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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Supplement to License Amendment Request for Measurement Uncertainty Recapture – Power Uprate – Response to Requests for Additional Information (TAC Nos. ME3015 and ME3016)

References:

1. Letter from Northern States Power Company, a Minnesota corporation, to the Nuclear Regulatory Commission, "License Amendment Request for Measurement Uncertainty Recapture – Power Uprate," L-PI-09-133, dated December 28, 2009, ADAMS Accession Number ML093650045.

2. Letter from T. Beltz (NRC) to M. Schimmel (NSPM), "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Requests for Additional Information (RAI) Associated With License Amendment Request Re: Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME3015 and ME3016)," dated March 17, 2010, ADAMS Accession Number ML100740039.

In Reference 1, Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, submitted a License Amendment Request (LAR) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, to increase the licensed thermal power as a result of a measurement uncertainty recapture (MUR) power uprate. In Reference 2, the Nuclear Regulatory Commission (NRC) Staff requested additional information to support review of Reference 1. Enclosure 1 to this letter provides the responses to the NRC Staff requests for additional information. NSPM submits this supplement in accordance with the provisions of 10 CFR 50.90.

The supplemental information provided in this letter does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the Reference 1 submittal.

In accordance with 10 CFR 50.91, NSPM is notifying the State of Minnesota of this LAR supplement by transmitting a copy of this letter to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Sam Chesnutt at 612-267-7546.

Summary of Commitments

This letter contains the following new commitment:

- NSPM will submit justification for the Main Steam system stress analysis, including piping, supports, and components by August 27, 2010.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on APR 19 2010

Mark A. Schimmel
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, PINGP, USNRC
Resident Inspector, PINGP, USNRC
State of Minnesota

ENCLOSURE 1

Response to NRC Requests for Additional Information, dated March 17, 2010, Regarding a License Amendment Request for a Measurement Uncertainty Recapture Power Uprate Project at the Prairie Island Nuclear Generating Plant

This enclosure includes responses from the Northern States Power Company, a Minnesota corporation (NSPM), to Requests for Additional Information (RAI) provided by the Nuclear Regulatory Commission (NRC) in a letter dated March 17, 2010 (ADAMS accession number ML100740039). These RAI responses are provided in support of NSPM's License Amendment Request (LAR) for a measurement uncertainty recapture (MUR) power uprate submitted December 28, 2009 (ADAMS accession number ML093650045).

Throughout this Enclosure, the MUR power uprate LAR is referred to as "the LAR," and the NRC RAI as "the RAI." This Enclosure quotes each RAI question in italics and each question is followed by the NSPM response.

1. Fire Protection Branch (AFPB) RAI 1:

The staff notes that Enclosure 2 to the LAR, Section II.2.26, "10.3.1 – Plant Fire Protection Program (Appendix R)," states that "...Evaluations conclude that the current safe shutdown analyses use an analytical core power of 1683 MWt (or higher) and, as such, the MUR Power Uprate has no effect on the plant equipment and systems credited with achieving safe shutdown. Likewise, evaluations also conclude that MUR PU has no impact on Appendix R manual action constraints..."

The staff requests the licensee to verify that the (1) measurement uncertainty recapture power uprate will not require any new operator actions, and (2) any effects from additional heat in the plant environment from the increased power will not interfere with existing operator manual actions being performed at their designated time and place.

NSPM Response:

The Appendix R Fire Protection Program evaluation addressed the safe shutdown analysis and verified, as noted in the RAI, that the current safe shutdown analyses use an analytical core power of 1683 megawatts-thermal (MWt) or higher, and the MUR power uprate has no effect on plant systems and equipment relied upon to achieve safe shutdown conditions. This evaluation considered the impacts of the MUR power uprate on system temperatures, operating environments, and operator actions.

(1) Operator Actions
NSPM has reviewed the Prairie Island Nuclear Generating Plant (PINGP) Appendix R Fire Protection Program Safe Shutdown analysis and verified that no new or additional operator actions are required for the MUR power uprate.

The Appendix R safe shutdown analysis is included in Operations Manual F5, Appendix E, "Fire Protection Safe Shutdown Analysis Summary," as noted in the LAR, Enclosure 2, Section II.2.26 (page 41). Appendix E to Operations Manual F5 describes the safe shutdown methodology and functions, and identifies safe shutdown systems, equipment, and certain manual operator actions. The operator actions identified in Appendix E do not change as a result of the MUR power uprate and no new operator actions are added.

Operator response times were also reviewed and the existing analyses were determined to bound the MUR power uprate. The analysis that evaluates plant response, system time constraints for Safe Shutdown, and the feasibility of performing credited post-fire operator manual actions assumed a 2.0% uncertainty of the initial power level (102% * 1650 MWt = 1683 MWt) and therefore bounds operation up to the core power level of 1677 MWt, when a 0.36% measurement uncertainty is taken into consideration (100.36% * 1677 MWt = 1683 MWt), as proposed for the MUR power uprate.

The discussion regarding the safety related cooling water system design in Section VI.1.C, in Enclosure 2 (page 87) of the LAR, also addressed safe shutdown cooling, including time limitations for facility cooldown. This discussion includes Appendix R timeliness requirements for cooling the reactor to cold shutdown conditions within 72 hours, and the PINGP evaluation and assumptions are not affected by the MUR power uprate. No changes are required to safe shutdown equipment or operator actions.

In summary, the PINGP Appendix R Fire Protection Program analyses and procedures are based on achieving safe shutdown from analyzed core power levels of 1683 MWt, and bound the MUR power uprate conditions. The current operator actions in existing plant procedures associated with an Appendix R condition will remain applicable for operation up to an uprated core power of 1677 MWt. Operator actions included in the current program are not changed and no new operator actions are required.

<u>(2) Impact of MUR power uprate on habitability of areas requiring operator actions</u>
To achieve and maintain safe shutdown conditions, the initial actions are to shut down the reactor, which removes the major heat source, leaving decay heat. Actions after shutdown will involve areas with systems that are generally cooling down below normal operating temperatures, with the exception of residual heat removal and associated systems to remove decay heat.

Following reactor trip, the power level quickly falls to decay heat levels, as stated in the LAR, Enclosure 2, page 32, Section II.2.11. The Appendix R cooldown analysis was performed at 1683 MWt (102% of 1650 MWt) and addressed decay heat resulting from operation at this power level.

In the Auxiliary Building and Turbine Building areas where operator actions may be required, a fire may incapacitate the heating, ventilating, and air conditioning (HVAC) systems. Loss of HVAC analyses use estimated heat emission from pipes and other inputs to room heatup calculations. After reactor trip, the system temperatures and motor heat inputs will generally fall, with the exception of residual heat removal and associated equipment. RHR heat inputs have been evaluated and the room heatup calculations are not affected by the MUR power uprate.

For any actions that may be performed prior to shutting down the reactor, certain areas of the facility will be slightly warmer after the MUR power uprate than at current power levels, as described in Environmental Qualification of Electrical Equipment discussion in the LAR, Enclosure 2, Section V.1.C, page 77. However, these temperature differences are not significant and will not affect operator access. The MUR uprate results in a slight decrease in main steam operating temperature and a slight increase in feedwater operating temperature, 2.6 °F from 434.9 °F to 437.5 °F. Feedwater lines do not affect temperatures in the control room or auxiliary feedwater pump room, which are the most limiting for access. In addition to this increase in fluid heat loads, the MUR power uprate also results in a small increase in motor heat loads in the Auxiliary and Turbine Buildings. These motors would not likely be running after a reactor trip as noted above. These changes have been evaluated to have a negligible effect on environmental temperatures.

Areas where operator actions may be required to achieve safe shutdown conditions are recognized to have potential habitability concerns, irrespective of the MUR power uprate. The current fire response program described in Operations Manual F5, Appendices D and E, includes precautionary notes that some actions are to be performed "if accessible" and that some actions may require use of protective measures in accordance with NSPM procedures on heat stress guidelines. Heat stress control measures may include ice vests, stay times, cooldown times, and administrative controls to limit worker risk. These worker protections and the need to use these procedures are not changed by the MUR program.

In summary, manual operator actions to achieve and maintain Appendix R safe shutdown conditions would be expected to be performed after a reactor trip, and would occur in operating environments that are not significantly impacted by the MUR power uprate. Appendix R safe shutdown analyses were performed for 1683 MWt and bound the MUR power uprate. The only change in area temperatures is due to a slight increase in feedwater temperature which is not significant and would not interfere with existing operator manual actions being performed at their designated time and place.

2. *AFPB RAI 2:*

Some plants credit aspects of their fire protection system for other than fire protection activities, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems.

If the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, credits its fire protection system for other than fire protection activities, the MUR-PU LAR should identify the specific situations and discuss to what extent, if any, the MUR-PU affects these "non-fire-protection" aspects of the plant fire protection system.

If the PINGP, Units 1 and 2, do not take such credit, the staff requests that the licensee verify this as well.

NSPM Response:

The PINGP Updated Safety Analysis Report (USAR) identifies that the fire protection system can supply makeup water to the spent fuel pool, in addition to fire protection activities. There are two fire hose stations near the spent fuel pool each rated for 95 gpm that are identified in USAR Section 10.2.2, "Spent Fuel Pool Cooling System, Performance Analysis," as being one of six water sources that are available in the unlikely event that all spent fuel pool cooling is lost and boiling occurs. The spent fuel pool cooling requirements have been evaluated for removal of decay heat associated with 102% of the current operating power (102% * 1650 MWt = 1683 MWt). Since the Spent Fuel Pool Cooling System analyses already assume conditions that bound the MUR power uprate conditions, this function is not affected by the MUR power uprate.

The PINGP USAR accident analyses in Chapter 14 do not take credit for the fire protection system for functions other than fire protection.

In emergency conditions that are beyond the design basis of the plant, PINGP damage mitigation procedure EDMG-2, "Guideline for Damage Mitigation Strategies," and Severe Accident Management Guidelines 1SAG-1, "Inject into the Steam Generators," and 1SAG-3, "Inject into the RCS," describe provisions to use the fire protection system for the following functions:

- Provide makeup to the spent fuel pool, as stated above
- Inject water into the steam generators, using portable equipment, hose adaptors, and specified connection locations
- Inject water into the reactor coolant system (RCS), using a specially constructed connecting piece and a specified flush connection

The capabilities to inject fire protection water into the steam generators and RCS are not considered design basis functions and are not affected by the MUR power uprate.

3. Steam Generator Tube Integrity and Chemical Engineering Branch (CSGB) RAI 1:

Protective coating systems provide a means for protecting the surfaces of facilities and equipment from corrosion and radionuclide contamination. Additionally, coatings used inside containment should be suitable and stable under design-basis loss-of-coolant accident conditions. It is unclear to the staff whether the protective coatings used inside containment were evaluated under power uprate conditions.

> *CSGB RAI 1: Please discuss whether the protective coating systems used in the containment remain qualified under MUR-PU conditions. Specifically, please discuss whether the temperature, pressure, and irradiation of the coatings during the qualification testing bounds the anticipated MUR-PU conditions.*

NSPM Response:

PINGP maintains a protective coatings program for interior surfaces and permanently installed equipment in the reactor building, as described in procedure H35, "Safety Related Coatings." Unqualified coatings could become a debris source after an accident and could contribute to blockage of sump strainers that could affect post-LOCA recirculation flow.

