NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
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September 17, 2001

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Response to Opportunity for Comment on TIA 2001-02, Design-Basis Assumptions for Non-Seismic Piping Failures (TAC Nos. MB1402 and MB1403)

By letter dated July 17, 2001, the NRC provided Nuclear Management Company, LLC, (NMC) with the opportunity to respond to the issues identified in Task Interface Agreement (TIA) 2001-02. The attachment to this letter is the Prairie Island response to the issues identified in TIA 2001-02.

In this letter we have made no new Nuclear Regulatory Commission commitments. Please contact Jeff Kivi (651-388-1121) if you have any questions related to this letter.

M K. Nazar

Mano Nazar
Site Vice President
Prairie Island Nuclear Generating Plant

A025

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg

Attachments:

1. Response to Task Interface Agreement (TIA) 2001-02

Attachment 1

Prairie Island Nuclear Generating Plant
Response to Task Interface Agreement (TIA) 2001-02

1. Cooling Water (essential service water) System Description

   The Cooling Water (CL) system provides cooling for both safety related and non safety related components. The system is shared between the two units. The system consists of two trains that are automatically split on a safety injection signal in either unit. The system is shown on the attached figures. There are several piping connections to the supply headers which supply water to cool non safety related loads. The single largest non safety related supply line (Turbine Building loads) can be isolated by closing a safety related motor operated valve. This valve closes automatically on a SI signal with a low pressure condition in the safety related supply header. All of the other isolation valves for these non safety related lines are manual valves. During a seismic event, without a SI signal, all of these valves (Turbine Building isolation and manual valves) would be open.

2. Task Interface Agreement 2001-02

   The subject TIA requests NRR assistance in resolving an issue regarding a plant position on CL System non safety related piping performance in a seismic event. In the hydraulic modeling of the CL system, the plant position has been that this piping would not catastrophically fail in a seismic event (this position is described in more detail later). Instead a single crack is postulated in each non safety related branch line off of the main supply headers. To provide consistent method in the hydraulic modeling, BTP MEB 3-1 was used to determine the size of the cracks assumed in the hydraulic model (Note that BTP MEB 3-1 and postulating moderate energy line breaks is not within the plant design basis.) The basis for this position was provided to the Region III inspectors during the Safety System Design and Performance Capability Inspection of the CL System (Inspection Report 50-282/00-13; 50-306/00-13). This position is based on performance of piping in actual seismic events.

   Included in the paper was a discussion that these non safety related portions of the CL System had been evaluated for a UBC Zone 1 earthquake load of 0.05g; however, this was included more for information and not as a basis for the position. This part of the discussion was based on a plant general criteria that QA Type IIB piping would also be Design Class II. The CL system piping diagrams show the non safety related piping as QA Type IIB. In accordance with plant design, Class II piping includes UBC Zone 1 seismic loads. Further review of the actual piping stress analyses indicates that non safety related piping to the first anchor from the safety related piping was included in the stress analyses. An analysis for the UBC Zone 1 loading on the non safety related piping has not been located, and it is now believed that the QA Type IIB categorization was used to ensure quality in materials and craftsmanship and not translated into the piping design. Based on this information, the second requested action in the TIA is not relevant.

3. Design & Licensing Bases

A review was performed of plant design and licensing basis to try and ascertain the bases and possible review of the current design of the CL system. The system configuration, with the non safety related lines connecting to the safety related lines is readily apparent and should have been easily noticed during the design and the review of the system.

This design and licensing review focused on the Safety Analyses Report, the AEC Safety Evaluation Report (SER) and any other applicable licensing correspondence.

The initial design, as documented in the FSAR, included automatic isolation of non essential loads on a SI signal. For example, FSAR, Amendment 12, dated 11/15/71, Page 9.6-9, with regards to the effect a SI signal has on the CL System states (in paragraph 3) that:

> "Isolates all unessential loads of the affected unit from the side of the split ring header that normally supplies these loads, except the turbine oil cooler which can be manually isolated from the control room."

Note that this still would not isolate the loads in response to a seismic event without a SI signal.

The AEC SER, initial issue dated 9/28/72, Page 9-9, states: "The ring header, which is shared by Units 1 and 2, is automatically divided into two headers and non-essential loads are isolated by a safety injection signal." Apparently, based on the above FSAR wording from Amendment 12, the AEC was acknowledging that the non essential loads were isolated by a SI signal. Again, this would not affect the isolation (or lack thereof) during a seismic event.

AEC Letter to NSP, dated February 27, 1974 (attached), states: "We understand from our discussion of February 15, 1974 that the safety injection signal will not isolate non-essential loads, as described in the Final Safety Analysis Report (FSAR) and that the FSAR will be changed to reflect the actual design." This letter acknowledges that non essential loads are not automatically isolated by a SI signal and asked that the FSAR be revised to reflect this configuration. The letter did not request justification for why the configuration deviating from the FSAR was acceptable or request that the system configuration be changed. This implies that the configuration was found to be acceptable; that is, only the documentation needed to be updated.

FSAR, Amendment 37 (dated 3/29/74), made the change to the FSAR requested in the 2/27/74 AEC Letter by removing the previously quoted paragraph.

FSAR, Appendix B, Table B.2-1 states that the Cooling Water System is Class I "Up to Class I Isolation Valves" and Class III for "All that is not Class I". Class I systems are designed for DBE loads and Class III have no seismic design. FSAR, Figures 9.6-3 and 9.6-4, clearly show the safety related/non-safety related boundaries, several of which occur at normally open manual valves. This is similar to the information in the current USAR (Table 12.2-1 and Figures 10.4-1 and 10.4-2).

Two conclusions can be made from the above review:

- The information in the SAR describes that the non safety related lines are not automatically isolated from the safety related lines. This description is in the Tables and Figures and is not contradicted by the text.

- This configuration was reviewed and found to be acceptable by the AEC. This is evidenced by the February 27, 1974 letter and that only the documentation needed to be updated to be consistent with the actual configuration.

