NSP

Northern States Power Company

1717 Wakonade Dr. E.
Welch, MN 55089
Telephone

January 29, 1997

10 CFR Part 50
Section 50.90

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

License Amendment Request Dated January 29, 1997
Amendment of Cooling Water System Emergency Intake Design Bases

Reference 1: NRC letter dated November 1, 1996 to Mr. E. Watzl, Vice President, Nuclear Generation, from Geoffrey E. Grant, Director, Division of Reactor Safety, entitled, "NRC SPECIAL INSPECTION REPORTS NO. 50-282/96015(DRS); 50-306/96015(DRS)."

Reference 2: NRC letter dated January 23, 1997 to Mr. E. Watzl, Vice President, Nuclear Generation, from A. Bill Beach, Regional Administrator, entitled, "NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $50,000 (NRC Special Inspection Report Nos. 50-282/96015; 50-306/96015)."

Attached is a request for changes to the Bases for the Technical Specifications, Appendix A of the Operating Licenses, and to the licensing basis for the Operating Licenses, for the Prairie Island Nuclear Generating Plant. This request is submitted in accordance with the provisions of 10 CFR Part 50, Sections 50.59 and 50.90 to address an unreviewed safety question relating to the Cooling Water System emergency intake line flow capacity.

In 1995, Prairie Island performed a self-assessment Service Water System Operational Performance Inspection (SWSOPI) and from that review determined

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that the Cooling Water System emergency intake line flow capacity could be less than the design assumptions following a single failure of man-made structures as a result of a seismic event. Plant tests performed in 1995 confirmed that the emergency intake line flow capacity is less than the original design value. In response to the 1995 test results, the plant staff determined complete dependence on the emergency intake line following a seismic event would occur after approximately three and one-half hours and thus sufficient time would be available for plant operators to manage cooling water system flow requirements. This evaluation considered Regulatory Guide 1.27 which does not require immediate loss of normal intake flow paths following a single failure of man-made structures, but rather allows for consideration of time-related effects for normal flow path obstruction. Accordingly the plant staff took remedial actions by implementing procedures for managing cooling water system loads following a seismic event.

By letter dated November 1, 1996 (Reference 1), "NRC determined that taking credit for the non-seismic intake canal and operator actions following an earthquake involved an unreviewed safety question." This submittal provides the basis for the plant remedial actions and requests NRC staff review and approval of this unreviewed safety question. By letter dated January 23, 1997 (Reference 2) the NRC has determined that a violation of NRC requirements occurred.

Exhibit A contains a description of the proposed changes, the reasons for requesting the changes, and the supporting safety evaluation and significant hazards determination. Exhibit B contains current Prairie Island Technical Specification Bases pages marked up to show the proposed changes. Exhibit C contains the revised Prairie Island Technical Specification Bases pages incorporating the proposed changes. Exhibit D contains current Prairie Island Updated Safety Analysis Report pages marked up to show the proposed changes. Exhibit E contains the revised Prairie Island Updated Safety Analysis Report pages incorporating the proposed changes. Exhibit F presents excerpts from the report on the evaluation of the intake canal response to a seismic event.

The NRC staff notified NSP by telephone calls on January 8 and 14, 1997 that this unreviewed safety question is a restart issue for Prairie Island when either unit shuts down. Since Prairie Island Unit 2 shutdown for refueling on January 25, 1997 and is scheduled to restart on March 5, 1997, NSP respectfully requests review and approval of this submittal under the rules of 10 CFR Part 50, Section 50.91(a)(6) where exigent circumstances exist.

Without review and approval of this license amendment request by the end of the Unit 2 outage, Prairie Island would be prevented from resumption of plant operation at a cost in excess of $135,000 per day for replacement power.

NSP has used its best efforts to make a timely application for this amendment and has not intended to create an exigency to take advantage of the procedure for exigent handling.

This issue was first raised by NRC inspections from November 21, 1995 to January 2, 1996 as a possible unreviewed safety question. NRC letter dated November 1, 1996 (Reference 1) stated that a determination had been made that an unreviewed safety question existed, and the letter went on to establish a predecisional enforcement conference to determine if violations were involved. The NRC Inspection Report attached to the letter indicated the staff needed additional information for the staff to make a definitive assessment of the licensee's approach of relying on operator intervention as an alternative and permanent modification. Thus, the majority of the predecisional enforcement conference on November 22, 1996 was focused on the unreviewed safety question and the effects of reliance upon operator actions for effective performance of systems important to safety. The sense left with NSP management was that further guidance would be forthcoming from the NRC on the scope of the unreviewed safety question. It was not until the January 14 telephone call with the NRC that it was clear no further definition of the unreviewed safety question would be provided. Furthermore, NSP was unaware until the January 8 telephone call with the NRC that resolution of the unreviewed safety question would be required prior to plant restart.

Thus, NSP has responded expeditiously to present this application for amendment at this date. Accordingly NSP requests this submittal be treated as an exigency in accordance with 10 CFR Part 50, Section 50.91(a)(6).

If you have any questions related to this License Amendment Request please contact Dale Vincent at 612-388-6758.

Michael D Wadley

Michael D. Wadley
Plant Manager
Prairie Island Nuclear Generating Plant

attachments: 7

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
State of Minnesota
Attn: Kris Sanda
J E Silberg

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT DOCKET Nos. 50-282
50-306

REQUEST FOR AMENDMENT TO
OPERATING LICENSES DPR-42 & DPR-60

LICENSE AMENDMENT REQUEST DATED January 29, 1997
Amendment of Cooling Water System Emergency Intake Design Bases

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Bases for the Prairie Island Operating License, Appendix A and the plant design bases as shown in the attachments labeled Exhibits A, B, C, D, E, and F. Exhibit A describes the proposed changes, reasons for the changes, and the supporting safety evaluation and significant hazards determination. Exhibit B contains current Prairie Island Technical Specification Bases pages marked up to show the proposed changes. Exhibit C contains the revised Technical Specification Bases pages incorporating the proposed changes. Exhibit D contains current Prairie Island Updated Safety Analysis Report pages marked up to show the design bases changes. Exhibit E contains the revised Updated Safety Analysis Report pages incorporating the proposed changes. Exhibit F contains excerpts from the report on the evaluation of the intake canal response to a seismic event.