The Safety Related Coatings program requires that new coating systems for areas inside containment be qualified and tested in accordance with American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." Existing coatings that have not been determined qualified are periodically inspected. The quantities of unqualified and degraded coatings are tracked to ensure that the total quantity of these coatings is maintained below that assumed in the sump performance analysis, as described in PINGP USAR, Section 6.2.2, "Safety Injection System, System Design and Operations."

The requirements for new coating systems described in procedure H35 specify qualification testing to the following pressure, temperature, and radiation levels:

- Temperature and pressure values in accordance with the post-Loss of Coolant Accident (LOCA) temperature and pressure profile contained in PINGP calculations; the post-LOCA profile includes a peak containment temperature of 274.4°F and a peak pressure of 45.5 psig.
- Radiation exposure of 2E8 Rads.

The current temperature, pressure, and radiation exposure requirements in the Safety Related Coatings procedure H35 bound the anticipated MUR power uprate conditions as follows.

Temperature and pressure requirements
The post-LOCA containment temperature and pressure conditions are derived from current LOCA analyses that are based on an analytical power level of 102% licensed thermal power, or an analyzed core power of 1683 MWt. The analyzed core power of 1683 MWt remains bounding for the MUR power uprate. The post-LOCA containment pressure and temperature conditions have been evaluated as part of the containment integrity analysis, as discussed in Section IV.1.C, "Environmental Qualification of Electrical Equipment," in Enclosure 2 of the LAR (page 77). The temperature and pressure qualification requirements are not changed by the MUR power uprate.

Radiation qualification requirements
The current PINGP post-LOCA radiation qualification requirements for equipment or materials inside the containment structure are 2.0E8 rads (beta), 4.81E7 rads (gamma), total integrated dose at one year after the LOCA event, as identified in PINGP Environmental Specification, H8-H. Specification H8-H states that these doses include the additional radiation levels expected after the MUR power uprate. The Environmental Qualification discussion in the MUR power uprate LAR, Section IV.1.C in Enclosure 2, states that radiation doses (normal + accident) will increase by approximately 1.64% after implementation of the MUR power uprate. This increase has already been reflected in Environmental Specification H8-H.

The radiation qualification requirements in the Safety Related Coatings Program, procedure H35, of 2E8 Rad reflect the MUR power uprate post-LOCA environment and no changes are required.

Test data for coatings used at PINGP indicate qualification testing at a temperature-pressure profile that includes a peak temperature of 323°F, a peak pressure of 48.4 psig, and a radiation exposure of 3E8 Rad, which bounds the PINGP requirements noted above, and indicates that coatings are qualified for the MUR power uprate conditions.

4. *Vessels and Internals Integrity Branch (CVIB) RAI 1:*

Staff has reviewed the proposed request in accordance with Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" with respect to the integrity of the reactor pressure vessel (RPV), and reactor internal and core support structures.

The staff has reviewed the information submitted by the licensee, and based on this review, determined the following information is required to complete the evaluation.

CVIB RAI 1: It was noted that the licensee's RPV surveillance capsules withdraw schedule is based on American Society for Testing and Materials (ASTM) E-185-82.

Please provide the actual schedule as to when the licensee plans to withdraw any capsules in the future for PINGP, Units 1 and 2.

NSPM Response:

Each reactor vessel at PINGP initially included six surveillance capsules, of which four capsules have been removed from each Unit per an established withdrawal schedule. The last capsule to be removed from Unit 1 was Capsule 'S' at 18.12 Effective Full Power Years (EFPY) and the last capsule removed from Unit 2 was Capsule 'P' at 17.24 EFPY. Two surveillance capsules remain installed in each reactor vessel.

NSPM recently described plans for withdrawing a surveillance capsule from each Unit in a response to RAI questions regarding the PINGP License Renewal Application (NSPM letter dated November 12, 2008, L-PI-08-097, ADAMS Accession number ML083370202). One of the remaining capsules in each unit will be withdrawn in accordance with the requirements of ASTM E-185-82, Section 7.6.2, when its neutron fluence exposure exceeds a peak end of life (EOL) vessel fluence, but prior to exceeding twice that fluence exposure. Based on a projected EOL value of 54 EFPY associated with license renewal, the PINGP limiting fluence values are as follows:

- Unit 1: between 5.162E19 and 1.032E20 n/cm^2
- Unit 2: between 5.196E19 and 1.039E20 n/cm^2

NSPM's schedules for capsule removal for each Unit are as follows, along with the lead factor and projected neutron fluence values (n/cm^2) for each capsule at the time of the next withdrawal:

- Unit 1: refueling outage 1R27, expected in Spring 2011, 31.6 EFPY
  Withdraw: Capsule N: Lead Factor = 1.77; projected fluence = 5.893E19 n/cm^2
  Standby: Capsule T: Lead Factor = 1.89; projected fluence = 6.292E19 n/cm^2

- Unit 2: refueling outage 2R27, expected in 2012, 32.2 EFPY
  Withdraw: Capsule S: Lead Factor = 1.72; projected fluence = 5.739 E19 n/cm^2
  Standby: Capsule N: Lead Factor = 1.72; projected fluence = 5.739 E19 n/cm^2

A revised capsule withdrawal schedule consistent with this information was submitted for NRC approval in accordance with the requirements of 10 CFR 50, Appendix H, in NSPM letter L-PI-10-029, "Request for Revision to Reactor Vessel Material Surveillance Capsule Withdrawal Schedule for PINGP," dated March 30, 2010 (ADAMS Accession Number ML100900089).

NSPM has no current plans for removing the remaining surveillance capsules (one in each Unit) from the PINGP reactor vessels.

5. CVIB RAI 2:

Has the licensee pulled any surveillance capsules for either PINGP Units 1 or 2 since Capsules S and P were pulled and evaluated? If so, what were the capsules and the analysis report results?

NSPM Response:

No surveillance capsules have been pulled for either PINGP Unit 1 or Unit 2 since Capsules S and P were removed and evaluated.

6. CVIB RAI 3:

For RPV internals of PWR-designed light-water reactors may be susceptible to the following aging effects. Discuss how each is affected by the MUR:

1. *Cracking induced by stress corrosion cracking (SCC), or irradiation assisted stress corrosion cracking (IASCC);*
2. *Loss of fracture toughness properties induced by radiation exposure for all stainless steel grades, or the synergistic effects of radiation exposure and thermal aging for cast austenitic stainless steel grades;*
3. *Stress relaxation in bolted, fastened, keyed or pinned RPV internal components induced by irradiation exposure and/or exposure to elevated temperatures; and*
4. *Void swelling (induced by radiation exposure).*

NSPM Response:

The aging effects identified in the RAI are addressed at PINGP in a plant-specific pressurized water reactor (PWR) Vessel Internals Program. This program was developed as a new License Renewal Aging Management Program and information regarding this program was provided to the NRC in Xcel Energy letter L-PI-09-044, dated May 12, 2009, "Supplemental Information Regarding Application for Renewed Operating Licenses," (ADAMS Accession number ML091620163). This program was also reviewed in Section 3.0.3.3.2 of the NRC's Safety Evaluation Report Related to the License Renewal of PINGP Units 1 and 2, dated October 16, 2009 (ADAMS Accession number ML092890209).

The PINGP PWR Vessel Internals Program is based on the Electric Power Research Institute (EPRI) Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227) and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The MRP-227 guidelines consider various aging factors including neutron fluence exposure, temperature history, and representative stress levels for determining relative susceptibility of PWR internals to postulated aging mechanisms that include SCC, IASCC, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, irradiation-enhanced stress relaxation and creep, and void swelling.

<ins>Effects of MUR</ins>
Of the aging factors identified in MRP-227, the MUR power uprate results in a small fractional increase in neutron fluence exposure and stress levels. NSPM reviewed aging management program requirements for impacts of the MUR power uprate. This

review discussed changes in environmental conditions due to the MUR power uprate and concluded that Nuclear Steam Supply System (NSSS) and Balance of Plant design parameter changes (e.g., temperatures and flow rates) are nominal, and the Aging Management Reviews performed to support the License Renewal Application bound the expected environmental conditions following the MUR power uprate. The MUR will not introduce new aging effects or mechanisms. Implementation of the new PINGP PWR Vessel Internals Program will ensure that the effects of aging on the reactor vessel internals will be adequately managed for the license renewal term.

The PINGP PWR Vessel Internals Program and the aging effects identified in the RAI question are described below. No changes to this program are required as a result of the MUR power uprate.

Program Description
The program implements the inspection of the reactor vessel internals components through the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, as augmented by the examination requirements, including inspection methods, frequencies, acceptance criteria and sample sizes, established in the EPRI guidelines for Westinghouse designed PWRs.

The PWR Vessel Internals Program addresses the management of aging effects of the PINGP reactor vessel internals components, both non-bolted and bolted. The PINGP reactor vessel internals consist of two basic assemblies, the upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and the lower internals assembly that can be removed, if desired, following a complete core off-load. The scope of the program does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. The scope also does not include welded attachments to the reactor vessel.

As stated in the NSPM letter dated May 12, 2009, the scope of the PINGP PWR Vessel Internals Program includes the following items that address the issues identified in the RAI question:

- Managing crack initiation and growth due to irradiation-assisted stress corrosion cracking (IASCC), primary water stress corrosion cracking (PWSCC), stress corrosion cracking (SCC) and fatigue
- Reduction of fracture toughness due to radiation and thermal embrittlement and void swelling. Loss of fracture toughness due to radiation and thermal embrittlement is of consequence only if cracks exist and the local applied stress intensity exceeds the reduced fracture toughness. Cracking, if it occurs, is expected to initiate at the surface and is detectable by the augmented inspections performed under this program.
- Changes in dimensions due to void swelling
- Loss of preload due to stress relaxation
- Loss of material due to wear, in reactor vessel internals components.

The program is based upon the examination requirements for Westinghouse designed PWRs provided in MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008, along with the implementation guidance described in NEI-03-08, "Guideline for the Management of Materials Issues," May 2003. MRP-227 has been submitted to the NRC for review. Following NRC review and approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The PINGP PWR Vessel Internals Program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A.

The program will include a plant-specific inspection plan for vessel internals based on the guidance of MRP-227 (MRP-227-A when issued). This plan will include the requirements established by the MRP for Westinghouse designed PWRs, as approved or modified by the NRC, and will define any proposed alternatives determined to be necessary. This plan will adequately manage aging at MUR power uprate conditions.

The NSPM letter dated May 12, 2009 included NRC commitments as follows: NSPM will submit the plant-specific inspection plan, including any proposed alternatives, for NRC review and approval at least 24 months prior to entry into the period of extended operation. NSPM will implement the PWR Vessel Internals Program, including the inspection plan, prior to the period of extended operation. There is a potential for program implementation to occur sooner in accordance with an industry commitment to implement the Inspection and Evaluation Guidelines within 24 months following issuance of the NRC-approved MRP-227-A. These are not new commitments and are not changed by this letter.

7. *Electrical Engineering Branch (EEEB) RAI 1:*

Provide detailed discussion about the affects (if any) of the MUR-PU on the plant direct current systems.

NSPM Response:

The Leading Edge Flow Meter (LEFM) equipment that was added for the MUR project is not supplied with direct current (DC) power. The Prairie Island Units 1&2 onsite DC electrical power systems are not affected by the MUR power uprate because no loads were added and existing power supplies were not affected.

8. EEEB RAI 2:

Provide more detailed discussion of the uprated loadings for each of the main generators with respect to their nameplate ratings. Also, provide the nameplate ratings and the uprated loadings of the generator step up transformers, plant service transformers and the main generator isolated phase bus.