Based on this information, it is reasonable to conclude that the system configuration (including the safety related to non safety related boundary interfaces) was reviewed and determined to be acceptable.

4. Plant Position

The plant's position is that the non-seismic cooling water piping will not catastrophically fail (pipe severance) during a seismic event. To the contrary, there is reasonable assurance that the piping will maintain it's pressure boundary integrity during a seismic event. This position is based on:

- Industry Experience

  Per NUREG 1061, Volume 2, Addendum, it is evident that above ground welded power plant piping does not fail due to inertial loads in a seismic event. This is based on evidence collected from several facilities that have experienced seismic events with accelerations significantly above that used for the design of Prairie Island. In most cases, the piping that experienced the strong motion earthquake excitations was designed to minimal or to no seismic criteria. The NUREG concludes that failure of piping is caused primarily by local conditions of weakness in the piping systems rather than global conditions of piping design or installation. Such weaknesses are identified as follows:

- Low piping flexibility in regions of large displacement demands.
- Low piping ductility
- Threaded pipe joints
- Corrosion or erosion
- Poor welding

Similar evidence is also collected in EPRI NP-5617. This report states that from the documented observations that welded piping systems are not susceptible to their own seismic inertial loads and that the seismic design should concentrate on the areas which have proven to be critical during past earthquakes; i.e.,

- Seismic anchor movement
- Interaction
- Corrosion

This information from industry experience indicates that potential vulnerabilities in non seismic piping can be identified and resolved through system inspections and specific component analyses in lieu of dynamic seismic analyses of the piping network.

This is also consistent with the basis for excluding piping from the SQUG reviews. The first paragraph of GL87-02 required that utilities verify the seismic adequacy of their equipment against SQUG criteria which were not available at the time the plants were licensed. Two documents were referenced as forming the basis for this requirement; NUREG-1211, Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants, and NUREG-1030, Seismic Qualification of Equipment in Operating Nuclear Power Plants. In defining the scope of the required seismic adequacy verification effort, GL87-02 stated:

> "The equipment to be included is generally linked to active mechanical and electrical components and cable trays. Piping, tanks, and heat exchangers are not included except those tanks and heat exchangers that are required to achieve and maintain safe shutdown must be reviewed for adequate anchorage."

The explanation for excluding piping and piping supports appears on page 5 of NUREG-1211:

"Experience data collected by SQUG and others and high-level seismic tests on piping conducted in foreign countries and in the U.S. show that piping is not susceptible to failure resulting from seismic inertia loads. The only observed instances of piping failure during the SQUG program to collect seismic experience data were due to relative movement of anchor points and inadequate or nonexistent anchorage of tanks or equipment for sites with zero period acceleration between 0.25 and 0.6g.

"In general, piping is found to have a high margin of safety for almost all the piping if only seismically induced inertia loads are considered. High stresses arise when piping runs through walls or is attached to a large vessel resulting in relative displacements. In piping design, seismic stresses are usually held to a small percentage (say 15%) of the overall allowable stress. In addition, seismic risk studies completed to date show that piping is not predicted to fail even at levels two to five times the SSE level."

- Auxiliary Building, Screenhouse and Turbine Building Qualifications

  The subject piping is routed through various areas of the plant; specifically, the Auxiliary Building, the Screenhouse and the Turbine Building. The Auxiliary Building is safety related and designed for DBE loads. Portions of the Screenhouse are safety related and portions are non safety related. Both the safety related and non safety related structural portions of the Screenhouse are designed for DBE loads to prevent adversely affecting the safety related portion of the building during a seismic event. Most of the Turbine Building is a non safety related structure. A portion of the building (referred to as the Class I Aisle) is safety related. These non safety related CL lines are routed through non safety related areas of the Turbine Building. However, these areas of the Turbine Building were designed to withstand a DBE to prevent it's failure in a seismic event from adversely affecting the Class I Aisle. Therefore, these portions of the building are designed to not collapse and damage the non safety related piping.

Although the piping is expected to maintain it's pressure boundary integrity during a seismic event, to be conservative, the hydraulic analysis includes postulated affects of a seismic event on the cooling water system by assuming a crack in each non safety related pipe off of the safety related supply headers. The CL system is a moderate energy system. For consistency, the size of each crack was determined using the method in BTP MEB 3-1, Section B.3.c. The results from this hydraulic modeling indicate that the system is capable of meeting it's design functions during this event.

5. Additional Considerations

   a. In response to the Unresolved Issue from the Inspection Report, the plant initiated a Condition Report to perform a review of the original position. The following steps specifically have been taken as part of this review:

      (1) A review of the cooling water system piping stress analyses was completed to determine which piping lines have been seismically analyzed. The design requirement for analyzing non Class I piping connected to Class I piping is as follows:

      > "Effect of lower design class piping connected to Design Class I piping shall be accounted for by including the lower design class piping run up to the first anchor point into the analysis of the Design Class I piping."

      The review of the stress analyses indicated that this criteria was followed. In some piping runs, the anchor point is the non safety related heat exchanger, and thus, the entire run was analyzed. In other cases, an anchor is provided on the piping run and the piping past the anchor was not included in the analyses.

      (2) An independent seismic expert was hired to perform a walkdown of the non safety related portions of the system which have not been seismically analyzed. The purpose of this walkdown was to identify any system vulnerabilities to a seismic event and implicitly judge the validity of this plant position. For the most part the conclusions from this walkdown was that the plant's position that the piping would maintain it's pressure boundary integrity was valid. However, the walkdown did identify selected vulnerabilities in the system. An initial evaluation assuming complete pipe breaks at these vulnerabilities concluded that the system would still be able to accomplish it's functions for the units to maintain safe shutdown in a seismic event.

   b. There is procedural guidance that specifically directs operators to reduce CL System demand in the event that the flow demand is greater than the limit for continuous operation (17,500 gpm). In the event of a reactor trip, the first procedure entered is E-0, Reactor Trip or Safety Injection. Without a SI signal, the operators transition to ES-0.1, Reactor Trip Recovery, in the fourth step of E-0. In the fifth step of ES-0.1, the operators check the status of the CL System. If the conditions are not normal (i.e., low header pressure), the operator is directed to reduce CL System demand per the applicable operating procedure. In addition, there is a low header pressure alarm on the Control Board for which the associated alarm response procedure directs the operator to the same procedure to reduce the demand on the system.