This letter and its attachments contain no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Michael D. Wadley
Michael D. Wadley
Plant Manager
Prairie Island Nuclear Generating Plant

On this 29th day of January 1997 before me a notary public in and for said County, personally appeared, Michael D. Wadley, Plant Manager, Prairie Island Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Marcia K LaCore

MARCIA K. LaCORE
NOTARY PUBLIC-MINNESOTA
HENNEPIN COUNTY
My Commission Expires Jan. 31, 2000

LICENSE AMENDMENT REQUEST DATED January 29, 1997

Amendment of Cooling Water System Emergency Intake Design Bases

EXHIBIT A

Description of the Proposed Changes, The Reasons for Requesting the Changes, and the Supporting Safety Evaluation/Significant Hazards Determination

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to the licensing bases for the Facility Operating Licenses and Bases for Appendix A, Technical Specifications:

BACKGROUND

This License Amendment Request revises Prairie Island design bases relating to the cooling water system emergency intake. The Prairie Island "Cooling Water System" is similar to the system commonly known in the nuclear industry as the "Service Water System." During the preparation and self assessment activities for the Prairie Island Service Water System Operational Performance Inspection (SWSOPI), an issue was identified by plant personnel concerning the capacity of the cooling water emergency intake line. The emergency intake line pre-operational flow test was reviewed and the results were compared to original design calculations. The results of the pre-operational flow test were less than the design value stated in the Updated Safety Analysis Report (USAR). It was decided that the pre-operational flow test should be repeated. The results of this most recent test, performed in November 1995, indicated that there had been some change in flow capacity since the pre-operational test. A safety evaluation was prepared to evaluate the effects of emergency intake line flow capacity less than the design value.

The NRC subsequently reviewed the safety evaluation and

> . . . concluded that the licensee's overall approach for the justification for continued operation was consistent with the guidance in Regulatory Guide 1.27. The staff considered that it was acceptable for the licensee to take credit for the non-seismic intake canal not becoming instantaneously blocked as a result of an earthquake. Likewise, it was also acceptable for the licensee to assume a time lag between when the downstream lock and dam fails and when the river reaches its minimum design-basis level assumed in plant analyses. This time lag was supported by valid physical assumptions that would occur following dam failure. Thus, allowing time for operator action to reduce cooling water system flow by isolating nonessential loads appeared to be acceptable, providing the operators could accomplish necessary actions in the time available. (Ref. 1)

However, the NRC concluded that taking credit for the non-seismic intake canal and operator actions following an earthquake involved an unreviewed safety question (References 1 and 2).

Cooling Water System Background

The cooling water system includes five pumps, all supplying a common discharge header (Figure 1). Two horizontal motor driven pumps, supplied by non-safeguards power supplies, are normally operating. Three vertical pumps are standby safeguards pumps. The motor driven vertical pump is powered by a safeguards bus which is supplied by either of two Unit 2 emergency diesel generators. The other two vertical pumps are diesel driven. The common discharge header supplies two main supply headers. One header supplies Unit 1 turbine building and both Unit 1 and Unit 2 Train A safeguards components. The other header supplies Unit 2 turbine building and both Unit 1 and Unit 2 Train B safeguards components. Lines branch off of the main supply headers to supply both non-safeguards and safeguards loads.

The supply to the cooling water pumps is normally from the intake canal (Figure 2) Water flows from the Mississippi River, through the Intake Screenhouse, through the intake canal, and into the intake bay in the Plant Screenhouse. The non-safeguards horizontal pumps take suction directly from the intake bay within the Plant Screenhouse which receives water from the intake canal. The supply to the safeguards vertical pumps continues through the intake bay in the Plant Screenhouse through either of two normally open sluice gates into the Class I safeguards emergency cooling water bay (Figure 3). The sluice gates are non-safety related equipment which have been evaluated and qualified to seismic criteria. An alternate supply through the emergency intake line, directly from the river, also supplies the emergency cooling water bay. A

concrete crib in the river channel protects the pipe inlet. The 36" steel pipe is buried beneath the river bed in non-liquefiable fill. The pipe runs under the intake bay and into the emergency cooling water bay.

Seismic Event Background

In the case of a seismic event, no credit is taken for structures or equipment which are not seismically qualified or evaluated. Off-site power is assumed to be lost since the switchyard is not seismically qualified. The horizontal pumps trip due to loss of their non-safeguards power supply and all three vertical pumps start on low discharge header pressure (start of the 121 motor driven pump will be delayed by either Unit 2 safeguards bus load sequencer). The 121 motor driven vertical safeguards pump is either blocked from starting or tripped by an interlock when both diesel driven pumps reach a predetermined speed. Fail-open air operated valves in the cooling water system are assumed to fail open since the instrument air system is not seismically qualified. This places maximum demand on the cooling water pumps.

Following a design basis seismic event the only assumed source of cooling water within the circulating water system is the intake canal which supplies the Plant Screenhouse intake bay. This means no credit is taken for make up flow from the river through the Intake Screenhouse and no credit is taken for the volume of water in the recycle canal. Since no makeup flow from the river through the intake screenhouse into the intake canal has been assumed, the converse is also true in that no out flow of intake canal water to the river is assumed. This is a conservative assumption.

If a flow path remained between the river and the intake canal, then the plant would continue to receive sufficient water from the river to meet the cooling water system demand. An estimate by the Army Corps of Engineers indicates that it will take roughly 50 hours for the volume of water upstream of the plant to flow downstream. Only then can the river level drop to the assumed minimum.

As the event proceeds, the cooling water pumps' flow rate is determined by the cooling water system demand. The water inventory in the emergency cooling water bay is made up from the intake bay through the sluice gates. Since no credit is taken for make up to the intake canal, that volume will eventually be depleted. Following depletion of the inventory in the intake canal, the only supply to the emergency cooling water bay will be the emergency intake line. Water flows through the emergency intake line as a result of the height difference between the river and the emergency cooling water bay (Figure 4). Thus, the level in the river determines the available flow rate through the emergency intake line. The cooling water system demand must be reduced to less than

the emergency intake line capacity to ensure adequate NPSH for the cooling water pumps.

PROPOSED CHANGES AND REASONS FOR CHANGES

The proposed changes to the licensing bases for the Prairie Island Operating License and changes to the Bases for Appendix A, Technical Specifications, are described below, and the specific wording changes are shown in Exhibits B, C, D, and E.

USAR Section 10.4.1.2.2 is changed to reflect the use of operator action to control cooling water system flow demand following a seismic event. The USAR is also changed to incorporate the results of the intake canal stability analysis. These changes are necessary to take credit in the Prairie Island licensing basis for the intake canal and operator actions following an earthquake. USAR Table 12.2-1 is revised to show the reclassification of the intake canal.

The bases for Technical Specification 3.3.D is changed to clarify the purpose of the emergency intake line. These changes are necessary for the Technical Specification Bases to agree with the event analysis assumptions.

SAFETY EVALUATION

The NRC (References 1 and 2) concluded that taking credit for the non-seismic intake canal and operator actions following an earthquake involved an unreviewed safety question. NSP has reviewed these issues and concluded the health and safety of the public is protected when credit is taken for the intake canal and operator actions following an earthquake. The bases for this conclusion follows.