NSPM Response:

NSPM evaluated power block equipment at PINGP, including the main generator, generator step-up transformers, main auxiliary (plant service) transformers, and isolated phase bus duct, for MUR power uprate loads and found that operation will remain within original design loading as described further below.

1. Main Generator

The nameplate ratings for the main generators at PINGP are as follows:

- Unit 1: 659000 kilovolt amperes (KVA), 20000 volts, 19024 amps, 0.90 power factor (PF), 3 phase, 60 Hz, 1800 rpm, 4411 rotor amps, 500 volts exciter voltage, 60 psig H_2, 46°C cold gas, stator temperature 64°C, rotor temperature 64°C, maximum inlet water temperature 95°F
- Unit 2: 659000 KVA, 20000 volts, 19023 amps, 0.90 PF, 3 phase, 60 Hz, 1800 rpm, 4411 rotor amps, 500 volts exciter voltage, 60 psig H_2, 46°C cold gas, stator temperature 64°C, rotor temperature 64°C, maximum inlet water temperature 95°F

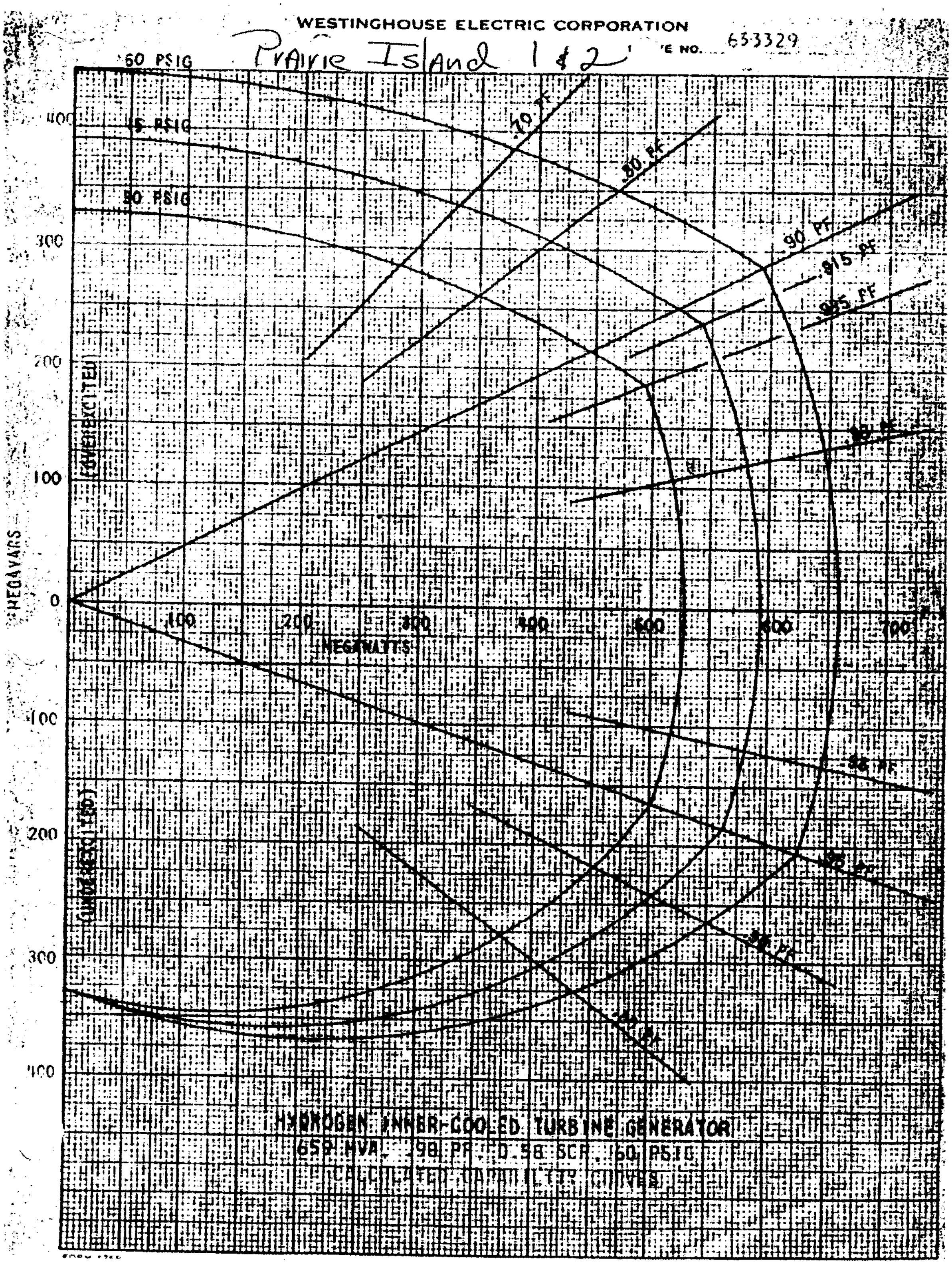
The generator capability curve for both Unit 1 and Unit 2 main generators is attached.

Evaluation
USAR Section 1.1.1 states that each reactor is capable of a gross electrical output of 575 MWe. The MUR power uprate will add approximately 9 MWe, depending on house load sharing, for a post-uprate gross electrical output of 584 MWe. This output is conservatively below the generator capabilities of approximately 592 MWe.

The main generators for PINGP Units 1 and 2 were evaluated for the following MUR power uprate conditions:

- Unit 1: 659000 kVA, 20 kV, 0.899 PF, 592917 kW at 60 psig, 46°C cold gas temperature
- Unit 2: 659000 kVA, 20 kV, 0.895 PF, 589546 kW at 60 psig, 46°C cold gas temperature

The main generator evaluation was performed using analytical and comparative techniques, based on a comparison against components of another generator of the same frame size. The evaluation included various subsystems and components of the generators. The evaluation concluded that the generator rotor shaft and rotor coil, generator stator core and coil, generator parallel rings, generator main lead and main lead bushings, generator flexible connectors, current transformers, generator hydrogen coolers, and exciter and automatic voltage regulator (AVR) for both Prairie Island Units 1&2 are capable of operating at MUR-PU conditions.



Generator Capacity Curves – Units 1 and 2

2. Generator Step Up Transformers

The generator step-up (GSU) transformer for each Prairie Island unit is a single, three-phase transformer with the following nameplate ratings:

600 MVA @ 55°C rise
672 MVA @ 65°C rise

For summer operations, the station will be able to meet Mid-West Independent System Operators (MISO) volts amperes reactive (VAR) and power factor requirements with GSU output below the 55°C rating for both leading and lagging operations.

In winter, generator real output is higher, and the GSU must operate slightly into the 65°C rating (<6 percent) for short periods of time to meet either leading or lagging power factor requirements from MISO. This condition of operating into the 65°C rating is considered to be acceptable since:

- The higher GSU rating at 65°C is intended for just such contingencies, short-term operations at higher demanded duties.
- Operation of the GSUs at output that exceeds the 55°C rating but falls below the 65°C rating is expected only infrequently. Based on historical data, this condition is estimated at less than 1 or 2 percent of GSU operating hours. Based on historical data, the GSUs are expected to exhibit margin and to operate below the 55°C rating for the great majority of the time.
- Limiting operation will occur in winter with station output at a maximum. Under these conditions, GSU forced air cooling has maximum capability (cold ambient temperatures) to maintain GSU winding temperatures within limits after the MUR power uprate.

3. Main Auxiliary Transformer

The plant service transformers at PINGP are designated as the main auxiliary transformers (1M and 2M) and these have nameplate ratings of:

20/4.16/4.16 kV, 24/32/40 MVA @ 55°C, 3-phase, 60 HZ and
26.9/35.3/44.8 MVA @ 65°C.

The transformers supply power to balance of plant (BOP) systems for each respective unit under normal operating conditions.

Changes to electrical loading under MUR power uprate conditions are limited to BOP systems which are most likely to be affected by the MUR power uprate. The BOP systems that are impacted by the MUR power uprate are the feedwater system, the condensate system, and the heater drains system. Analysis of these systems at the increased power level produced new pump operating points. The combined total brake horsepower (BHP) of the feedwater pumps, heater drains pumps, and condensate pumps results in a net increase in station electrical load from 11.646 to 11.741 MVA, or a change of approximately 0.095 MVA for these pumps. The small increase in house loads resulting from MUR power uprate is within the capacity of the main auxiliary transformers (1M and 2M) and is acceptable.

4. Isolated Phase Bus

The isolated phase bus duct transmits power from the generator terminals to the generator step-up (GSU) transformer. The bus is rated at 20,000 amps. This rating fully supports the generator capability curve over the full range of generator voltage. Since there is no proposed change to the generator capability curve for MUR power uprate, there is no impact on the design duty for the isolated phase bus duct due to the MUR power uprate, and the isolated phase bus duct is acceptable for MUR power uprate operation.

Historically, output at the generator terminals has been less than 600 MVA (versus the generator's 659 MVA rating), or about 90 percent of the generator's nameplate rating. Electrical current in the isolated phase bus is expected to increase by an amount similar to the MUR power uprate increase in core thermal power, or by <2 percent. Under these conditions, the bus duct will continue to have adequate operating margin under the MUR power uprate conditions.

The isolated phase bus also provides auxiliary power to the 1M (Unit 1) and 2M (Unit 2) main auxiliary power transformers. The auxiliary branch of the bus which supplies these transformers has substantial margin (>10 percent) for current and MUR power uprate operations (1,500 amps rating versus the limiting analytical electrical current duty of 1,350 amps).

Cooling for the isolated phase bus is by forced cooling from the main branch and by "self cooling" for the auxiliary branch. The cooling is designed to support the 20,000 and 1,500 amp ratings for these buses. The ratings will continue to have adequate margin for MUR power uprate operations.

9. *Instrumentation and Controls Branch (EICB) RAI 1:*

On Pages 5 and 6 (Item I.C) in Enclosure 2 of the LAR, the licensee proposes to operate at uprated power in both "Alert" mode and "Fail" mode before the next scheduled daily Power Range Nuclear Instrumentation calibration.

Please list all conditions (e.g., one main steam pressure input fails) in these two modes separately, and provide additional explanation if necessary.

NSPM Response:

If the LEFM receives an "Alert" or "Fail" mode indication, NSPM proposes to continue to operate the facility at the 100% uprated power until the next daily surveillance of the power range Nuclear Instrumentation System (NIS), as required by Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.2. Operation at the licensed power level in this condition is permissible because the power range Nuclear Instruments (NIs) are not affected by the LEFM condition and are considered operable for providing required reactor protection for the remaining duration of this 24 hour surveillance interval, as described in Enclosure 2 to the LAR, item I.D.1 (page 9).

The LEFM system status indications are described as follows:

- Normal: An LEFM system "Normal" is displayed when all the feedwater flow, temperature, and header pressure signals for feedwater loops A and B are normal and operating within design limits. Calculated power level error associated with the LEFM flow measuring system in this condition is 0.36%.