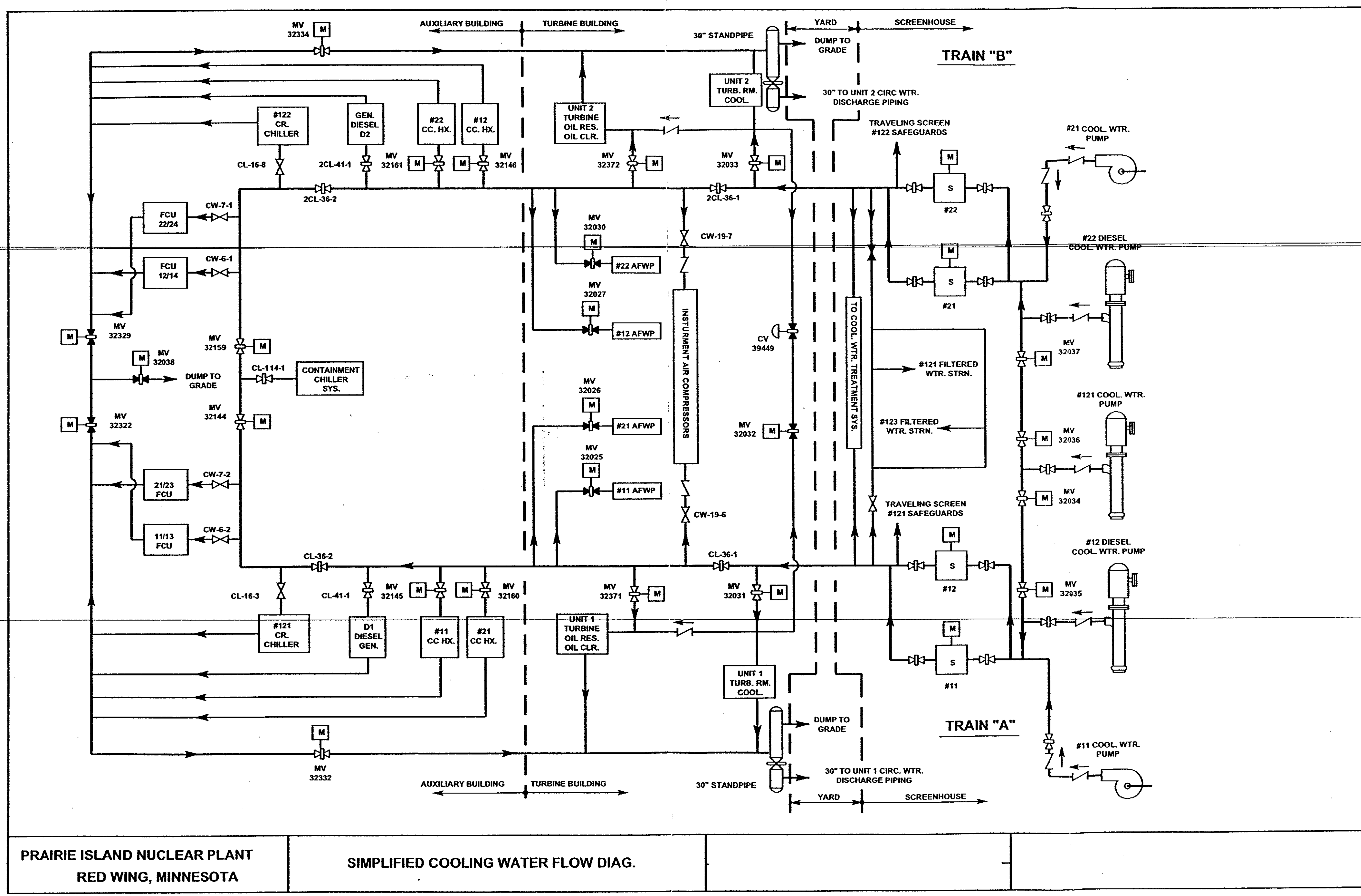
6. Conclusion

In conclusion, the plant believes that the current configuration is consistent with the original design and licensing basis as understood by the regulatory agency at the time. Based on industry experience and the actual robustness built into the system, there is reasonable assurance that the piping will maintain its pressure boundary integrity during a seismic event. However, to be conservative, the plant assumed a single crack in each non safety related line off of the supply header in the hydraulic modeling of the system. The results from this hydraulic modeling indicate that the system is capable of meeting it's design functions during this event.

Attachments

1. Figure 1 - Simplified Cooling Water System Flow
2. USAR Figure 10.4-1A - Flow Diagram Unit 1 & 2 Cooling Water - Screenhouse
3. USAR Figure 10.4-1B - Flow Diagram Unit 1 Cooling Water - Turbine Bldg.
4. USAR Figure 10.4-1C - Flow Diagram Unit 1 Cooling Water - Aux. Bldg.
5. USAR Figure 10.4-1D - Flow Diagram Unit 1 Cooling Water - Containment
6. USAR Figure 10.4-2A - Flow Diagram Unit 2 Cooling Water - Turbine Bldg.
7. USAR Figure 10.4-2B - Flow Diagram Unit 2 Cooling Water - Aux. Bldg.
8. USAR Figure 10.4-2C - Flow Diagram Unit 2 Cooling Water - Containment
9. Letter from NRC to NSP, dated February 27, 1974