Non-Seismic Intake Canal

The bank stability of the intake canal during a seismic event was not originally evaluated. Prairie Island was not specifically committed to Safety Guide 27 (the revision of Regulatory Guide 1.27 which existed when Prairie Island was licensed), but it was used as guidance in the evaluation of the ultimate heat sink. The original analysis conservatively assumed that the intake canal would become completely blocked, but did not specify a time frame in which this would occur. Safety Guide 27 states that it is not necessary to assume the instantaneous collapse of canal walls or dams. Therefore, a time frame was credited for operator action. Subsequently, an intake canal bank slope stability study was performed to determine to what degree, if any, the bank would slough off into the intake canal. The study consisted of soil

borings, laboratory testing of soil samples, and electronic Cone Penetration Tests. A six page excerpt from this study containing the summary of the testing and conclusions is included as Exhibit F. The overall conclusion of this study is that the intake canal walls will not liquefy or lose strength during a seismic event with the intake canal at normal pool level.

Therefore, crediting the stability of the intake canal walls is technically justified and does not result in the possibility of a malfunction of a different type than previously evaluated.

Operator Action

The NRC staff determined that an Unreviewed Safety Question existed

> . . . because the change to the licensee's design basis of requiring operator actions: (1) might increase the probability of a malfunction of equipment important to safety previously evaluated in the FSAR because operator intervention was now being relied upon for effective performance of systems important to safety, (2) might result in the possibility for creating an accident or malfunction of a different type than evaluated previously in the FSAR because making the effective performance of systems important to safety reliant upon human intervention could potentially introduce unanalyzed failure modes caused by operator acts of omission or commission. (Ref. 1)

Each of these questions is evaluated as follows.

(1) might increase the probability of a malfunction . . .

Following a seismic event operator action is required to reduce cooling water system flow demand to less than or equal to the capacity of the emergency intake line. The Seismic Qualification Utility Group (SQUG) Generic Implementation Plan was reviewed for guidance when relying upon operator action to assure safe shutdown following an earthquake. The Plan states, "Where operator actions are relied upon to achieve and maintain safe shutdown, the licensee will ensure that appropriate procedures are available which consider the time within which actions must be taken, and that operators have been trained in the use of these procedures." Through the use of the Prairie Island simulator, NSP has determined that recognition of the event is also important. Each of these four items: recognition, procedures, timeliness, and training is addressed below.

Operator recognition of a seismic event will come from several indications. An earthquake of sufficient magnitude to damage Lock & Dam No.3 and the circulating

water system civil structures would be expected to produce detectable ground motion in the control room and operator notification from outside of the control room. Additionally, there is an annunciator in the control room that is fed from the seismic monitors. This annunciator has a unique horn to set it apart from other control room annunciators.

A plant procedure provides guidance to the operator for reducing cooling water system flow demand to less than the capacity of the emergency intake line. The operator is directed to isolate cooling water loads in a predetermined, logical manner. The cooling water load management actions ensure that those components required for safe shutdown continue to receive sufficient flow while those not required are isolated. All of the operator actions are completed in the control room on the main control boards of the two units. The simulator and operators were involved in the verification of the procedure to ensure completeness, understanding and usability. Copies of this procedure are located in the control room.

The time frame within which the operator actions must be completed is based on the available volume of water in the intake canal. The cooling water system flow demand is conservatively calculated to be 29,750 gpm. Additionally, all fire protection system pumps take suction from the intake bay. All pumps except the diesel driven fire pump are powered by non-safeguards sources, and therefore will not operate. The diesel fire pump and the fire protection system are designed to Class II seismic requirements. Therefore, the diesel fire pump can be expected to operate after a seismic event. Adding in the design flow rate of the diesel fire pump (2000 gpm) to that of the cooling water system gives a total flow rate of 31,750. The volume in the intake canal will provide approximately 25 minutes per foot of water depth. Nominal intake canal level is 8 feet above the minimum river level assumed after a seismic event. This provides approximately 3-1/2 hours of operation at 31,750 gpm before the intake canal can no longer provide flow to the safeguards cooling water pumps. As cooling water system flow demand is decreased, the available time increases. Additional water volume in the recycle canal has not been considered in this calculation. Since there is no physical barrier between the intake canal and the recycle canal this water volume serves as additional margin. Also, no make up flow was credited from the river through the Intake Screenhouse.

Special training on the new procedure was provided to each operating crew prior to assuming shift duties immediately after the procedure was issued. Training on this procedure was included in the normal operator requalification training during 1996. Additional operator training is scheduled for 1997 and training on this procedure will be incorporated into the continuing training for operators.

Thus, on the basis of the above considerations, NSP concluded that requiring operator actions does not significantly increase the probability of a malfunction of equipment important to safety from that previously evaluated.

(2) might result in the possibility for creating . . .

The possibility that an act of omission or commission might create a malfunction of a different type than has been analyzed previously was evaluated. Upon recognition that a design basis earthquake has occurred, the control room operator is required to perform a procedure to reduce cooling water system demand to match the capacity of the emergency intake line. The probability of a design basis earthquake at Prairie Island which would require use of this procedure is extremely low, thus the probability of any subsequent activities are even lower.

Operator acts of omission would mean that the operator was directed to perform a procedural task, but failed to initiate the required task. This procedure involves two types of activities to assure successful management of cooling water loads: monitoring system parameters and isolating cooling water system loads. Failure to perform a step directed by the procedure could result in system demand exceeding the capacity of the emergency intake line.

The procedure to manage cooling water system loads is comparatively simple and the operators are allowed sufficient time to implement. There are no new tasks introduced with inclusion of operator actions in management of cooling water loads. It is extremely unlikely that an operator would terminate execution of this procedure with the system demand exceeding the emergency intake line capacity and have the condition remain unnoticed. Once some of the cooling water loads have been isolated, the time available to successfully complete the procedure is extended since the cooling water flow is reduced. The stated objective of the procedure is to ensure cooling water flow demand is reduced to within the capacity of the Emergency Intake Line and the operator would realize that the procedure objective had not been met. Specifically, steps within the procedure require checks of the cooling water system flow to verify that the procedure objectives have been met. Operators, by training and position responsibility, are required to monitor system performance. Operator training also specifically stresses the importance of observing system response to actions such as changes in system flow when valves are closed or pumps are shut off. Success of operator actions is immediately verifiable from the control room. Also, control room management would be present with responsibilities to check on operator and system performance. Specifically the shift supervisor is charged with the responsibility for observing overall plant and system performance from a "big-picture" perspective to assure that plant needs are met.

