly smooth
410	Flat	0.5	3 microns, slight roughening
Inconel 600	Flat	<0.2	10 microns, isolated 10 micron pits
Inconel 600	Tubular	0 to <<1	Isolated, 16 micron pits on expanded surfaces
304	Double U-bend apex	-	2-12 no pitting at crevice surfaces
316	Double U-bend apex	-	4-8 no pitting at crevice surfaces
Inconel	Double U-bend apex	-	4-16 no pitting at crevice surfaces, some pitting outer apex, ~4 microns
304	U-bend straight portion	-	2-4, but mostly 2 microns
316	U-bend straight portion	-	2-8 no pitting
Inconel 600	U-bend straight portion	-	2-12 minor pitting
Inconel 600/347	Welds	<0.2	Smooth on finished surfaces Roughened (20 micron) on as welded surfaces
304	Flat preautoclaved**	<< .1	Less than 2 microns, mostly smooth
347	Flat preautoclaved	< .3	4 microns isolated, mostly smooth
Inconel 600	Flat preautoclaved	< .4	9 microns max, 12 microns in expanded portion

* Measured by micrometer, before/after exposure, total exposure 40 hours
** Preautoclaved, 21 days at 300°C, PWR water chemistry

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TABLE A-4

SUMMARY OF CORROSION DATA, CP&L TYPE 304 SPECIMENS

Chemical Treatment	Weight Losses			Metallographic Observations	
	Sample No*	Weight Loss (mg/cm²)	Oxide Loss** (microns)	Max Penetration (microns)	Appearance
AP/LOMI					
	A	1.7	3.5	4	Smooth, a few small pits
	1	2.2	4.4	4	Smooth, a few small pits
	E	1.5	3.1	4	Smooth, a few small pits
	5	3.1	6.2	4	Smooth, a few small pits
NP/LOMI					
	B	2.2	4.4	6	Smooth, some isolated pits
	2	3.3	6.6	<2	Smooth
	F	2.0	3.9	<2	Smooth
	6	3.7	7.4	<2	Smooth
LOMI					
	C	1.3	2.6	4	Smooth, some isolated pits
	3	2.3	4.6	<2	Smooth
	G	1.4	2.8	<2	Smooth
	7	2.5	5.1	<2	Smooth

* Numbered specimens 1 through 7 sensitized
Letter designate specimens A through G solution heat treated
** Oxide loss assumming density of 5 gm/cm³

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A2.3.4 Continuing Corrosion Evaluation Work

A comprehensive program, sponsored by EPRI (EPRI-RP-2296-4) is currently being carried out at Battelle Pacific Northwest Laboratories. The program includes type 304 and Iconel 600 alloys in the solution annealed, as well as in the sensitized conditions.

In this program, duplicate sets of specimens will undergo the following test sequence:

- o Exposure in LOMI solutions
- o Examination for general and localized corrosion, including metallography
- o Autoclaving in BWR and PWR conditions to ascertain that protective corrosion films are formed in post-decontamination situations
- o Testing under constant extension rate conditions in PWR/BWR environments to verify resistance to intergranular stress corrosion cracking mechanisms.

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A3.0 PROCESS SYSTEM DESIGN AND OPERATION

A3.1 System Description

Quadrex Corporation wll provide the process system that will be used to mix, heat, control the temperature, monitor the concentration of chemicals, inject solutions into the system, remove chemicals from the plant system, and rinse at the end of the campaign. Radioactive species will be removed by means of an ion exchange system connected to the process system.

The basic system will consist of a water storage tank of sufficient capacity to contain the volume of one recirculation loop (approximately 4,000 gallons); a surge tank (400 gallons) for mixing the LOMI solution; heat exchangers for cooling and for heating the LOMI solution; an ion exchange system to remove radioactive species; and pumps, valves, and interconnecting piping or hoses to circulate and inject the LOMI solution. Schematic details of the system are shown in figure A-3.

The typical processing steps are:

- o Drain recirculation piping and components
- o Purge plant system with nitrogen gas
- o Fill system, mix, and heat LOMI solution to 90°C
- o Deoxygenate LOMI solution by nitrogen gas purge
- o Confirm low levels of oxygen in solution
- o Inject LOMI and circulate
- o Inject additional LOMI through valve drain connections
- o Monitor LOMI reaction
- o Monitor activity removal
- o Monitor corrosion rates (Corrosometer)
- o Stop reaction drain and rinse
- o Continuous cleanup of activity through ion beds.