Figure 1 Simplified Cooling Water System Flow



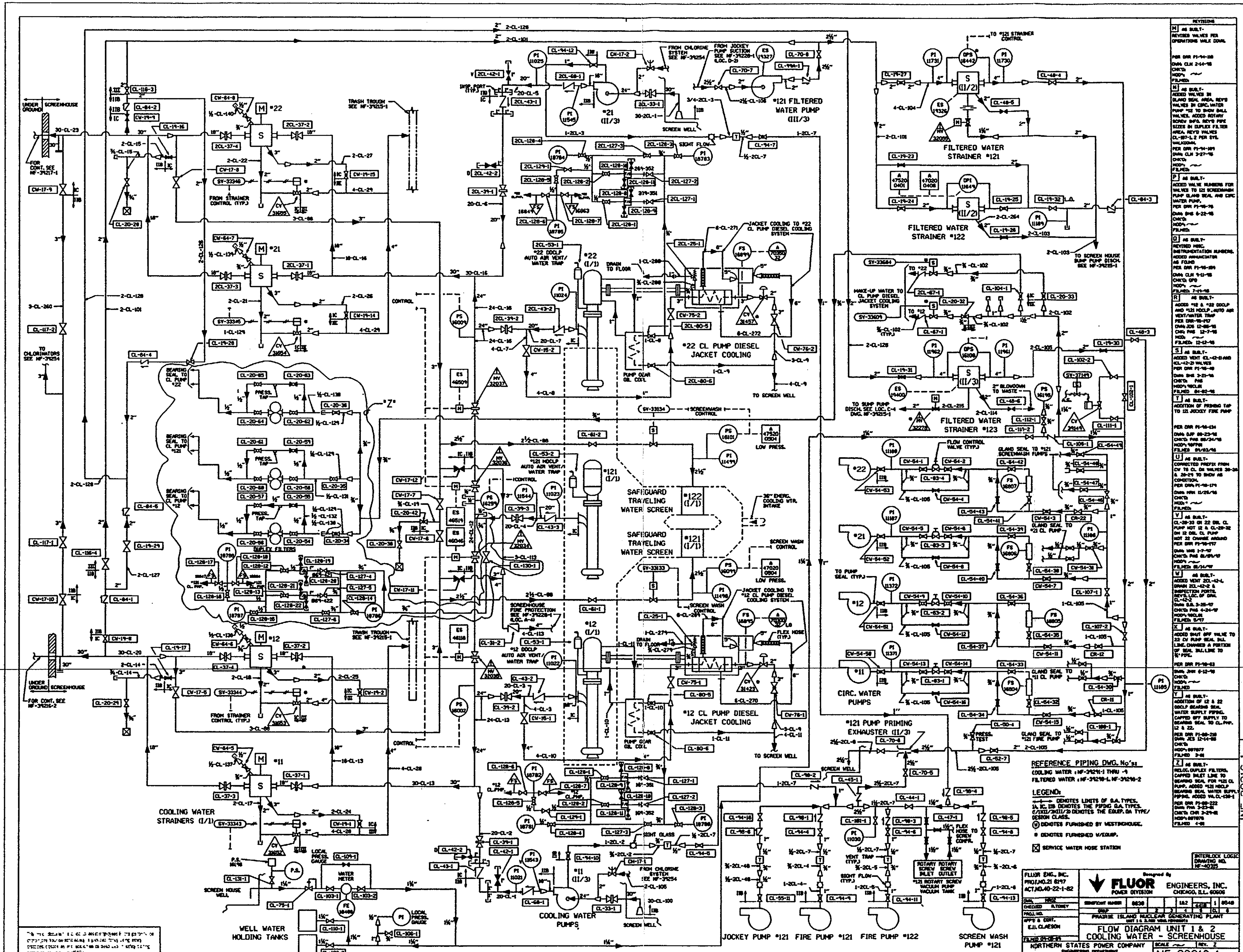