Based on these considerations, NSP concluded that operator acts of omission are very unlikely to cause failure to properly reduce cooling water system flow within the capacity of the emergency cooling water intake line. These acts would not result in the possibility for creating a malfunction of a different type than previously evaluated.

Operator acts of commission would mean that while the operator was executing the procedure to reduce cooling water flow that a required task is performed incorrectly when attempted or an extraneous task is performed which defeats successful completion of the procedure. Since all operator acts required by procedure are in the control room, only operator acts which can be performed in the control room are considered as viable acts of commission. Evaluation of acts of commission determined that three types of acts bound the possibilities of acts which will defeat the procedure: (1) acts which would increase the cooling water system demand; (2) acts which would deprive systems required for safe shutdown of the plant from receiving cooling water; and (3) acts which would isolate a cooling water pump incorrectly.

If the cooling water system flow were significantly increased, rather than decreased as required by the procedure, the time to implement the procedure would be reduced and the possibility of damaging both operating cooling water pumps would be increased. As stated previously, the instrument air system is not seismically qualified and thus, fail-open air operated valves are assumed to fully open and increase the system flow when the earthquake occurs. With the assumed post-seismic loss of instrument air and non-safeguards power, there are no extraneous control room operator acts which could significantly increase system flow. Therefore, acts which would increase flow are excluded from further consideration.

Acts of commission which would deprive systems required for safe shutdown of the plant from receiving cooling water were evaluated. The components needed for safe shutdown, to which cooling water flow is not isolated are (1) auxiliary feedwater, (2) emergency diesel generator, (3) control room chiller, and (4) component cooling. The evaluation considered whether the isolated component will fail within a short time frame, what alarms and /or indications will alert the operator to the error, and whether backup equipment is available.

> (1) To evaluate the effect on auxiliary feedwater, it is assumed that the cooling water supply has already been established. For the cooling water supply to the auxiliary feedwater pumps to be completely isolated, four separate control switches must be operated. Manipulating all four switches is not considered a single act of commission and is considered extremely unlikely. In each case, the associated auxiliary feedwater pump will trip on low suction pressure when the supply valve is closed. The pump trip will be annunciated in the control room at

the panel where the supply valve was operated. This provides immediate feedback to the operator. The trip function protects the pump from damage. Only one auxiliary feedwater pump per unit is needed for adequate decay heat removal.

(2) The evaluation of reduction of flow to the emergency diesel generators (EDG) only applies to Unit 1. The Unit 2 emergency diesel generators do not use cooling water for heat removal. Reduction of cooling water flow to the Unit 1 EDGs is very unlikely. There are no remotely operated valves that can completely isolate flow to a cooling water header when both diesel driven cooling water pumps are operating (most limiting for supply line capacity). Also, there are no remotely operated valves that isolate the branch lines off of the main headers which supply the Unit 1 EDGs. This requires a control room operator to instruct an outplant operator to locally operate a manual valve to isolate the supply to an EDG. Such an act would no longer be considered a simple act of commission and is considered extremely unlikely. However, an annunciator would alert the control room operator to a problem with the Unit 1 EDG if it were inadvertently isolated. Only one EDG per unit is needed to maintain safe shutdown. Additionally, Unit 2 EDGs, which do not require cooling water, can be cross tied to supply Unit 1 loads in accordance with procedures available in the control room.

(3) The control room chillers are similar to the Unit 1 EDGs, in that the cooling water supply can only be isolated by a local manual valve. Again, these actions are considered unlikely and outside the scope of an operator act of commission. If the cooling water supply were isolated to a chiller, the chiller would trip. This would annunciate in the control room. Heatup studies indicate that the equipment in the rooms cooled by the chiller take one to two hours to heat up to a temperature that could affect their operability which allows the operators sufficient time to restore cooling. Only one chiller is needed to maintain safe shutdown.

(4) To completely isolate cooling water to all component cooling water heat exchangers (CCHX), four separate control switches must be operated. The procedure for reducing cooling water system flow does direct the operator to close two of these valves, one on each unit. If the remaining two valves were closed, then cooling water to all CCHX s would be isolated. Assuming that one component cooling (CC) pump per unit continued to run, then the CC fluid would heat up slowly. After a seismic event, the only component transferring heat to the CC system is the reactor coolant pump thermal barrier heat exchanger. This heat input is very low and would take several hours to raise the temperature of the CC system fluid significantly. The reactor coolant pump leakoff temperature

would increase and annunciate in the control room. The operator can maintain reactor coolant pump seal cooling by reestablishing cooling water to the CCHX or starting a charging pump. The ECCS pumps and heat exchangers cooled by the CC system do not and are not required to operate after a seismic event. Therefore, their availability is not reduced nor are they a heat input to the CC system.

Thus, NSP concluded an operator act of commission which results in loss of function of essential plant systems due to loss of cooling water flow is not a credible event.

The third consideration is the possibility of isolating a cooling water pump incorrectly. The procedure directs the operator to secure one of the two operating cooling water pumps. It is not important to safety which pump is secured. Securing one pump allows the running pump to operate nearer its operating point. If the non-running vertical pump switch were to be operated incorrectly, then both running pumps would continue to operate. In this case both pumps would operate at, or just less than, desired minimum flow. This is a long term pump wear issue, but not a pump failure issue. Therefore an error of this type does not significantly increase the possibility of a malfunction of equipment of a different type than previously evaluated.

NSP concluded that operator acts of commission related to implementing the procedure to manage cooling water loads do not result in the possibility for creating a malfunction of a different type than previously evaluated.

Probabilistic Human Reliability Analysis (HRA)

To also provide additional assurance that these proposed changes to the Prairie Island licensing bases do not adversely affect safe operation of the plant, NSP has performed a probabilistic evaluation of operator acts of omission or commission. The results presented below are preliminary. If the final results change any conclusion presented here, the results will be transmitted by separate letter.

The possibility that an act of omission or commission (during operator cooling water system load management following a design basis seismic event, a very low probability event) might create a malfunction of a different type than previously analyzed was evaluated. A probabilistic analysis using the methodology described in NUREG/CR-1278, "Handbook of Human Reliability Assessment with Emphasis on Nuclear Power Plant Applications" was performed and is discussed below.

To determine the relative significance of cooling water load management on the ability to cope with a seismic event, a comparative investigation of another operator evolution was undertaken. Analysis of implementation of the procedure to reestablish auxiliary feedwater (AFW) pump operation with suction from the cooling water system was performed. This operator evolution is credited and described in the Prairie Island USAR. Aligning the suction to the AFW pumps to the cooling water supply is necessary because the normal supply from the Condensate Storage Tank (CST) is not seismically qualified and is, therefore, not relied upon following a seismic event. Operator implementation of this procedure was analyzed using the same HRA methodology (NUREG/CR-1278) that was used for the cooling water load management operator action. The purpose of this analysis was to compare the calculated Human Error Probability (HEP) of the previously established auxiliary feedwater system operator action with the cooling water load management HEP. This comparison is appropriate since both procedures are performed using the same control room staff under the same event conditions.