- Alert: An LEFM system "Alert" alarm indicates a loss of redundancy and the calculated power level error associated with the LEFM flow measuring system in this condition is 0.54%. An "Alert" alarm is caused by:
    - Loss of a single process input
        - Loss of a single flow plane (loss of one or more flow transducers in a flow plane) on either feedwater line
        - Loss of a single flow plane (loss of one or more flow transducers in a flow plane) in both feedwater lines
        - Loss of a single redundant spool piece resistance temperature detector (RTD) on either line
        - Loss of a single redundant feedwater header pressure input
    - Loss of a single Electronics Unit redundant component. The Electronics Unit includes two redundant systems, each of which includes a separate power supply with 5 volt, +12 volt, -12 volt, and 180 volt outputs; four (4) acoustical signal processing units (APUs) which transmit and receive ultrasonic flow signals; and a central processing unit (CPU) which performs flow and temperature calculations, system self checks, and

system verifications. The loss of any one of these components would produce an "Alert" alarm.
- Process input or output is calculated outside a pre-determined allowable range
- Internal self-check indicates system parameters that exceed pre-established limits and affect a single plane in one or both loops; for example, problems could be identified with the Global Synchronization Signal (GSS) board, signal Rejects, signal Transit Time, path high gain, or speed of sound

- Fail: An LEFM system "Fail" alarm indicates a loss of function and the power level error reverts to the 2% error associated with the venturi flow meters. A "Fail" alarm is caused by:
  - Loss of both redundant process inputs
    - Loss of both flow planes (A & B) on a single or both feedwater lines
    - Loss of both redundant spool piece RTDs on a single loop
    - Loss of both feedwater header pressure inputs
  - Failure of both redundant components in the Electronics Unit, such as both 180 volt power supplies
  - A process input or output is calculated outside a pre-determined allowable range by both CPU units
  - Loss of the data link between the LEFM system and the Emergency Response Computer System (ERCS)
  - Internal self-check indicates system parameters that exceed pre-established limits and affect multiple planes in one or both loops; for example, problems could be identified with the GSS board, signal Rejects, signal Transit Time, path high gain, or speed of sound

10. EICB RAI 2:

On Pages 17 and 18 (Item I.G) in Enclosure 2 of the LAR, the licensee proposes to operate at uprated power in "Alert" mode with indefinite allowed outage time (AOT), and "Failed" mode with a 7-day AOT.

Please provide the data of known transmitter drift and nozzle fouling during the AOT to justify these proposed AOTs.

NSPM Response:

Operation at 100% uprated power with the LEFM system in either the "Alert" or "Fail" modes is only permitted until the next daily surveillance of the Nuclear Instrumentation System (NIS), as required by Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.2. As described below and on pages 6 and 9 of Enclosure 2 to the LAR, if the LEFM system is not restored to "Normal" operating status at that time, a reactor power reduction is required. Power must be reduced by an amount commensurate with the increase in uncertainty associated with the condition of the feedwater flow measurement system, i.e., either 0.54% or 2.0%.

In addition, the following background information is provided regarding use of the NIS and other power indication systems at PINGP to display "indicated power," as used in this response:

- When the Emergency Response Communication System (ERCS) computer is in service, reactor power is indicated and trended using the Thermal Power Monitor (TPM) display screen in the Control Room. The TPM display screen provides current calculated power indication based on the LEFM and Venturi calorimetric calculations, the average NIS power, and allowable power based on LEFM system status. In addition, the TPM display provides a 12-hour trend of the selected power monitor system. The Operator can select output from either the calorimetric or the NIS for trend display.

- Typically CALORIMETRIC is the selected indication of current reactor power. The operator can select either the LEFM (LCALM) or Venturi (VCALM) heat balance calculations as the displayed calorimetric output. When the LEFM system is in service, LEFM is the preferred calorimetric source. In the event the LEFM system fails, the calorimetric automatically shifts to the Venturi output. In the event the calorimetric program or the ERCS is not available, reactor power is monitored directly from the NIS indication in the control room.

The LEFM operating Modes and proposed AOTs are discussed as follows:

Normal Operation

"Normal" operation of the system includes normal functioning of the LEFM meter in each feedwater loop, wherein each Caldon CheckPlus™ LEFM meter contains two flow monitoring planes, and each plane includes four ultrasonic flow paths. The power measurement uncertainty of operating in the "Normal" mode with both planes in each meter operable is calculated to be 0.36%.

Alert Mode

As described in the response to the previous question, if any one plane in either or both LEFM meters is inoperable, the "Alert" alarm is received. This condition represents a loss of redundancy and the power measurement uncertainty is increased from 0.36% to 0.54%.

Operator response actions in this condition are as follows:

- Operation at licensed core power (1677 MWt, as indicated on the TPM and based on calorimetric heat balance calculations using feedwater flow measurements from the LEFM planes still in operation, LCALM) may continue until the next scheduled daily power calorimetric surveillance. The TPM computer program will change the Allowable Power Level from 1677 MWt (100%) to 1674 MWt (99.82%), and will display an alarm if the calculated current power level is greater than this Allowable Power Level.

  Justification: Operation in this condition is allowable because the Power Range Nuclear Instruments (NIs) are not affected by the LEFM condition and are considered operable for providing required reactor protection until their next daily calorimetric surveillance as required by TS SR 3.3.1.2. The alarm condition alerts operators to the potential need to reduce power if the LEFM is not restored to Normal status by the next daily calorimetric heat balance surveillance.

- If the LEFM is not restored to Normal operating conditions by the next scheduled daily calorimetric calibration, and if the LEFM remains in the "Alert" mode, then power level will be reduced to less than or equal to 1674 MWt (99.82%), as determined by the calorimetric heat balance calculated using the LEFM (LCALM). This condition may continue indefinitely until the LEFM is restored.

  Justification: With the LEFM in "Alert," the LEFM system measurement uncertainty increases from 0.36% to 0.54%. Reducing allowable power to less than or equal to 1674 MWt (analyzed power level of 1683 MWt – 0.54% [9 MWt] = 1674 MWt) corresponds to the increase in the LEFM measurement uncertainty during operation in the "Alert" Mode (operating with only one plane of flow sensors in either or both feedwater loops). The LEFM performs self-checks and does not exhibit drift, as illustrated by the fact that Cameron (Caldon) LEFM system can operate without calibration, with either one or two flow measurement planes. It is noted that this condition of a single measurement plane represents the configuration of a 'Caldon Check™' ultrasonic flow measurement system. Operating indefinitely with the LEFM in the "Alert" mode, in the reduced power condition, is equivalent to operating with a 'Caldon Check™' system as has been used in other nuclear plant flow monitoring systems.

Fail Mode

As described in the response to the previous question, if both LEFM planes in either or both feedwater loop meters are inoperable, the "Fail" alarm is received. This condition represents a loss of LEFM function. The power measurement calorimetric heat balance computer program automatically changes to use of the venturi-based flow measurement system (VCALM), and the assumed power measurement uncertainty is increased to 2%, which is the uncertainty associated with the current venturi-based flow measurement system.

Operator response actions in this condition are as follows:

- Operation at licensed core power (1677 MWt as indicated on the TPM and based on calorimetric heat balance calculations using feedwater flow measurements from the venturi-based system, VCALM) may continue until the next scheduled daily power calorimetric. The Thermal Power Monitor (TPM) will change the Allowable Power Level from 1677 MWt (100%) to 1650 MWt (98.38%), and will display an alarm if the current calculated power level is greater than this Allowable Power Level.

  Justification: Operation in this condition is allowable because the Power Range Nuclear Instruments (NIs) are not affected by the LEFM condition and are considered operable for providing required reactor protection until their next daily calorimetric calibration as required by Technical Specification SR 3.3.1.2. The alarm condition alerts operators to the potential need to reduce power if the LEFM is not restored to Normal status by the next daily calorimetric heat balance.

- If the LEFM remains in the "Fail" mode by the next scheduled daily calorimetric calibration, then power level will be reduced to less than or equal to 1650 MWt. This condition may continue for up to seven (7) days with the last known valid correction factors until the LEFM is restored. If the LEFM is restored to the "Alert" condition, then actions appropriate for that condition are taken as described above.

Justification: With the LEFM in "Fail," the calorimetric automatically reverts to using the venturi feedwater flow measurement system (VCALM) with the original assumed measurement uncertainty of 2%. Operation at less than or equal to 1650 MWt is consistent with the accuracy of the remaining operable flow measurement system, and is also consistent with operations prior to the MUR power uprate.

The thermal power measurement correction factors are used to normalize the power levels determined from the venturi-based feedwater flow measurements to the more accurate LEFM flow measurements. The correction factors are essentially continuous calibration factors for the venturi flow measurement system.

Operation for up to seven (7) days while using the last known good correction factors (prior to LEFM failure) in the venturi-based calorimetric heat balance calculation (VCALM) provides assurance that the licensed thermal power level is not exceeded when the calorimetric power calculation program automatically shifts from the LEFM-based feedwater flow measurements (LCALM) to VCALM. The seven-day period provides time to restore the LEFMs to their Normal status.

After seven (7) days, if the LEFM is not restored to service, correction factors for the VCALM feedwater flow and temperature inputs will be re-set to 1.0. If the correction factors are greater than 1.0, power level will be reduced an additional amount equivalent to the combined average of feedwater flow correction factors, as described in the LAR, Enclosure 2, pages 9 and 10. These actions will limit the deviation between actual core power and calculated core power determined using the venturi-based feedwater flow measuring system.

Instrument Drift and Nozzle Fouling Considerations

Evaluation and trending of correction factors allows identification of changes in deviations between the LEFM and venturi feedwater flow measurement systems, and could provide indication of instrument drift or nozzle fouling. As explained in Enclosure 2 to the LAR, item I.D.1 (page 9), plant specific trending of the venturi feedwater flow and temperature correction factors indicates potential drift of the feedwater venturi flow and temperature RTD instrumentation is less than 0.1% within seven days. Trending of correction factors reflects all factors that could contribute to differences between the LEFM and venturi flow measurement systems, such as individual instrument drift, instrument bias, or environmental effects. Attributing the 0.1% deviation in correction factors to instrument drift is conservative and bounds drift values based on manufacturer's data for the venturi flow measurement components. For the same seven-day time interval, flow measurement drift based on instrument vendor data is calculated to be approximately 0.012%.

As noted above, the LEFM system performs self checks and is not subject to drift. The LEFM CheckPlus™ systems have been installed since the February 2008 outage at PINGP Unit 1, and since the October 2008 outage in Unit 2. No drift or calibration issues regarding the LEFMs have been experienced.

NSPM and the LEFM vendor (Cameron) have reviewed the LEFM and venturi feedwater flow data for trends since installation of the Unit 1 and Unit 2 LEFM systems. This review has not identified any indications between the LEFM and venturi data which would represent nozzle fouling or defouling events.

11. Mechanical and Civil Engineering Branch (EMCB) RAI 1:

Section IV.1.B.ii in Enclosure 2 of Reference 1 [NSPM LAR for MUR-PU dated December 28, 2009] indicates that the cumulative usage factors for the PINGP systems, structures, and components (SSCs) within the scope of the LAR are bounded by the current licensing basis under the proposed MUR-PU conditions.

Based on the fact that the PINGP operating license is currently being considered for renewal by the NRC staff, please indicate whether consideration has been given to the impact of the proposed MUR-PU conditions on the fatigue evaluations for the SSCs which are within the scope of this LAR.

Please provide justification that demonstrates that the 60 year plant life, for which review has been requested, has been accounted for in the fatigue evaluations of the SSCs included within the scope of the MUR-PU LAR.

NSPM Response:

The fatigue evaluations performed to qualify the affected NSSS components to MUR uprate conditions do consider the anticipated PINGP operating license renewal to sixty (60) years.