FIGURE 10.4-1A REV 23

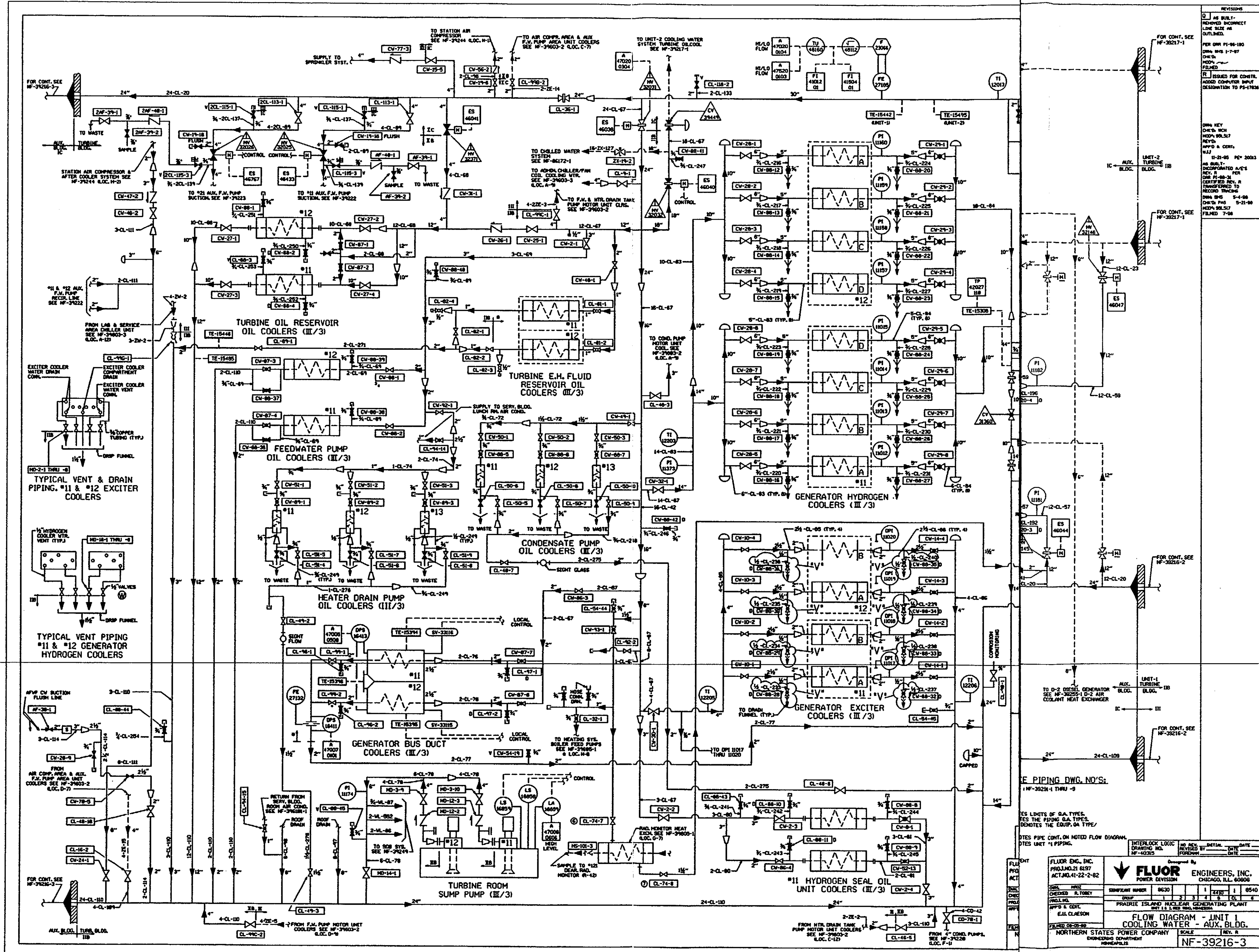


FIGURE 10.4-1C REV. 18