In the probabilistic analysis of operator performance, the level of stress that the operator is experiencing plays a significant role. NUREG/CR-1278 identifies two main stress levels, optimum stress or high stress. Optimum stress is defined as the level of perceived stress that is conducive to optimum performance. High stress is defined as levels of stress higher than the optimum stress level. High stress is further broken down into two categories, moderately high and extremely high stress levels. Moderately high stress is defined as the level of stress that will be moderately disruptive to system-required behavior for most people. Extremely high stress is defined as the level of stress in which the performance of most people will deteriorate drastically. This is likely to occur when the onset of the stressor is sudden and the stressing situation persists for long periods. This level of high stress is associated with the feeling of threat to one's physical well-being or to one's self-esteem or professional status. The selection of the operator stress level assumed for this analysis (optimum, moderately high, or extremely high), and the factors used in the selection of that stress level, are identified below.

Errors of omission involve failure of the operator to initiate performance of a system-required task or action. The cooling water load management activity required by procedure is assumed to be unsuccessful if either of the critical operator actions described below are omitted. Errors of commission involve incorrect performance of a system-required task or action, given that a task or action is attempted, or the performance of some extraneous task or action that is not required by the system and which has the potential for contributing to some system-defined failure.

Critical operator tasks were identified for each procedure to be implemented. Failure of any of these critical tasks, whether due to omission or commission, was assumed to

lead to failure of the overall procedure implementation. Additionally, a review of non-critical tasks required by the procedure was included to determine their potential impact on the successful completion of the critical tasks.

Cooling Water Load Management

Upon recognition that a design basis earthquake has occurred, the control room operator is required by procedure to monitor intake bay level to confirm whether it is necessary to reduce cooling water system demand to match the capacity of the emergency intake line. This involves two critical types of activities:

1. Monitor cooling water system parameters.
2. Isolate cooling water system loads as necessary to balance flow demand with flow supply through the emergency intake line.

Failure to initiate or correctly perform one of these critical actions directed by the procedure is assumed to result in system demand exceeding the capacity of the emergency intake line. The operator is further directed to reduce the number of operating cooling water pumps to one. This action is intended to provide additional confidence that safeguards bay level will be preserved, and to prevent operation of the cooling water pumps at or below their minimum flow ratings. Pump operation at or below minimum flow is not operationally desirable, but would still allow the system to successfully provide its heat sink function.

Both of the critical tasks were reviewed to identify the potential for operator acts of commission which would result in unsuccessful completion of cooling water load management. In addition, the potential for detrimental operator acts of commission during the non-critical task to reduce the number of operating cooling water pumps were included in the evaluation.

There are multiple indications of the need to initiate cooling water load management available to the operator in the control room. Procedures are available in the control room and the operators are trained in their use. All operator actions are performed in the control room. Furthermore, there is substantial time to perform these actions. It is calculated that the supply to the cooling water pumps will be solely from the emergency intake line approximately 3-1/2 hours after the seismic event (assuming no action to reduce cooling water system demand). Moderately high stress is assumed due to the heavy task load, although there is a relatively long time available to respond.

The preliminary results of the human reliability analysis indicate a HEP of 1.1E-2 for implementing the cooling water load management procedure. Note that the time

calculated to be available for performance of this action (3-1/2 hours) is very conservative as described in the seismic event background on page 3, since it assumes there is no makeup from the river through the new intake structure immediately following the seismic event and it does not take credit for reduced cooling system flow as cooling water loads are successfully isolated. As cooling water loads are isolated, the time available to successfully complete the procedure is significantly extended since the cooling water flow is reduced. As a result, this HEP is conservative.

Reestablishing AFW Pump Operation with Suction from Cooling Water

Critical tasks involved in performing this procedure include:

1. Identification of the need to perform the switchover
2. Open the cooling water suction supply valve to the motor-driven AFW pump (MDAFWP)
3. Reset the low suction pressure trip for the MDAFWP
4. Close the motor-operated steam generator discharge isolation valves for the MDAFWP
5. Restart the pump
6. Jog open the discharge isolation valves to achieve the desired AFW flow

These actions must be performed for the MDAFWPs of both units. Also, time is somewhat limited to perform this evolution. Using the conservative assumption that all feedwater flow is lost at the initiation of the seismic event, operator action is required in approximately 90 minutes based on MAAP code analysis. The loss of offsite power causes the reactor coolant pumps to stop. This removes a large source of heat input to the steam generators and extends the time available for operator action prior to loss of the steam generators as an effective heat sink.

There are multiple indications of the need to switch the suction supply to the AFW pumps available to the operator in the control room. Procedures are available in the control room and the operators are trained in their use. All of the critical operator actions are performed in the control room. Only one pump in operation with flow to one steam generator (per unit) is required for safe shutdown. Stress is moderately high due to the limited time and the seismic event.

The preliminary results of the human reliability analysis indicate a HEP of 6.8E-2 for this operator action. As with the cooling water load management procedure, the time assumed to be available for performance of the procedure (90 minutes) is conservative, since it assumes the AFW pump suction supply from the CSTs is lost immediately following the seismic event, resulting in AFW pump trip on low suction pressure and

loss of all feedwater to the steam generators from the start of the event. Various alarms and other indications are available to help the operator diagnose the event. Procedures are available that would direct the operator to respond to the AFW pump suction problem prior to loss of pump suction. As a result, this HEP is conservative.

Conclusions of Probabilistic Analysis

This unreviewed safety question is concerned with the additional operator actions associated with cooling water system flow demand reduction due to the capacity of the emergency intake line. Compared to the HEP for reestablishing the AFW pump operation with suction from cooling water, a previously established operator evolution, the HEP for the cooling water load management procedure is lower. Therefore, this analysis compliments the conclusion drawn from the deterministic review, that operator acts of omission or commission associated with reduction in cooling water system flow do not result in the possibility for creating a malfunction of a different type than previously evaluated.

Conclusions of Safety Evaluation

Based on the deterministic considerations discussed above, taking credit for the intake canal and reliance upon operator action for management of cooling water loads is safe and the health and safety of the public is protected following a design basis seismic event. A probabilistic analysis supports these conclusions to the extent that management of cooling water loads is affected by operator acts of omission or commission. Furthermore, NSP believes that implementation of the procedure for management of cooling water loads following a seismic event improves plant safety.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

1. The proposed amendment will not involve a significant increase in the probability <u>or consequences of an accident previously evaluated</u>

Operation of the Prairie Island plant in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>Probability</u>

The accident of concern for this issue is a seismic event. None of the proposed changes can have any effect on the probability of a seismic event.