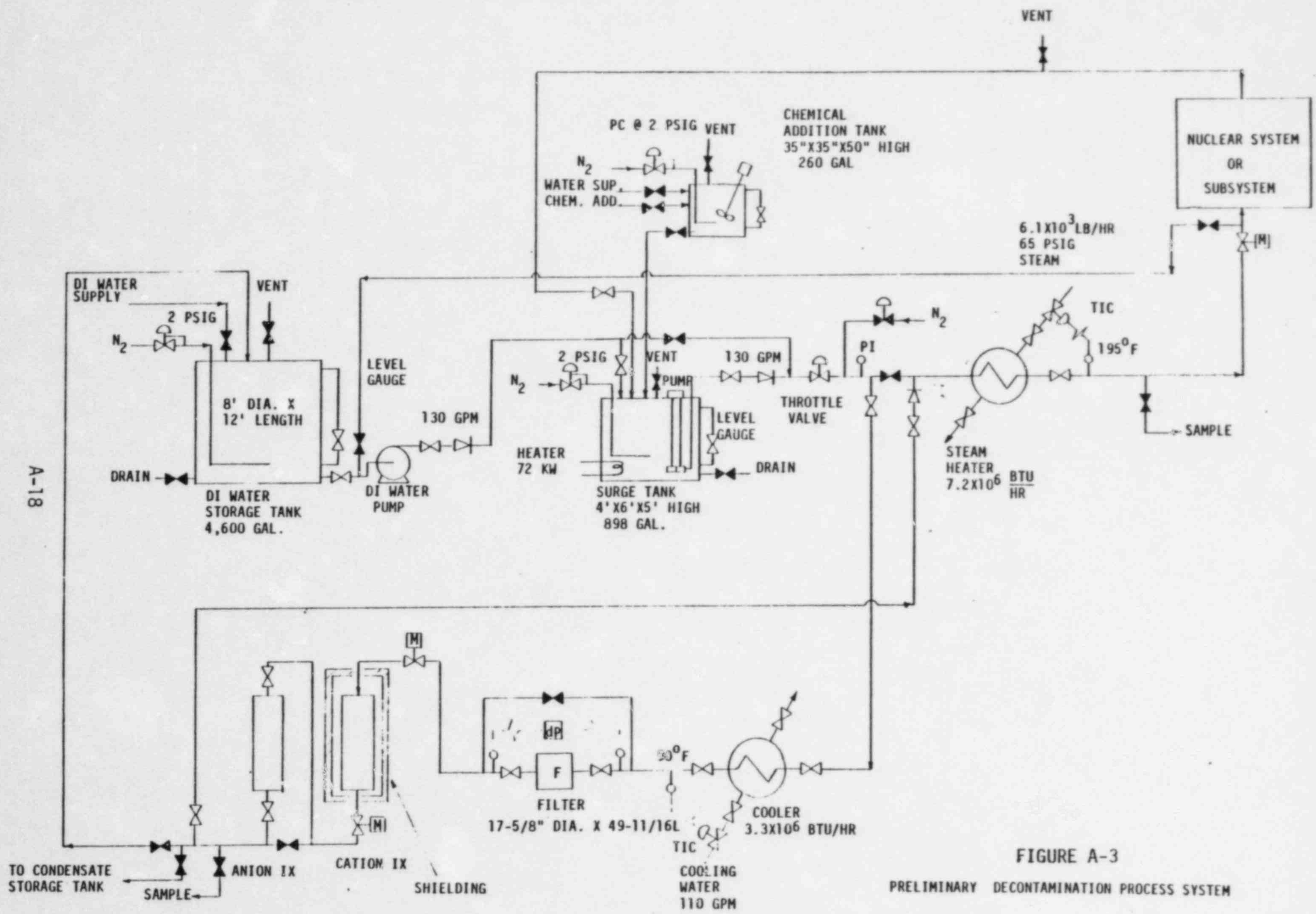

FIGURE A-3

PRELIMINARY DECONTAMINATION PROCESS SYSTEM

A3.2 LOMI and Chemical Supply

A portion of the LOMI chemicals will be formulated in the United Kingdom and shipped to Monticello. The remaining chemicals will be purchased in the United States. Specifications and procurement of chemicals will be provided by Quadrex Corporation. Chemical analysis for the reagents will be made available upon request.

A3.3 Monitoring and Sampling Equipment

Chemistry control will be performed by Quadrex, using United Kingdom personnel from the Central Electricity Generating Board (CEGB) and Babcock & Woodall-Duckham, to monitor and adjust the critical parts of the process. The United Kingdom personnel are those from the original team which developed the LOMI chemical process at the CEGB Berkeley Laboratories.

Continuous sampling of the liquid inventory will be performed at all times as the reactor loop to be cleaned is filled. Teflon sampling lines will be provided to transport a stream of liquid to a convenient space where a temporary field laboratory will be set up. Discarded liquids will be either collected into temporary storage containers or channeled into the plant radwaste system.

Process monitoring will encompass the following:

- o Atomic absorption (AA) for cations of interest to control the process, such as Fe, Cu, Ni, plus some others of lesser importance to be specified at a later date.
- o pH and conductivity, instruments to be provided by NSP.
- o Ultraviolet spectrophotometer for V^{++} and manganese analysis, if an oxidation step is necessary.
- o Colorimeter for continuous analysis of V^{++} when this ion is injected into the stream.

- Corrosometer will be provided to measure on-line corrosion rates of representative alloys.
- Nitrate-specific ion electrode (if needed) and oxygen analysis kits will be provided. A portable oxygen analyzer is available for use at the plant site.

A4.0 WASTE DISPOSAL

A4.1 Resin Disposal

Quadrex Corporation will be responsible for all resin disposal, including the activity which was removed from the plant system and collected by the ion exchange columns.

Activity removed from the plant components will be concentrated into ion exchange resins. The resin disposal operation will include:

- Equipment set up at the site
- Operations for the system during the decontamination activities
- Demineralizer vessels and resins plus all connections, miscelaneous fixtures needed to add, move, or mix resins
- Shipping casks and shipment of wastes.

A4.2 Recirculation Pipe Disposal

Quadrex will provide an onsite crew of waste handling specialists to prepare the removed pipe sections either for shipment to a fixed base decontamination facility for decontamination or for shipment to a burial site if the pipe sections are found to be activated or contaminated in a manner that makes decontamination unreasonable.