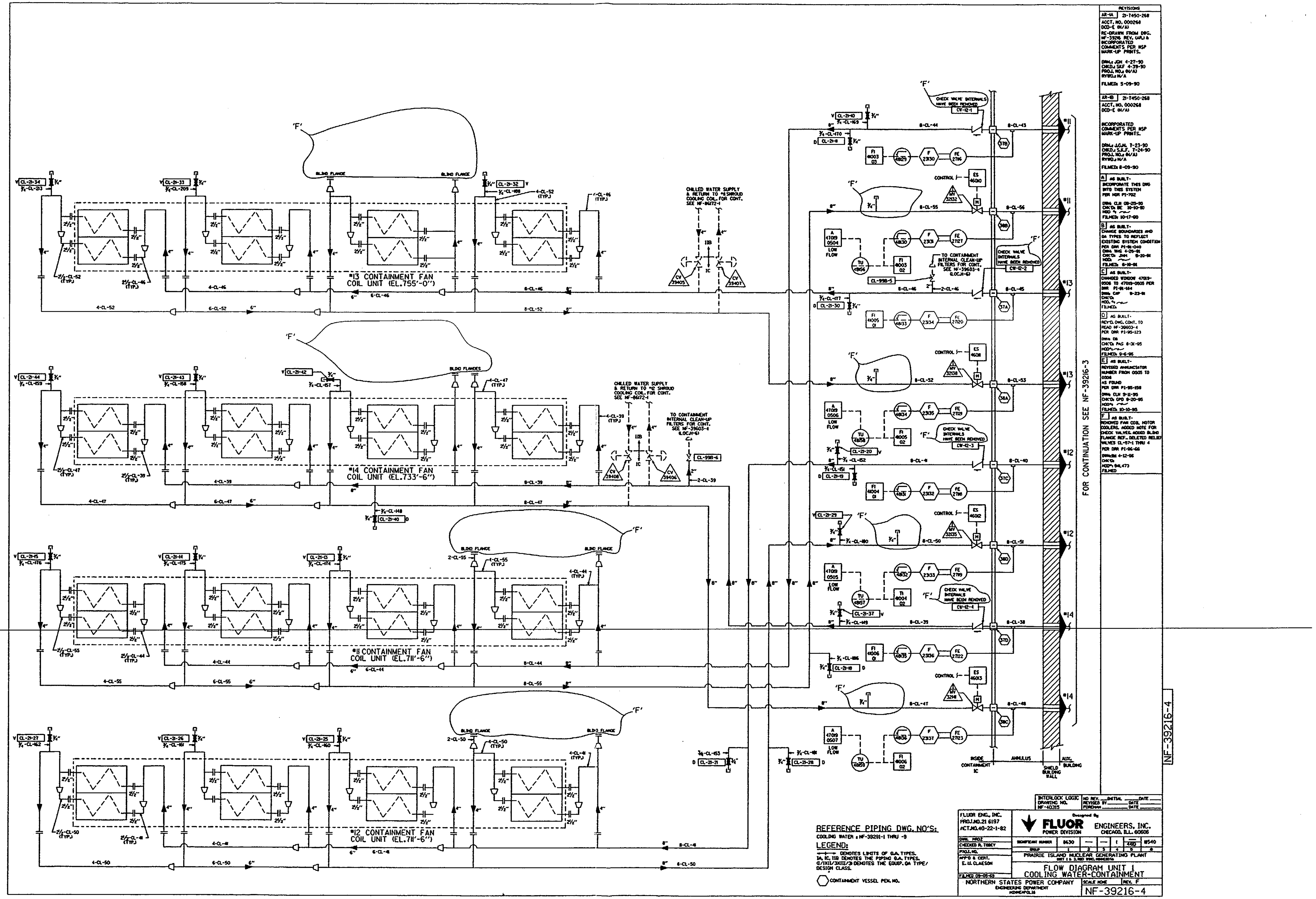
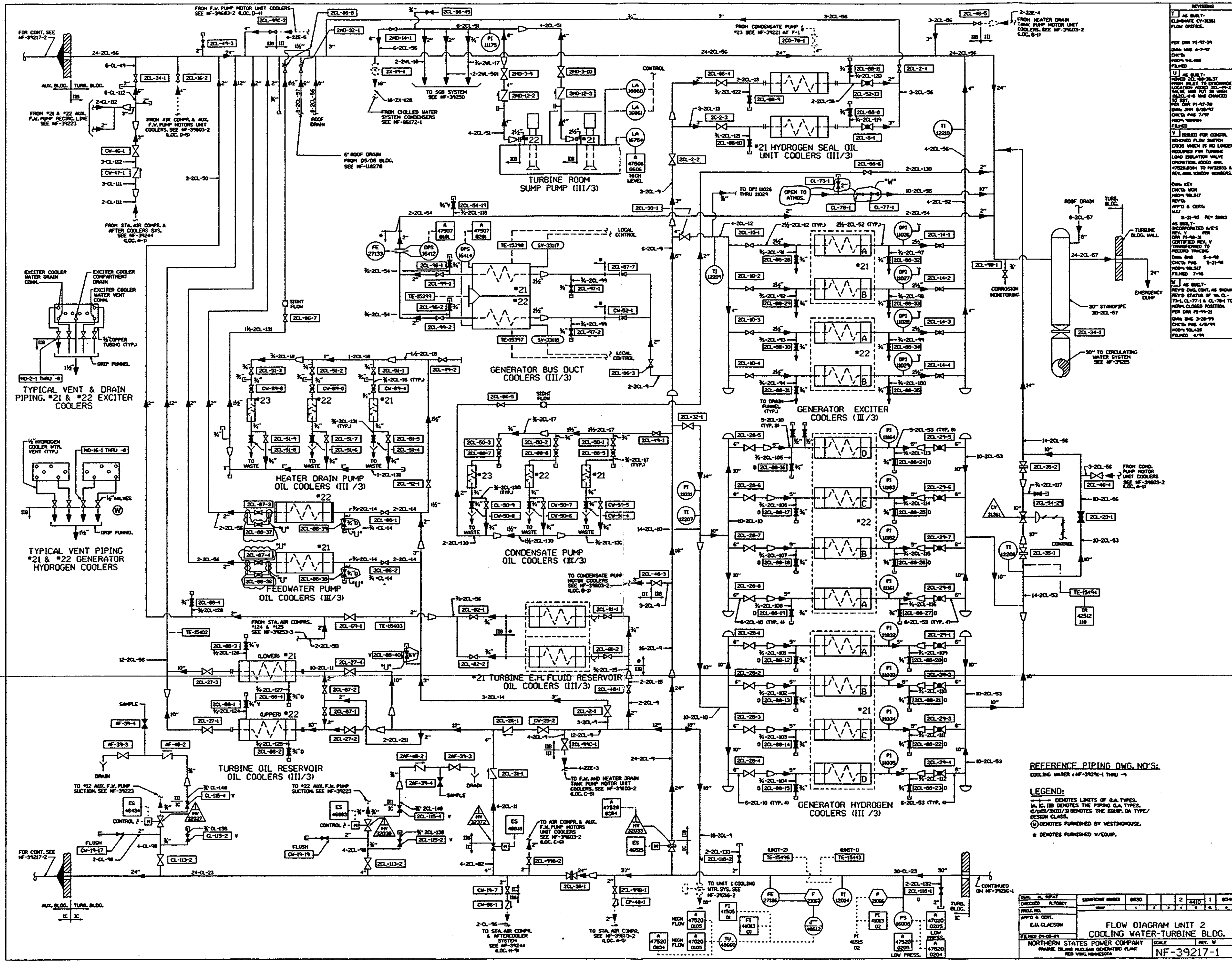