<u>Consequences</u>

(1). The intake canal has been evaluated for stability during a postulated seismic event. The results of the evaluation demonstrates that the banks of the canal will not liquefy or lose strength during the event. Therefore, taking credit for the intake canal stability does not increase the consequences of an accident previously evaluated.

(2). The use of operator action for systems important to safety to perform properly has been evaluated. There are adequate indications to allow the operator to recognize the occurrence of the event. A procedure provides guidance to the operator for reducing cooling water system demand. This procedure is available in the control room and all actions are accomplished in the control room. Adequate time is available for the operator to perform the tasks and to get feedback on the actions' success or failure. The operators have been trained on the use of the procedure and continuing training is planned. Therefore, the use of operator action does not significantly increase the consequences of an accident previously evaluated.

(3). The potential for operator acts of omission or commission while reducing cooling water system demand has been evaluated.

An operator act of omission while initially performing the procedure to reduce cooling water flow could result in cooling water system demand exceeding the emergency intake line capacity. However, due to the long time period within which the procedure must be implemented, control room management oversight and control room indications and alarms, it is unlikely that this condition would not be corrected.

Three types of operator acts of commission while performing the procedure to reduce cooling water flow were considered. (1) Acts which could increase flow and damage the cooling water pumps are not credible since the cooling water system flow is assumed to be near its maximum due to loss of the instrument air and non-safeguards power when the earthquake occurs. (2) Acts which would reduce flow to systems required for safe shutdown of the plant were evaluated. These acts would be indicated by control room alarms and corrected or out-plant actions would be required which involves more than a simple act of commission, thus, loss of function of supported systems due to loss of cooling water flow is not considered credible. (3) Acts which isolate a cooling water pump incorrectly were considered. This is a long term wear issue, but not a pump failure issue.

Operator acts of omission or commission have also been evaluated probabilistically. This evaluation demonstrated that the probability of an act of omission or commission is comparable to or less than other operator evolutions which have previously been licensed for effective performance of systems important to safety. This compliments the conclusions from the deterministic evaluation that these changes do not involve a significant increase in the probability of a previously evaluated accident.

Therefore, the potential of an operator act of omission or commission does not significantly increase the consequences of an accident previously evaluated.

Conclusion

In total the changes proposed by this license amendment request do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed

The Cooling Water System is provided in the plant to mitigate accidents and it is not a design basis accident initiator, thus these proposed changes do not increase the possibility of a new or different kind of accident.

The consideration of operator acts of omission or commission is limited to those acts arising from performance of the cooling water load management procedure. The evaluation of these actions showed that a new or different type of accident is not created.

In total, the possibility of a new or different kind of accident from any accident previously evaluated would not be created by these changes to the plant licensing basis or amendments to the Cooling Water Technical Specifications.

3. The proposed amendment will not involve a significant reduction in the margin of safety

The proposed changes do not involve a significant reduction in a margin of safety because the current Technical Specifications requirements for safe operation of the Prairie Island plant are maintained or increased. Plant margin of safety may be reduced by the reduced flow capacity of the emergency intake line. However, plant margin is restored by the remedial operator actions which preserve safe plant operation. Analysis shows that the intake canal will not fail during a seismic event and thus sufficient time for reducing cooling water system demand is provided. The procedure for reducing cooling water demand has been demonstrated on the plant simulator and operators have been trained. This procedure can be performed entirely from the control room. Thus, the changes proposed in this license amendment request do not involve a significant reduction in the margin of safety. Additionally, probabilistic evaluation compliments the conclusion that the likelihood for successful reduction of the cooling water system flow is very high.

Conclusion

Therefore, a significant reduction in the margin of safety would not be involved with these changes in the plant licensing basis and Cooling Water Technical Specification amendments.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by Nuclear Regulatory Commission regulations in 10 CFR Part 50, Section 50.92.

ENVIRONMENTAL ASSESSMENT

Northern States Power Company has evaluated the proposed changes and determined that:

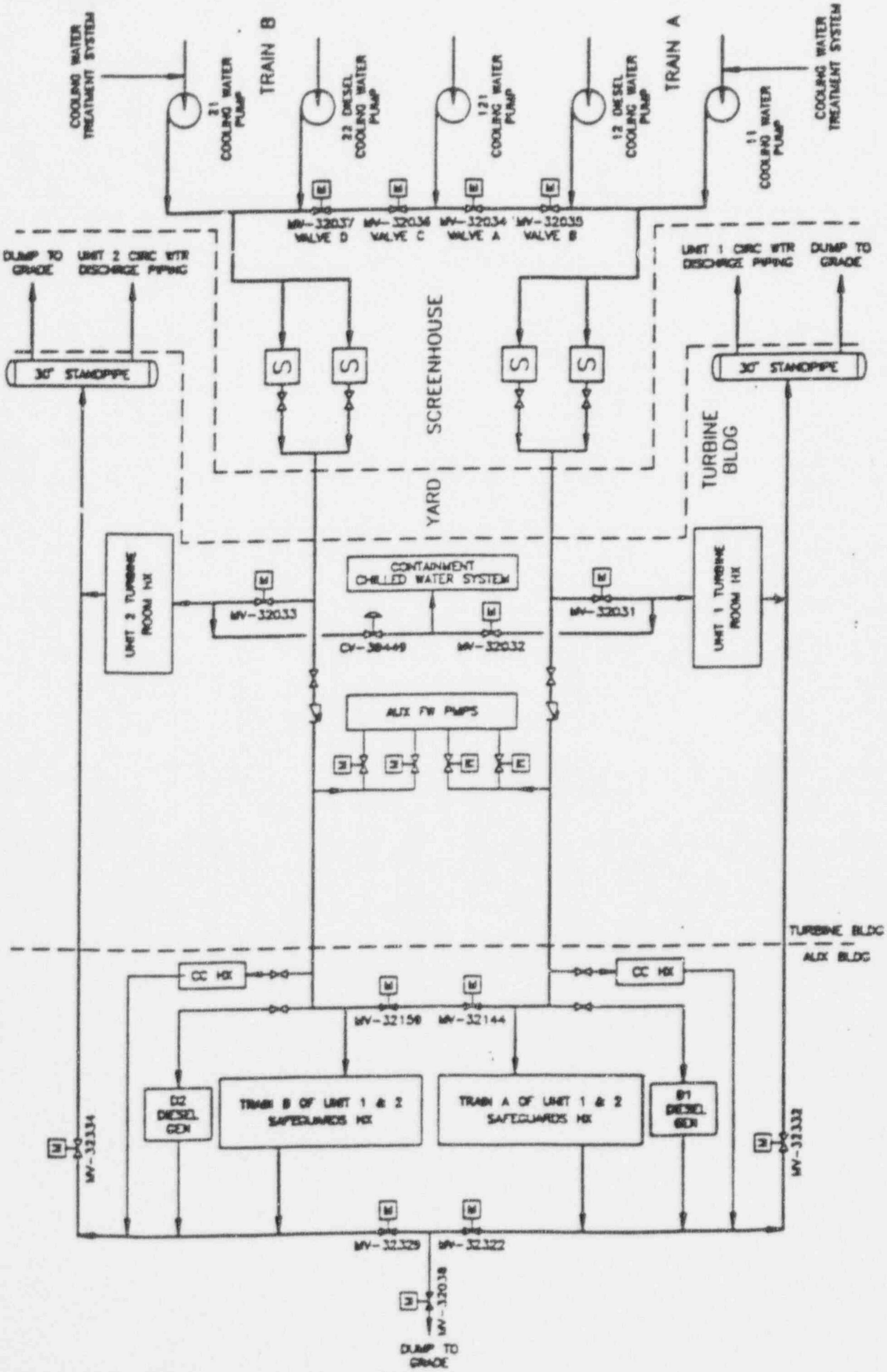
1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.