USAR Section 4.1.4 describes the transients considered in the NSSS component fatigue analyses and Table 4.1-8 of the USAR defines the quantity of occurrences for each transient assumed in the fatigue analyses. USAR Sections 3 and 4 describe the NSSS components subject to fatigue analysis.

An evaluation of transient occurrences determined that the original quantity of transients used for fatigue evaluation for a forty-year plant life remains valid for fatigue evaluation for an anticipated sixty-year plant life. The PINGP Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and records actual transient occurrences. The actual quantity of transient occurrences for the first thirty years of plant life were prorated, linearly, to determine the quantity of projected transient occurrences for the remaining thirty years of a sixty-year plant life.

This data was included in the PINGP License Renewal Application (LRA) submitted April 11, 2008 (ADAMS Accession number ML081130663). For each transient condition evaluated, Table 4.3-1 of the LRA identifies the number of design cycles, number of actual transients experienced through September 30, 2006, and the number of transients projected for a sixty-year plant life. The quantity of transient occurrences projected for a sixty-year plant life is less than the number of design cycles originally assumed for the forty-year plant life.

For the MUR power uprate evaluations of the major NSSS components, the original design quantities of occurrences were retained, as defined in USAR Table 4.1-8 with few exceptions, offering considerable margin over the projected quantity of transient occurrences expected for a sixty-year plant life. The exceptions include certain subcomponents that were evaluated for fatigue using less than the original design number of transient cycles. The quantity of transient occurrences used in the fatigue analyses for these subcomponents still provides significant margin when compared to the number of cycles expected during the 60-year plant life, and fatigue usage factors are conservatively less than 1.0.

In addition to the design transients defined in USAR Table 4.1-8, some NSSS sub-components are evaluated for other, more time- or power-dependent, fatigue degradation mechanisms like flow-induced vibration, gamma heat generation and thermal stratification. The MUR Uprate fatigue evaluations for these mechanisms also considered the anticipated sixty-year plant life.

The fatigue evaluations of the major NSSS components used transient profiles that bound MUR Uprate operating conditions and the quantity of transient occurrences defined in USAR Table 4.1-8, with few exceptions. All component fatigue evaluations performed for the MUR power uprate program yielded acceptable results, with cumulative usage factors less than 1.0, for an anticipated 60-year life.

12. EMCB RAI 2:

As a result of higher-than-design moisture carryover (MCO) percentages for the Unit 2 steam generators, which would be realized at MUR uprate conditions, Section IV.B.i in Enclosure 2 of Reference 1 [NSPM LAR for MUR-PU dated December 28, 2009] (for balance of plant piping and components) indicates that "...a revision of the MS [main steam] stress analysis to ensure MSSV [main steam safety valve] thrust force is acceptable at the higher-than-design MCO condition..." was initiated in support of the proposed MUR-PU.

Please provide justification which demonstrates that the Unit 2 MS piping, MS piping supports, and MS system components remain acceptable by comparison to the design-basis code allowable values (or other design-basis qualification standard) when considering the MSSV thrust forces at the higher-than-design MCO conditions.

NSPM Response:

NSPM's re-analysis of the Main Steam (MS) piping system is still in progress and scheduled completion dates are still being discussed with the engineering firm that has been performing the analyses. The re-analysis is scheduled for completion in mid-August 2010. NSPM will provide justification that demonstrates the stress levels in Main Steam system piping, components and supports remain less than the allowable levels by August 27, 2010.

The following new commitment is identified for this action:

> NSPM will submit justification for the Main Steam system stress analysis, including piping, supports, and components by August 27, 2010.

13. *EMCB RAI 3:*

Section IV.1.A.ii in Enclosure 2 of Reference 1 [NSPM LAR for MUR-PU dated December 28, 2009] states that operation at the proposed MUR-PU conditions will not adversely affect the structural integrity of the reactor vessel internals and core support structures.

Please verify whether the current analysis of record (AOR) remains bounding for the reactor vessel internals and core support structures at the proposed MUR uprate conditions. If the AOR is not bounding, wholly or in-part, please provide the updated analyses results for the core support structures and/or reactor vessel internals which are not bounded under the proposed MUR uprate conditions.

NSPM Response:

The current analysis of record remains bounding for the reactor vessel internals and core support structures at the proposed MUR Uprated conditions.

NSPM performed a single set of evaluations/analyses for the reactor vessel internals that envelop both the MUR Uprate operating conditions and the transition to 422V+ Fuel. The evaluations/analyses constitute the AOR and address the following:

1. The thermal-hydraulic behavior of the coolant flow and its effect on reactor vessel internals pressure drop, core bypass flows, reactor pressure vessel fluid temperatures, hydraulic lift forces and baffle joint momentum flux.
2. Gamma heat generation rates and the thermal effects on core baffle plates, former plates, core barrel, baffle-former bolts, barrel-former bolts, thermal shield, and the upper and lower core plates.
3. The effects of the thermal-hydraulic conditions on fuel rod stability; reactivity control cluster assembly (RCCA) drop time performance; and reactor internals hold-down spring performance.
4. The effects of LOCA, seismic, flow-induced vibration and RCP-induced vibration on stress levels, fatigue usage and/or displacement, as appropriate, in internals and core support components.

The reactor vessel internals evaluations/analyses are described in more detail in NSPM's License Amendment Request for the transition to 422V+ Fuel, NSPM Letter L-PI-08-047, dated June 26, 2008 (ADAMS Accession Number ML081820137). Supplemental information is provided in response to Requests for Additional Information in NSPM Letter L-PI-09-034, dated March 12, 2009 (ADAMS Accession Number ML090721087). The evaluations/analyses concluded that performance of the reactor internals and core support structures meets current license basis acceptance criteria at the proposed MUR uprate operating conditions.

14. Containment and Ventilation Branch (SCVB) RAI 1:

Please provide the current (operating at 1650 MWt power) and the MUR-PU (operating at 1677 MWt power) reactor vessel cold leg inlet fluid and hot leg outlet fluid temperatures. For the short term loss-of-coolant accident containment response analysis, please explain why the mass and energy release for the cold leg and hot leg breaks in the current analysis bounds the same for the analysis under MUR-PU conditions.

NSPM Response:

The reactor vessel cold leg inlet and hot leg outlet fluid temperatures, for both the current 1650 MWt licensed Rated Thermal Power (RTP) level and for the proposed uprated 1677 MWt RTP level are as follows:

Parameter	Temperatures at Current RTP (°F)	Temperatures at Uprated Power (°F)
T-hot	592.1	592.6
T-cold	527.9	527.4
T-avg	560.0	560.0

These temperatures were provided in Table IV.1B.4 in Enclosure 2 to the LAR (page 55) and their basis is as follows.

Temperatures at Current RTP
The temperatures at current RTP conditions are included in PINGP USAR, Table 3.2-1, "Thermal and Hydraulic Design Parameters," for 1650 MWt reactor power. These temperatures are associated with an NSSS power of 1657 MWt (1650 MWt + 7 MWt reactor coolant pump heat).

Temperatures at Post-MUR Uprate Conditions
Temperatures for post-MUR power uprate conditions were developed by Westinghouse for the MUR power uprate program (Reference IV.2.5 in the LAR). These temperatures are associated with a 1690 MWt NSSS power level, which corresponds to a core power of 1683 MWt and 7 MWt for reactor coolant pump heat. The 1683 MWt analyzed core power level reflects the proposed uprate core power level of 1677 MWt with a 0.36% uncertainty.

The 0.5 degree differences in hot leg and cold leg temperatures between the current RTP conditions and the uprated conditions shown in the table above are not considered significant relative to LOCA containment response analyses. The difference in water densities at these temperatures does not result in a significant difference in mass flows, and the temperature difference is within the 4 degree uncertainty assumed for RCS temperature values in USAR Appendix K, "Containment Pressure Response to LOCA." Therefore, mass flow values in the current licensing basis (CLB) are not affected by the MUR power uprate.

Short Term LOCA Containment Response

There are two aspects to containment response to a LOCA event, one involving subcompartment pressurization and the other involving peak environmental conditions within the containment vessel.

Subcompartment pressurization issues:

- Subsequent to the original licensing activities, leak-before-break (LBB) methodology was applied to the main reactor coolant system piping. Application of LBB has reduced the mass and energy releases to values well below the original licensing basis analysis.
- USAR Section 4.6.2.3, "Elimination of Large Primary Loop Pipe Rupture as the Structural Design Basis," states that with LBB, the probability or likelihood of large pipe breaks occurring in the primary coolant system loops of PINGP Units 1 and 2 is sufficiently low such that dynamic effects associated with postulated pipe breaks in these facilities need not be a design basis. This section also states that one of the dynamic effects that does not have to be addressed is the reactor vessel cavity or subcompartment pressurization including asymmetric pressure effects.
- With the application of LBB methodology to main RCS piping, postulated piping break locations need not include large loop piping, but smaller branch connections such as the accumulator injection line and pressurizer surge line must be considered. Differential pressures resulting from postulated line breaks in these branch locations were evaluated for MUR conditions along with other legacy parameter changes. This evaluation concluded that faulted differential pressure loads considered in the CLB analysis of record (AOR) remain bounding for MUR conditions.

Peak containment conditions:

- LOCA mass and energy releases for cold leg and hot leg breaks are currently analyzed for a core thermal power of 1683 MWt, which is consistent with the uprated 1677 MWt RTP, with a 0.36% uncertainty. This power level is included in PINGP USAR, Appendix K, "Containment Pressure Response to LOCA," Section 14.6, "Large Break LOCA Analysis," and Section 14.7, "Small Break LOCA Analysis," in both the existing USAR version and in updates currently in progress.

Bounding mass and energy release:

- The existing containment integrity evaluations remain valid for MUR conditions because the analyzed values for reactor vessel average temperature, thermal design flow, and RCS pressure are the same for both current power levels and for MUR uprate power levels, and therefore the energy that can be released from the RCS at MUR conditions is essentially the same as the AOR. This comparison of energy releases was performed by Westinghouse in support of an LAR to use Westinghouse 422V+ fuel, which was approved in License Amendments 192/181. That evaluation was performed to address potential future operating conditions and included MUR uprated power levels.

Conclusion:

The difference between cold leg and hot leg fluid temperatures for current and post-MUR power uprate conditions is not significant. Also, the short term LOCA containment response analysis for the MUR power uprate conditions shows that subcompartment pressurization loads are acceptably less than allowable loads and the peak LOCA containment pressure and temperature conditions are bounded by the CLB. For the short term LOCA containment response analysis, the mass and energy releases resulting from a hot leg or cold leg break are essentially the same at MUR and current operating conditions.

15. Reactor Systems Branch (SRXB) RAI 1:

Describe and provide drawings of the location where the unified field mechanics [UFMs, ultrasonic flow meters] are installed. The information should be sufficient for the NRC staff to perform an in-depth comparison of the Alden Labs test configuration with the in-plant configuration.

NSPM Response:

The Cameron (formerly Caldon) Leading Edge Flow Meters (LEFM) are installed in straight piping runs in the 16-inch feedwater lines at PINGP. The LEFMs in Unit 1 are installed downstream of the venturi flow nozzles, and in Unit 2 the LEFMs are upstream of the venturi flow nozzles, due to slight differences in arrangement.