FIGURE 10.4-1D REV. 18



FLOW DIAGRAM UNIT 2
COOLING WATER-TURBINE BLDG.
NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA
NF-39217-1

FIGURE 10.4-2A REV. 20

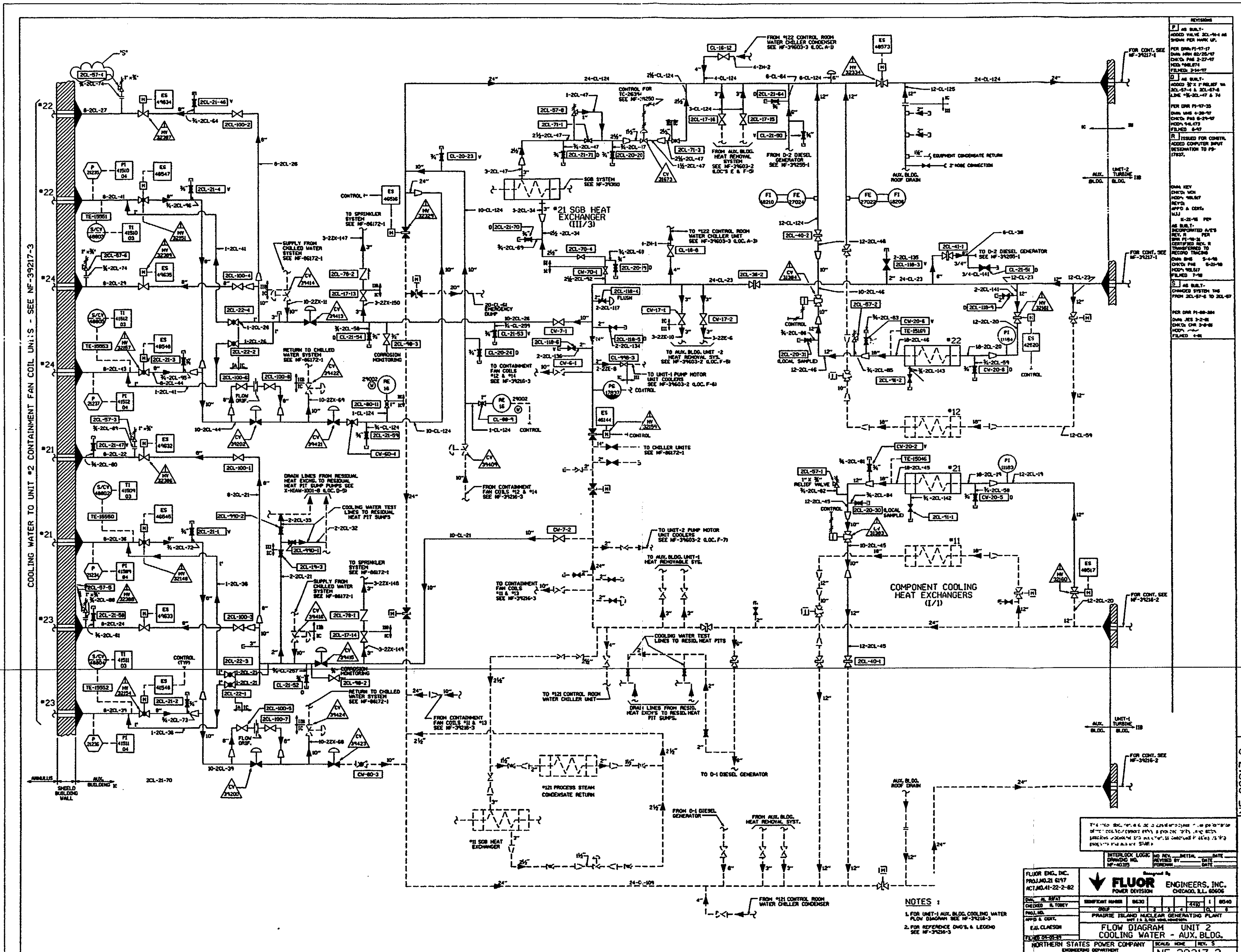


FIGURE 10.4-2B REV 23

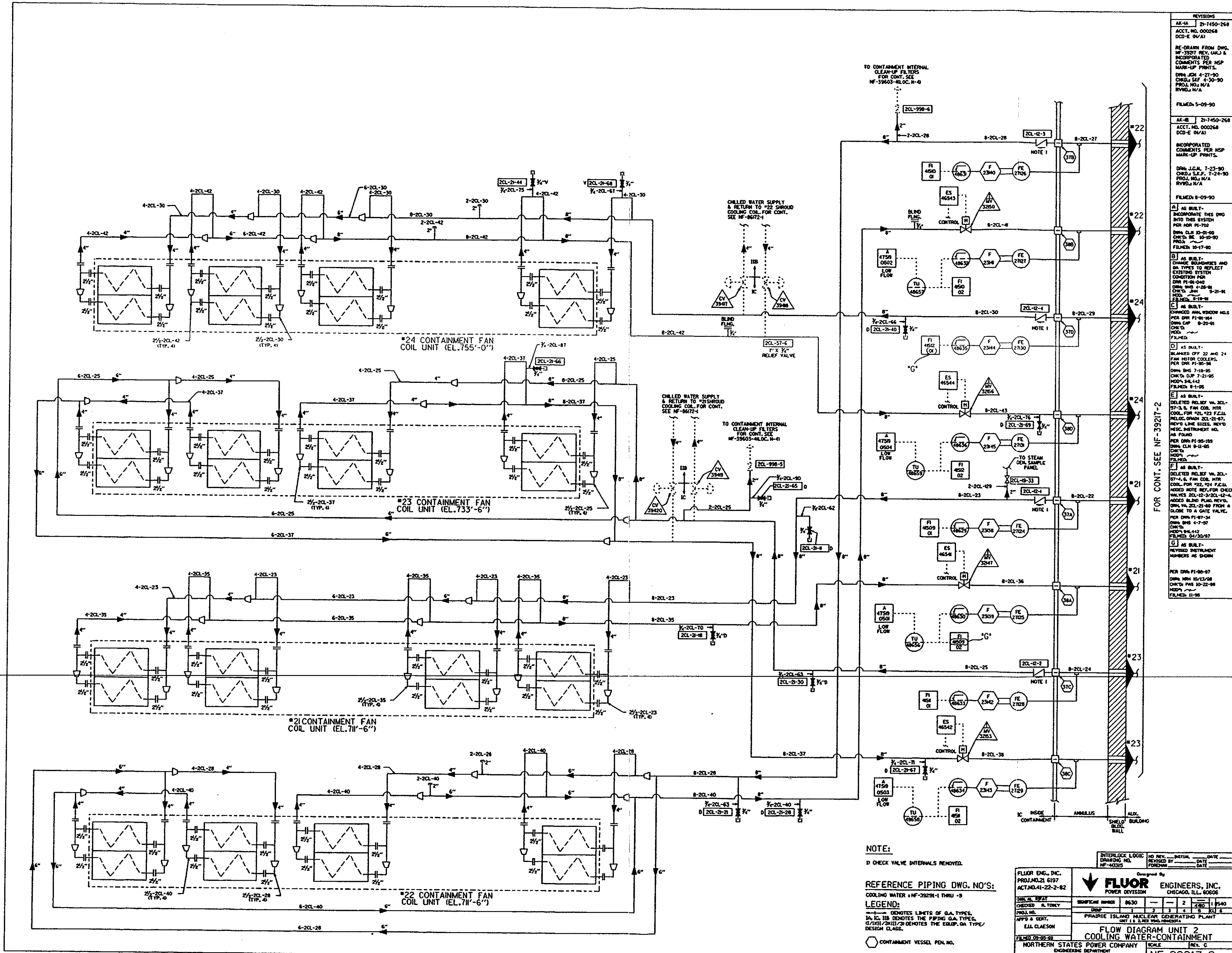


FIGURE 10.4-2C REV. 18

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

NUCLEAR SUPPORT SERVICES DEPT.
L. O. MAYER
D. W. JAMES
D. M. MUSOLF
L. L. TAYLOR
M. H. VOTH
NSS FILE

February 27, 1974

Docket Nos. 50-282
50-306

Mr. L. O. Mayer
Director of Nuclear Support Services
Northern States Power Company
414 Nicollet Avenue
Minneapolis, Minnesota 53401

Dear Mr. Mayer:

We have reviewed the results of preoperational tests performed on the diesel generators and the diesel-driven emergency cooling water pumps in the Prairie Island Nuclear Generating Plant. The identification of these test reports and our evaluation of these test results are enclosed. The test results and our evaluation of them were discussed in a meeting with you and other representatives of Northern States Power Company on February 15, 1974. The purpose of this letter is to request that information in addition to that contained in the test reports be included in the final reports to the Directorate of Licensing in response to our enclosed evaluation and our discussions with you on February 15, 1974.