EXHIBIT A
FIGURE 1



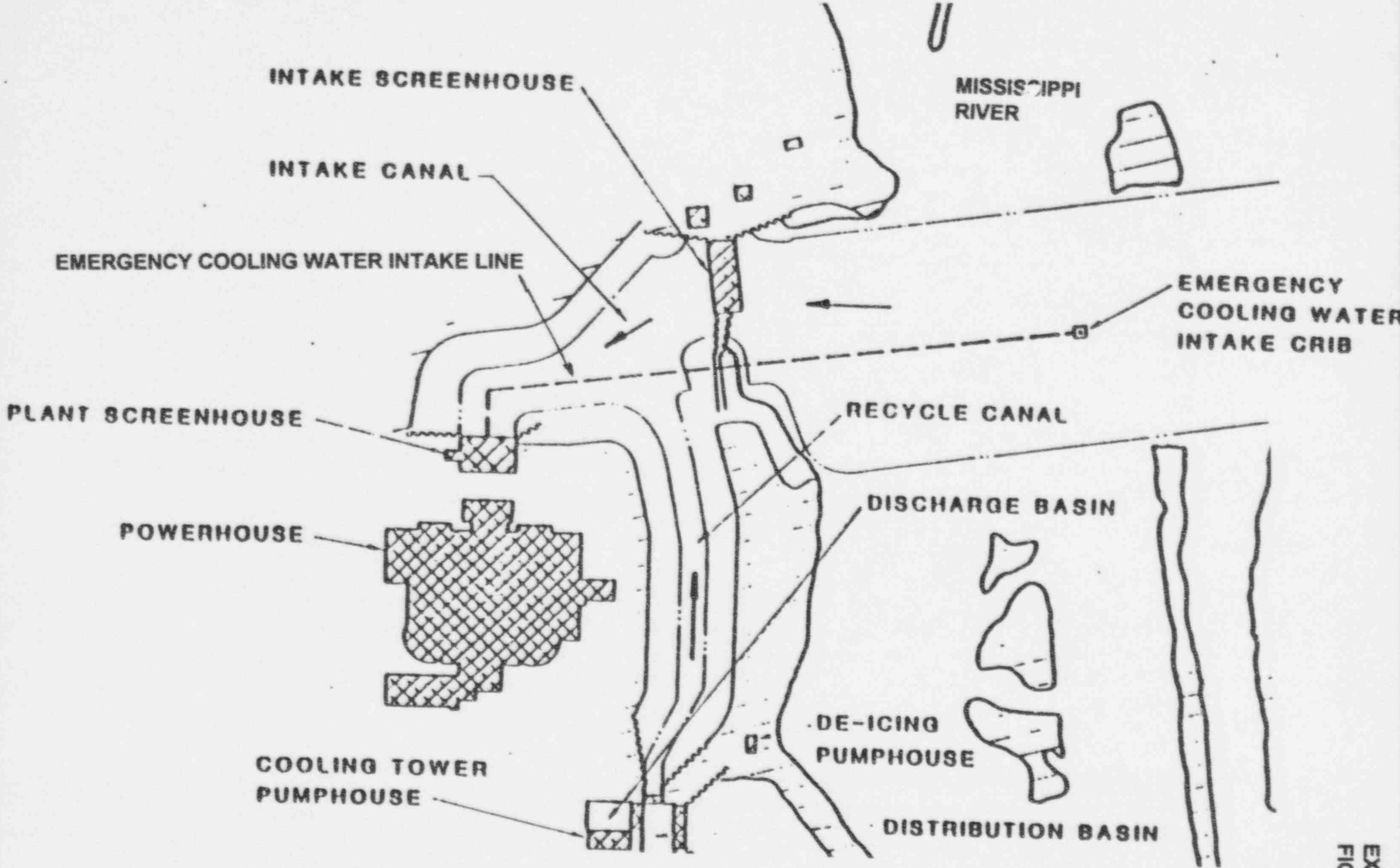


EXHIBIT A
FIGURE 2