The installed positions are shown on the following drawings included as Enclosure 2:

- PINGP Unit 1 Feedwater piping isometric drawing XH-106-130, Revision 78, which shows the as-built location of the LEFM flow meters and venturi flow nozzles
- PINGP Unit 2 Feedwater piping isometric drawing XH-1106-246, Revision 77, which shows the as-built location of the LEFM flow meters and venturi flow nozzles

Cameron tested each LEFM at Alden laboratories to determine the individual meter factor and individual path normalized velocities, as part of Unit-specific calibration testing described in the LAR, Enclosure 2, Section I.1 (page 2). The test configuration for each unit is shown on the following drawings, also included in Enclosure 2:

- Caldon drawing SKRSH-32C.DWG, sheets 1 and 2, which show the configuration of the hydraulic model used at Alden laboratories to calibrate the LEFMs installed in Unit 1
- Caldon drawing SKRSH-32B.DWG, sheets 1 and 2, which show the configuration of the hydraulic model used at Alden laboratories to calibrate the LEFMs installed in Unit 2

As noted in the LAR, Enclosure 2, section I.1.D.4 (page 13), the PINGP LEFM flow elements are installed in the same piping configuration as tested at Alden laboratories, with the exception that the installation of each LEFM in Unit 2 is eight (8) inches closer to the inlet of the venturi flow nozzles than in the Alden laboratories configuration. The as-installed location of each LEFM was reviewed by Cameron and determined to meet the requirements of Caldon Topical Reports ER-80P and ER-157P. Post-installation commissioning testing verified that the actual installations remain bounded by the calibration assumptions.

ENCLOSURE 2

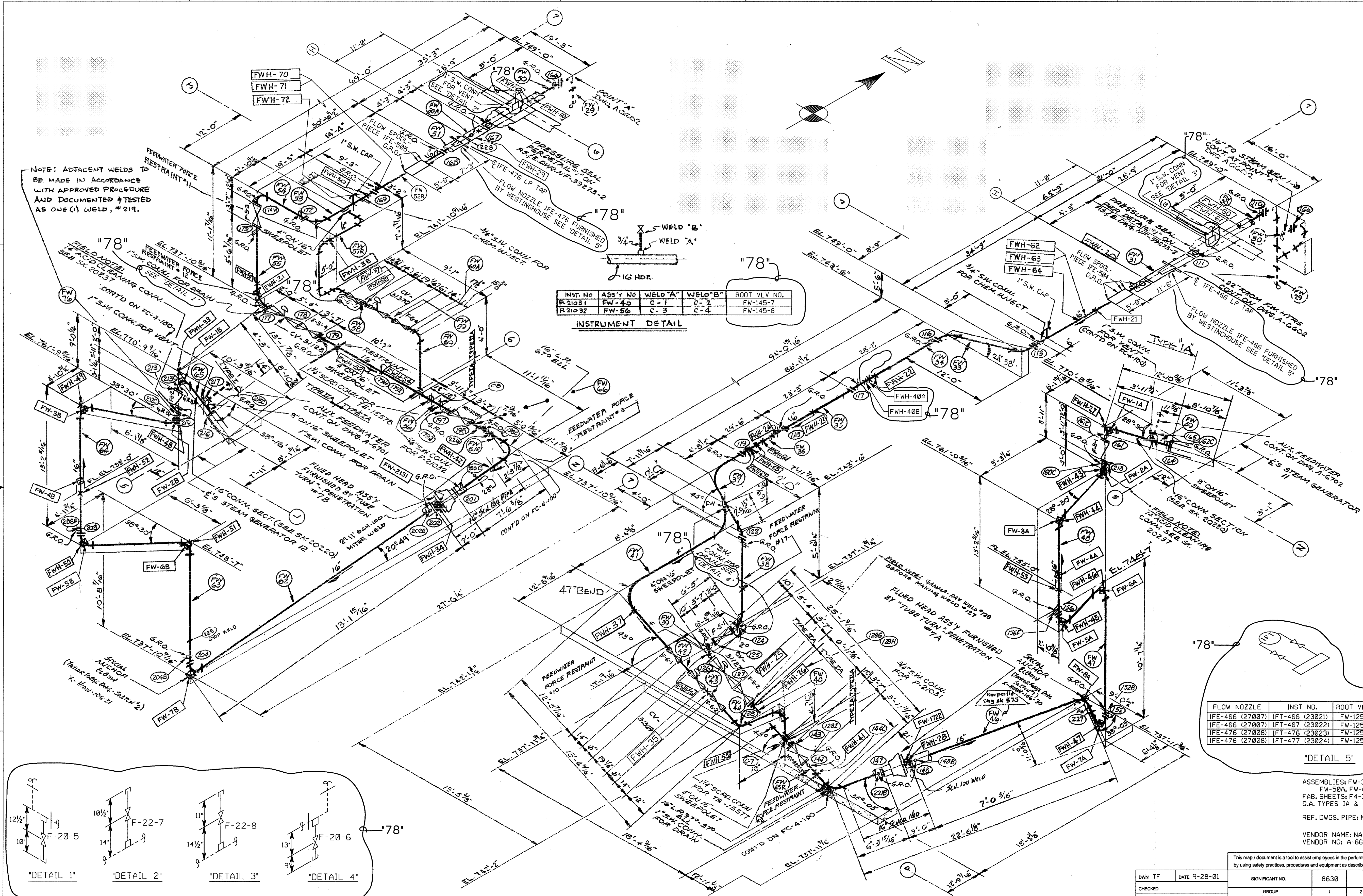
LEFM Flowmeter Installation and Test Drawings

This enclosure includes the following drawings to support the response to RAI Question: *Reactor Systems Branch (SRXB) RAI 1.*

- PINGP Unit 1 Feedwater piping isometric drawing XH-106-130
- PINGP Unit 2 Feedwater piping isometric drawing XH-1106-246
- Caldon drawing SKRSH-32C.DWG, sheets 1 and 2
- Caldon drawing SKRSH-32B.DWG, sheets 1 and 2

6 pages follow

XH-106-130



INST. NO	ASS'Y NO	WELD "A"	WELD "B"	ROOT VLV NO.
P-21031	FW-40	C-1	C-2	FW-145-7
P-21032	FW-56	C-3	C-4	FW-145-8

INSTRUMENT DETAIL

"DETAIL 1" "DETAIL 2" "DETAIL 3" "DETAIL 4"

FLOW NOZZLE	INST NO.	ROOT VLV LO	ROOT VLV HI
1FE-466 (27007)	1FT-466 (23021)	FW-125-1	FW-125-2
1FE-466 (27007)	1FT-467 (23022)	FW-125-3	FW-125-4
1FE-476 (27008)	1FT-476 (23023)	FW-125-5	FW-125-6
1FE-476 (27008)	1FT-477 (23024)	FW-125-7	FW-125-8

"DETAIL 5"

REVISIONS

A AS BUILT- RELOCATED VENT VALVES F-22-7 AND F-22-8.

PER DRR PI-01-105

DWN: TF 10-01-01
CHK'D: CMR 10-3-01
MOD#: SPCE ME-0534
FILMED 11-01

76 ISSUED FOR CONSTRUCTION

REVISE PER DCP 05FW02 PART A, ADD FLOW SPOOL-PIECE 1FE-504 & 1FE-505 PER 05FW02

DWN: DMN 11-28-05
CHK'D: LAW 6-14-06
MOD#: EC-1009
REVIEWED: JOH
APP'D & CERT:
JAMES O. HILL
6-14-06 PE# 18863

ISSUED FOR CONSTRUCTION

CORRECTED TYPO ERRORS FE-27008 WAS FE-27007 FWH-29 WAS FWH-21 PER 05FW02

DWN: DMN 4-20-07
CHK'D: LW 4-20-07
MOD#: EC-1009
REVIEWED: JH 4-20-07
APP'D & CERT:
JAMES O. HILL
4-20-07 PE# 18863

AS BUILT- INCORPORATED A/E's REV. 76 PER DRR PI-08-042 CERTIFIED REV. 76, TRANSFERRED TO RECORD TRACING.

DWN: DMN 6-19-08
CHK'D: LW 6-27-08
MOD#: EC-1009
FILMED: 7-08

77 AS BUILT- RELOCATE FWH-35

PER DRR PI-08-047

DWN: DMN 7-10-08
CHK'D: PMB 7-14-08
MOD#: EC-8295
FILMED: 7-08

78 AS BUILT- REV'D FWH-40 TO FWH-40A, 40B, FE- 27007 TO 1FE-466, FE-27008 TO 1FE-476, LOC OF VENT VLVS. F-22-7, 8. ADDED DET 2, 3 PER SPCE-ME-0534, DET 1, 4, 5 & ROOT VLV NOS. FW-145-7,8 TO INSTR DETAIL.

PER DRR PI-09-043

DWN: DB 4-28-09
CHK'D: JK 4-29-09
MOD#: EC-14091
FILMED 9-09

ASSEMBLIES: FW-30 THROUGH FW-66
FW-50A, FW-60A & FW-76
FAB. SHEETS: F4-30 THROUGH F4-72 & 76
Q.A. TYPES IA & IIA AS NOTED.

REF. DWGS. PIPE: NF-39273-1 & 2

VENDOR NAME: NATIONAL VALVE AND MFG. CO.
VENDOR NO: A-6603

NOTES:

1. ALL BUTTWELDS TO BE T.I.G. WELDS (SK.18580) EXCEPT SPEC. BACK RINGS (SK.20229) & M.AW. TO BE USED FOR WELDS 126 127 132 132 133 134 135 136 137 177 178 188 190 191 192 193