The following additional information should be provided.

A. Diesel Generator Qualifications Tests

1. Provide an evaluation of the range of voltages expected at the 4160-volt buses for expected auxiliary loads applied to the transformers supplying the buses. Discuss the relative time intervals that various voltage levels will occur at the bus during a typical plant fuel cycle. We understand from our meeting of February 15, 1974 that most of the time during plant operation these buses will have a voltage near the nominal voltage (4160 volts) rather than near the upper limit (4500 volts) as had been previously understood for our enclosed evaluation.

2. Provide the rated voltage for the diesel generators and for motors that operate engineered safety features. Discuss the potential for damage if this equipment is operated continually at the maximum or minimum voltages and frequencies within the expected operating range. Provide the bases that justify your estimate of potential damage.

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3. Provide the data and evaluation of recent tests of a diesel generator in which two safety injection pumps were started simultaneously with the generator voltage regulator setpoint at 4160 volts. We understand from our February 15, 1974 discussions that the generator voltage regulator setpoint for the tests referenced in our enclosed evaluation was 4320 volts and that Unit 1 will be operated with the regulator setpoint near that value until Unit 2 safety injection tests are run.

B. Diesel Driven Cooling Water-Pump System Qualification Tests

1. Include data from other startup tests that have demonstrated that a safety injection signal will automatically start the diesel-driven pumps and close valves to divide the ring header into two headers. We understand from our discussions of February 15, 1974 that the safety injection signal will not isolate non-essential loads, as described in the Final Safety Analysis Report (FSAR) and that the FSAR will be changed to reflect the actual design.

2. Provide an evaluation of the reduction in non-essential cooling water flow and the adequacy of the cooling water flow to diesel generators following a loss of offsite power with both units operating and assuming a single failure, based on data obtained during Unit 1 100% power tests. We have concluded that the test referenced in our enclosed evaluation demonstrates that the flow distribution to the equipment served by the cooling water system meets that required by the FSAR for Unit 1 operation, based on a review by the Directorate of Regulatory Operations (RO Inspection Report No. 050-282/73-35).

The final report on diesel generator tests should be submitted to the Directorate of Licensing by March 15, 1974 in accordance with Technical Specification 6.7.B.3 Item 7. The final report on diesel driven cooling water pump tests should be submitted within 3 months after completion of the Unit 1 startup tests at 100% power operation in accordance with Technical Specification 6.7.B.3 Item 8.

Sincerely,

Karl Kniel

Karl Kniel, Chief
Light Water Reactors Branch 2-2
Directorate of Licensing

Enclosure:
As stated

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PRAIRIE ISLAND UNIT 1

Evaluation of Startup Test Results

A. Diesel Generator Qualification

Northern States Power has completed a test program consisting of twenty start and complete sequence loading of all safety loads for each onsite diesel generator. The results are contained in Addendum A to P27.4.3 test report "Plant Response to Safeguards With Concurrent Station Blackout (Diesel Loading) - Emergency Diesel Generator Response." This test program was performed to assure that although the recommendations of Safety Guide 9 were not fulfilled, the diesel generators are capable of satisfying their safety functions reliability.

Our review of the results of the test program indicates that the testing performed to date may not be acceptable because the voltage and frequency levels were not set at the rated values stated in the FSAR. The steady state voltages that were recorded for diesel generator (DG) were between 4216 and 4699 volts and for DG 2 were between 4216 and 4345 volts. Likewise, it was indicated that the steady state frequency for run #2 was between 59 and 60 Hz for DG 1 and between 57.7 and 58.85 for DG 2. No other runs included frequency in the test results. Run #2 on DG 2 indicated a drop in frequency to 55.5 H_z and increase in frequency to 63.5 H_z when initial load was placed on the DG. In discussion with the applicant, it was indicated that these DG's would be operated throughout life at voltage levels between 4400 and 4500 volts.

The design ratings of the DG as indicated in the FSAR, are 4160 volts and 60 H_z. All of the safety related motors in the plant have the same design ratings.

Since we have no assurance that the design of this equipment (DG and loads) have been qualified at the operational voltage and frequency levels indicated by the applicant, we recommend that the applicant re-run the twenty tests on each DG using the design ratings established by the manufacturer or provide assurance that the reliability of the DG and loads operating at these unusual voltage and frequency levels is equal to that when operating at the rated levels stated in the FSAR. This information should be provided for our evaluation.

B. Diesel-Driven Cooling Water Pump System Qualification

Northern States has completed cooling water system tests. The results of which are contained in "Prairie Island Nuclear Generating Plant Unit 1, Operating Test Procedure Number 16, Cooling Water System." The purpose of this test was to confirm that the cooling water system would supply cooling water to the components indicated in the FSAR system description and flow diagrams, and to verify the system control, interlock and alarm functions.

We have determined from our review of the results of this test that the design as identified on Pages 9-8 and 9 of the safety evaluation report, has not been confirmed with regard to the following:

a) Automatic start of the diesel-driven cooling pumps by a safety injection signal.

b) Automatically dividing the ring header into two headers and isolating non-essential loads on a safety injection signal.

c) Automatically reducing cooling water flow in non-essential systems from low pressure in the discharge header.

We have concluded that all other aspects of the instrumentation and electrical equipment performed in accordance with the design requirements during this test. Additionally, we determined that the diesel driven pumps 300 start tests successfully completed without failure. We have not reviewed the adequacy of the rate of flow of water through each equipment since this is outside our scope of responsibility.

We understand that the above deficiencies will be included in other test programs and the test results should be provided for our evaluation.