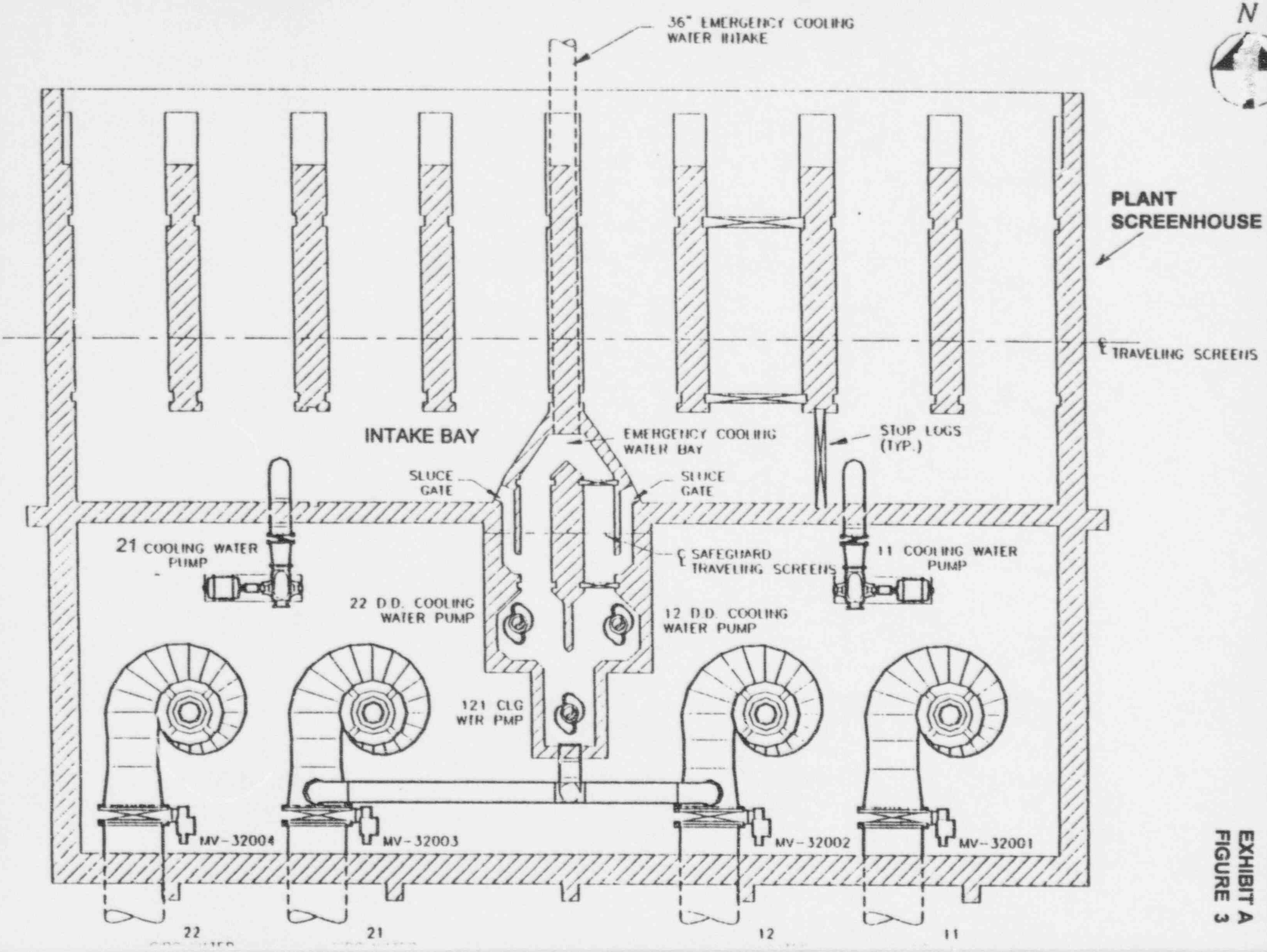


EXHIBIT A
FIGURE 3

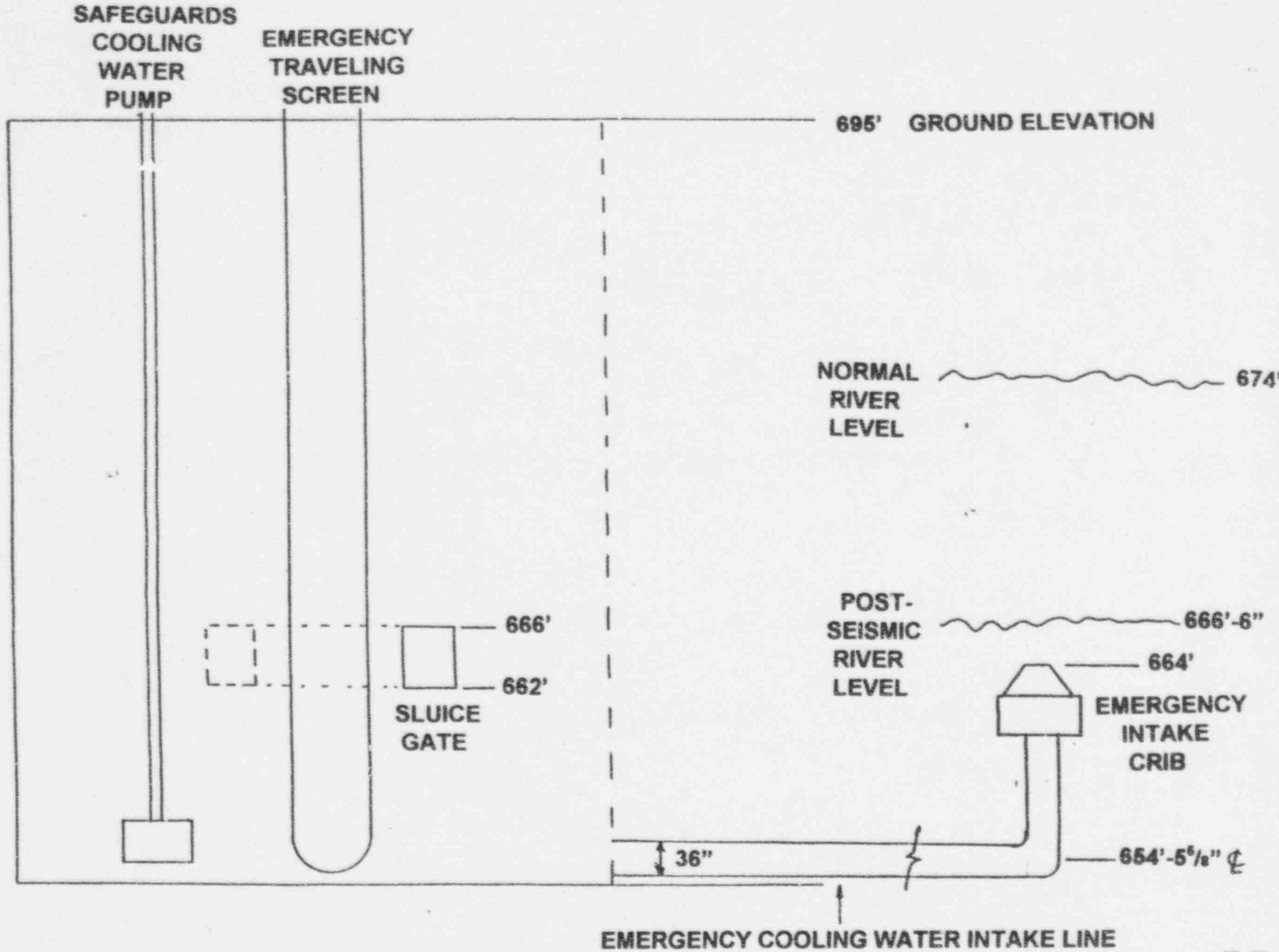


EXHIBIT A
FIGURE 4