SIGNIFICANT NO.	8630		400	1	9072	1	3120
GROUP	1	2	3	4	5	CL	6

DWN TF DATE 9-28-01
CHECKED
PROJECT NO.
APP'D & CERT.
FILMED

FEEDWATER PIPING

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

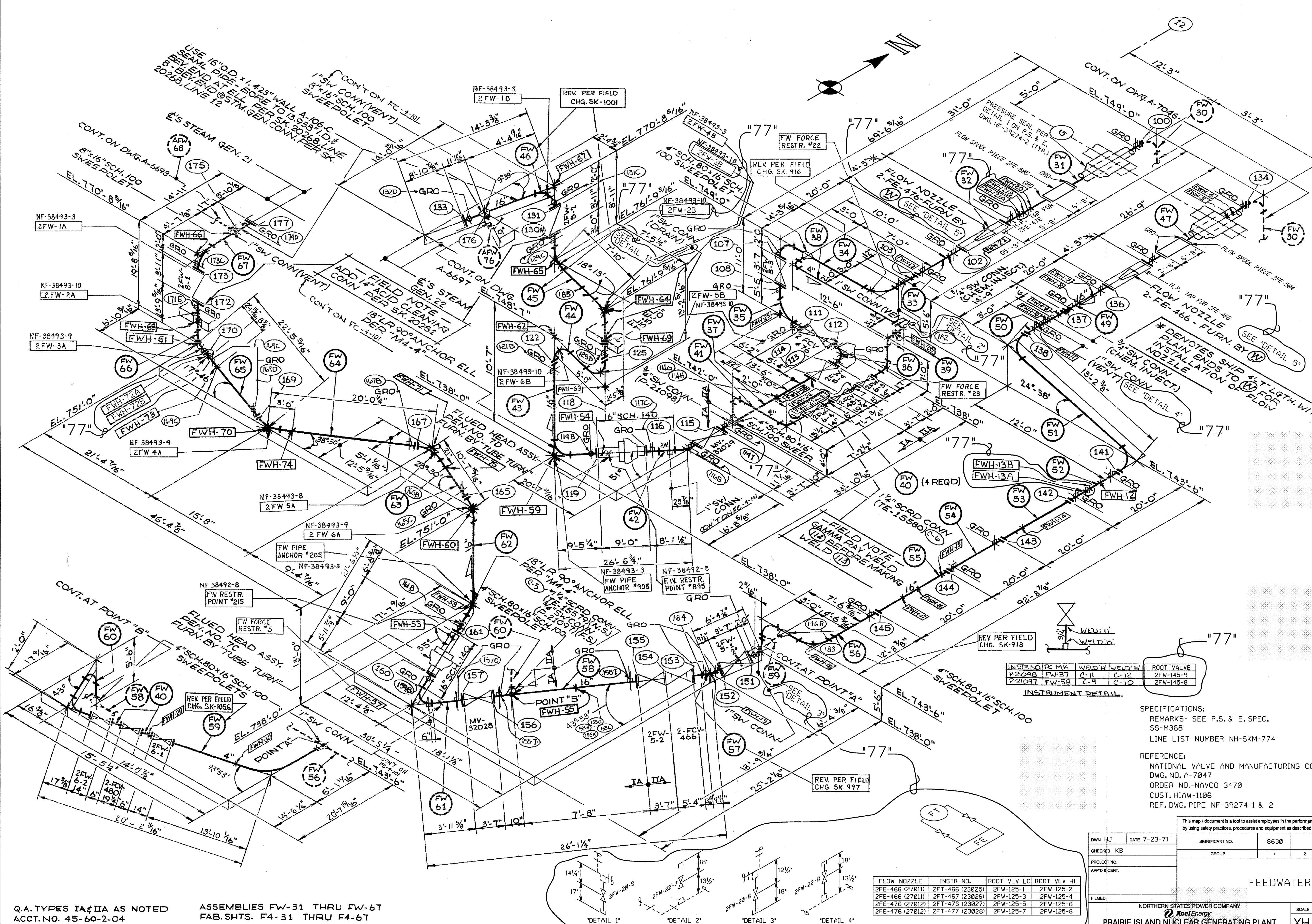
SCALE NONE REV 78

XH-106-130

XH-106-130.DGN NSP GENERATION CAD

$$USERNAME$$ $$$$$SYTIME$$$$$$$SYDATE$$$$$$ $$$$$$$$$DESIGNFIL$$$$$$$$$

XH-1106-246



INSTR NO	PC MK	WELD 'A'	WELD 'B'	ROOT VALVE
P-21098	FW-37	C-11	C-12	2FW-145-9
P-21097	FW-58	C-9	C-10	2FW-145-8

INSTRUMENT DETAIL

FLOW NOZZLE	INSTR NO.	ROOT VLV LO	ROOT VLV HI
2FE-466 (27011)	2FT-466 (23025)	2FW-125-1	2FW-125-2
2FE-466 (27011)	2FT-467 (23026)	2FW-125-3	2FW-125-4
2FE-476 (27012)	2FT-476 (23027)	2FW-125-5	2FW-125-6
2FE-476 (27012)	2FT-477 (23028)	2FW-125-7	2FW-125-8

"DETAIL 5"

SPECIFICATIONS:
REMARKS- SEE P.S. & E. SPEC. SS-M368
LINE LIST NUMBER NH-SKM-774

REFERENCE:
NATIONAL VALVE AND MANUFACTURING CO.
DWG. NO. A-7047
ORDER NO.-NAVCO 3470
CUST. HIAW-1106
REF. DWG. PIPE NF-39274-1 & 2

Q.A. TYPES IA & IIA AS NOTED
ACCT. NO. 45-60-2-04

ASSEMBLIES FW-31 THRU FW-67
FAB. SHTS. F4-31 THRU F4-67

REVISIONS

A ISSUED FOR CONSTR.- ADDED LEADING EDGE FLOWMETER.
DWN: KEY
CHK'D: PJD
PROJ#: E-94L472
REV'D:
APP'D & CERT:
J.H. HALL
4-18-95 PE# 4-20-9

AS BUILT- INCORPORATED A/E'S REV. A PER DRR PI-95-232 CERTIFIED REV. A TRANSFERRED TO RECORD TRACING
DWN: BMS 12-29-95
CHK'D: PAS 1/4/96
MOD#: 94L472
FILMED 1/8/96

B AS BUILT- DELETED SEISMIC RESTRAINT FWH-24.
PER DRR PI-97-123
DWN: JDS 9/8/97
CHK'D: PAS 9-11-97
MOD#:
FILMED 9-97

76 ISSUED FOR CONSTRUCTION
ADDED FLOW SPOOL PIECES AND DIMENSIONS PER DCP Ø5FWØ2, PART A. (Ø5FWØ2)
DWN: DMN 11-18-05
CHK'D: LAW 6-14-06
MOD#: EC-390
REVIEWED: N/A
APP'D & CERT:
JAMES O. HILL
6-14-08 PE# 18863

ISSUED FOR CONSTRUCTION
REVISED DIMENSIONS PER EC-13220. (Ø5FWØ2)
DWN: DMN 9-24-08
CHK'D: LAW 9-25-08
MOD#: EC-390
REVIEWED: N/A
APP'D & CERT:
JAMES O. HILL
9-25-08 PE# 18863

AS BUILT- INCORPORATED A/E's REV. "76" PER DRR PI-08-091 CERTIFIED REV. "76", TRANSFERRED TO RECORD TRACING.
DWN: LAB 11-7-08
CHK'D: LAW 11-8-08
MOD#: EC-390
FILMED: 12-08

77 AS BUILT- ADDED VENT & DRAIN DETAILS 1 THRU 4, & ROOT VALVE DETAIL 5. ADDED ROOT VALVE NOS. FOR P-21097 & P-21098. REV'D HNGRS FWH-13 TO 13A & B, FWH-21 TO 21A & 21B, FWH-27 TO 27A & 27B, AND FWH-72 TO 72A & B. REV'D 2-FW-2B TO 2-FW-3B & 2-FW-3B TO 2-FW-2B PER WALKDOWN.
Ø4NS15
PER DRR PI-09-043
DWN: JEK 4-28-09
CHK'D: DB 4-30-09
MOD#: EC-14091
FILMED 9-09

SIGNIFICANT NO.	8630			2	M481 / 9072	1	
GROUP	1	2	3	4	5	CL	6

DWN HJ DATE 7-23-71
CHECKED KB
PROJECT NO.
APP'D & CERT.
FILMED

FEEDWATER PIPING

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

SCALE NONE REV 77

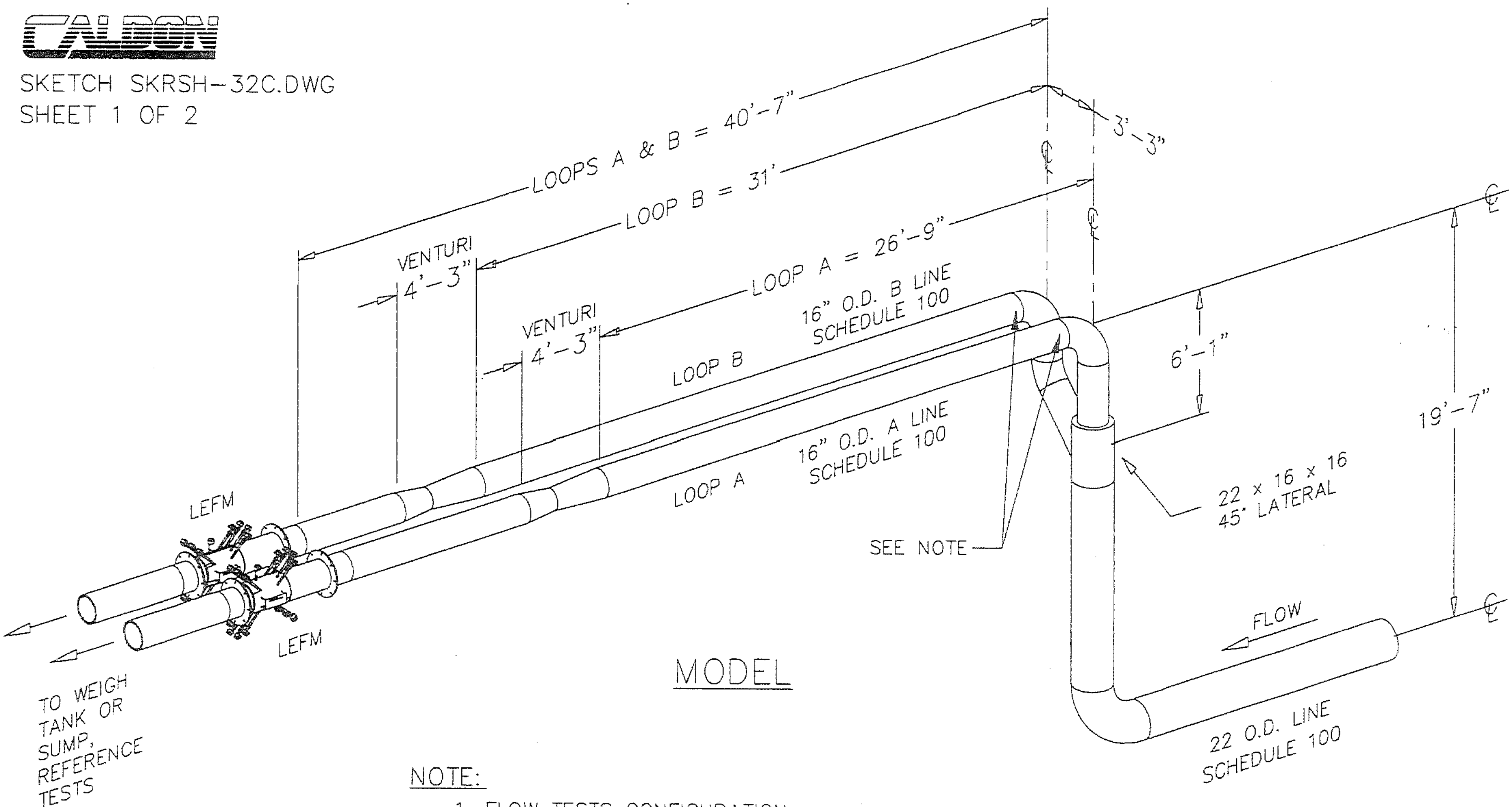
XH-1106-246

XH-1106-246.DGN NSP GENERATION CAD

$$$$$$$SYTIME$$$$$$$USERNAME$$$$$$$$$SYDATE$$$$$$$
$$$$$$$$$$DESIGNFIL$$$$$$$$$$

CALDON

SKETCH SKRSH-32C.DWG
SHEET 1 OF 2



NOTE:

1. FLOW TESTS CONFIGURATION:

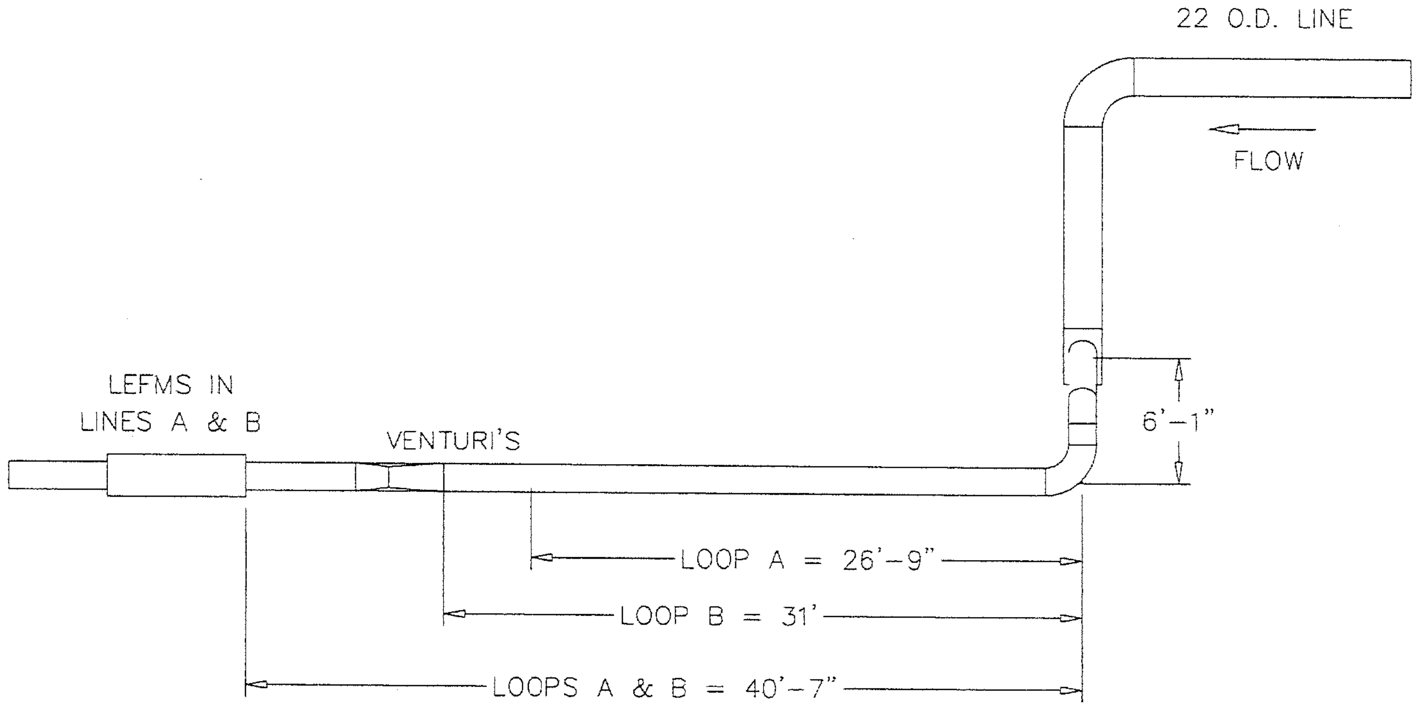
   A: FLOW TESTS A-2 & B-3 IN SECTION 6.2 WILL USE HALF MOON PLATES AT THE EXIT OF BOTH ELBOWS.

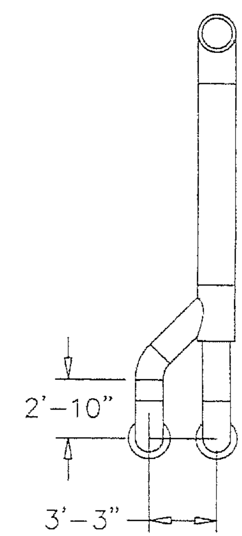
   B: FLOW TESTS IN SECTION 6.3 USE A MITSUBISHI FLOW CONDITIONER INSTALLED AT THE EXIT OF BOTH ELBOWS AND WILL HAVE THE HALF MOON PLATES REMOVED.

TYPICAL PIPING CONFIGURATION
AND LEFM LOCATION
PRAIRIE ISLAND 1

CALDON

SKETCH SKRSH-32C.DWG
SHEET 2 OF 2

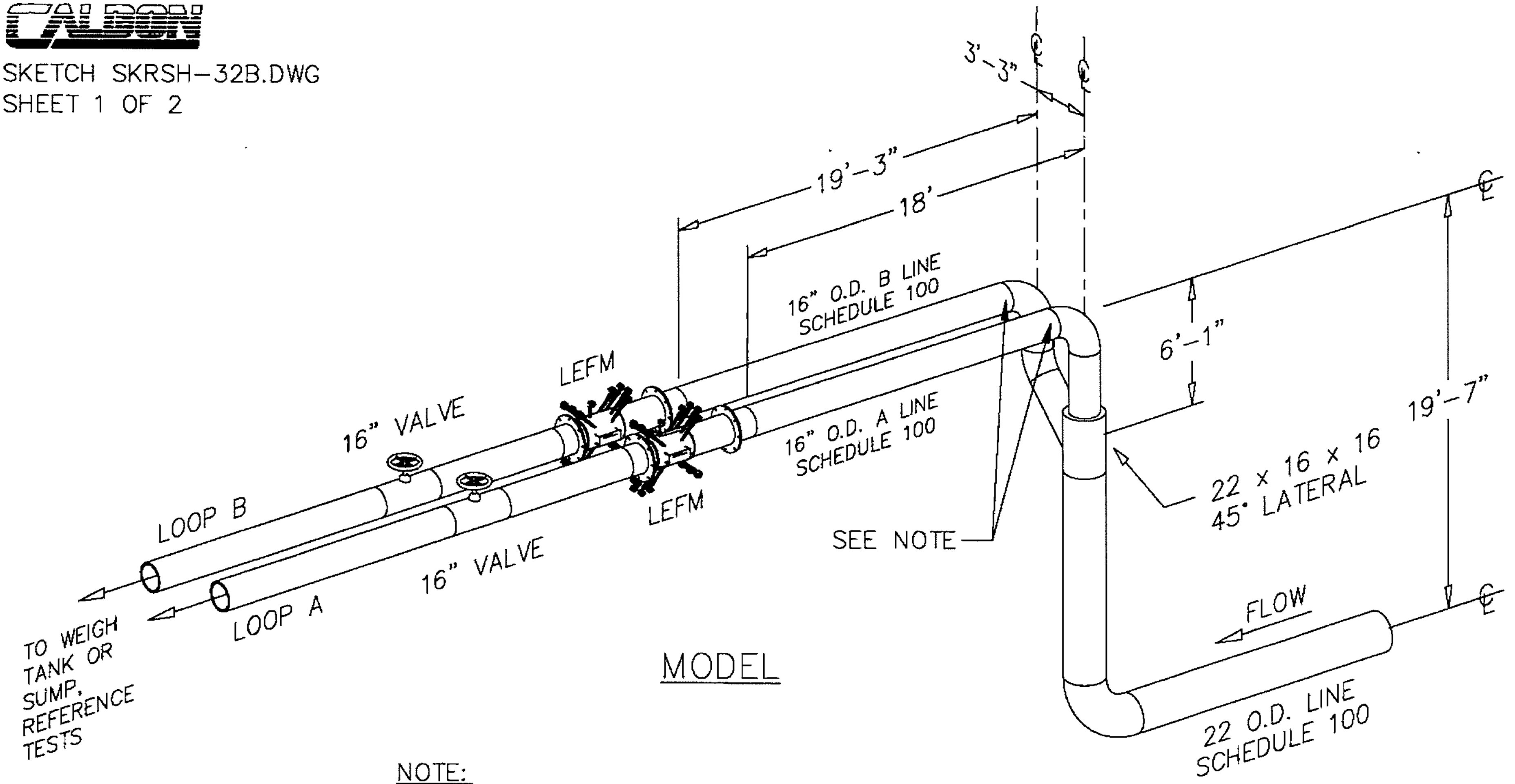




TYPICAL PIPING CONFIGURATION
AND LEFM LOCATION
PRAIRIE ISLAND 1

CALDON

SKETCH SKRSH-32B.DWG
SHEET 1 OF 2



MODEL

NOTE:

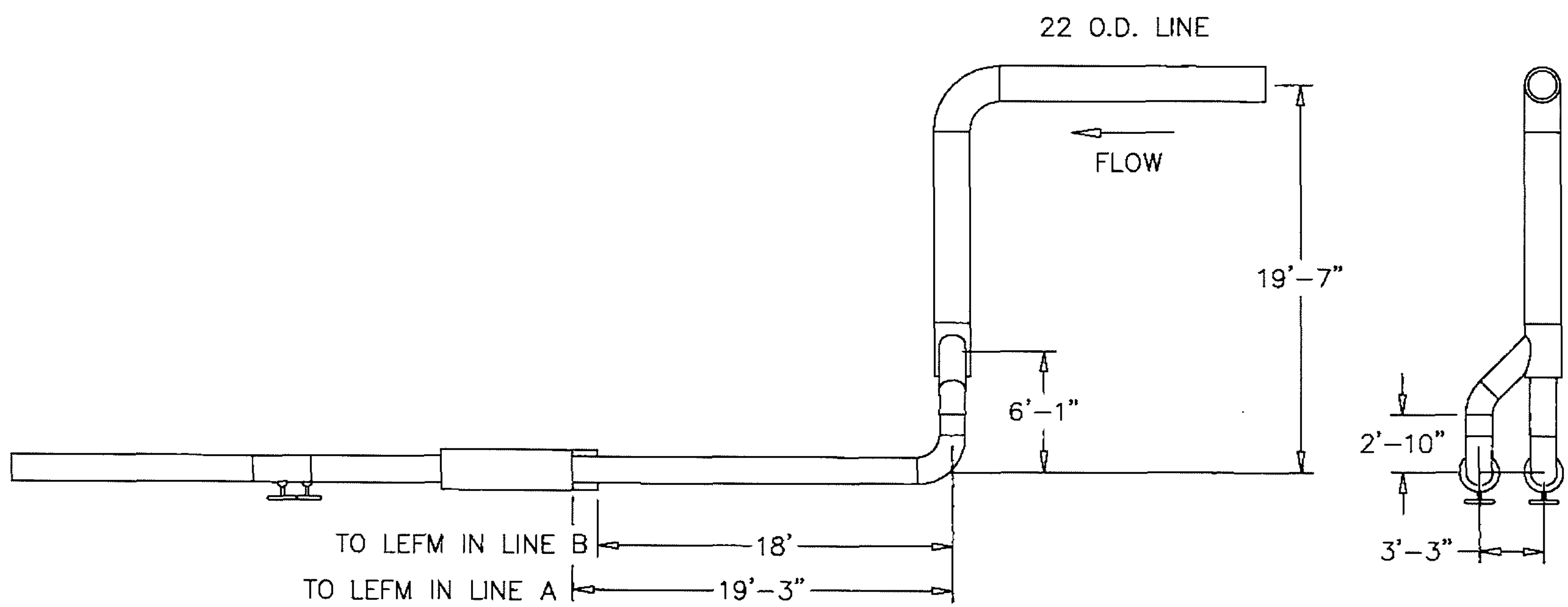
1. FLOW TESTS CONFIGURATION:

   A: FLOW TESTS A–2 & B–3 IN SECTION 6.2 WILL USE HALF MOON PLATES AT THE EXIT OF BOTH ELBOWS.

   B: FLOW TESTS IN SECTION 6.3 USE A MITSUBISHI FLOW CONDITIONER INSTALLED AT THE EXIT OF BOTH ELBOWS AND WILL HAVE THE HALF MOON PLATES REMOVED.

TYPICAL PIPING CONFIGURATION
AND LEFM LOCATION
PRAIRIE ISLAND 2

SKETCH SKRSH-32B.DWG
SHEET 2 OF 2



TYPICAL PIPING CONFIGURATION
AND LEFM LOCATION
PRAIRIE ISLAND 2

upm