CENTERIOR ENERGY

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Docket Number 50-346

License Number NPF-3

Serial Number 2318

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Request for Enforcement Discretion Regarding Technical Specification 3/4.7.5.1, Ultimate Heat Sink

Ladies and Gentlemen:

Enclosed is a request for Nuclear Regulatory Commission (NRC) enforcement discretion for the Davis-Besse Nuclear Power Station (DBNPS), Unit 1 Operating License Number NPF-3, Appendix A, Technical Specifications (TS). The proposed request involves TS 3/4.7.5.1, Ultimate Heat Sink, which presently requires that the ultimate heat sink (UHS) average water temperature be ≤ 85°F during plant Operating Modes 1 through 4. The NRC enforcement discretion would allow plant operation in these Modes with an UHS average water temperature ≤ 90°F.

The following provides the justification for this request, including the circumstances surrounding this situation, compensatory actions, the safety significance and potential consequences, the justification for the duration for which the discretion is being requested and the conclusions that a significant hazards consideration does not exist and that no irreversible environmental consequences would result.

Toledo Edison, as operator of the DBNPS, requests that the NRC grant this request for enforcement discretion in an expeditious manner. This request will be followed up with a license amendment request per discussions with the NRC Staff.

1. Requirements for Which Enforcement Discretion is Requested

   Enforcement discretion is requested from Technical Specification Limiting Condition for Operation 3.7.5.1, which requires that the ultimate heat sink average water temperature be ≤ 85°F during plant operating Modes 1 through 4. With this requirement not met, the plant is required to be placed in Mode 3 (Hot Standby) within 2.5 hours and in Mode 5 (Cold Shutdown) within the following 30 hours.

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The ultimate heat sink for the DBNPS is Lake Erie, which is the source of cooling water for the Service Water System. Lake water flows through an intake water system consisting of buried conduit to the intake canal. The intake canal flows to the intake structure, where the service water pumps are located.

An open forebay area ahead of the intake structure serves as an ultimate heat sink reservoir for an ensured source of water in case of an extreme lowering of the lake due to meteorological conditions or collapse of the intake canal or submerged pipe from an earthquake. The system described herein complies with Regulatory Guide 1.27.

2. Discussion of Circumstances Surrounding the Situation

The temperature of the ultimate heat sink peaked at 83.9°F on August 17, 1995. The daily peak temperature continues to trend upward. Under the current TS, the plant is required to be placed in Mode 3 (Hot Standby) within 2.5 hours of exceeding a UHS average temperature of 85°F, and in Mode 5 (Cold Shutdown) within the following 30 hours. As shown in the enclosure, a forced shutdown under this circumstance is unwarranted since continued full power operation with an UHS average temperature of up to 90°F is acceptable.

3. Compensatory Actions

During the period for which enforcement discretion is requested, the following compensatory measures will be taken:

a. UHS Average Temperature will be monitored on an hourly basis. If UHS Average Temperature exceeds 90°F, a plant shutdown in accordance with the requirements of the present TS will be commenced.

b. Operating personnel will be notified by Standing Order to ensure that these actions are maintained.

4. Evaluation of Safety Significance and Potential Consequences of the Proposed Request

A Safety Assessment and Significant Hazards Consideration (SASHC) has been prepared and is enclosed.

5. Justification for Enforcement Discretion Duration

Toledo Edison is requesting that the requirement to follow the shutdown requirement of TS 3/4.7.5.1 be waived for a period of 30 days.

6. Basis for Conclusion of No Significant Hazards Consideration and No Unreviewed Safety Question

   As described in the attached Safety Assessment and Significant Hazards Consideration, Toledo Edison has determined that a significant hazard and an unreviewed safety question do not exist. As a result, there is no potential detriment to the public's health and safety.

7. Basis for Conclusion of No Irreversible Environmental Consequences

   An Environmental Assessment has been prepared and is enclosed.

   As described in the Environmental Assessment, Toledo Edison has reviewed the proposed request against the criteria of 10 CFR 51.30. The requested enforcement discretion does not involve a significant hazards consideration, does not increase the types or amounts of effluents that may be released offsite, and does not increase individual or cumulative occupational radiation exposures. Accordingly, Toledo Edison finds that the requested enforcement discretion, if approved by the Nuclear Regulatory Commission, will have no significant impact on the environment and no Environmental Impact Statement is required.

This request for enforcement discretion has been reviewed and approved by the DBNPS Station Review Board.

Should you have any questions or require additional information, please contact Mr. William T. O'Connor, Manager - Regulatory Affairs, at (419) 249-2366.

Very truly yours,

John K. Wood
for JPS

MKL/laj

cc: W. L. Axelson, Director, NRC Region III, Division of Reactor Projects
L. L. Gundrum, NRC Project Manager
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J. R. Williams, Chief of Staff, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
Utility Radiological Safety Board

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION

TITLE:

Request for Enforcement Discretion Regarding Technical Specification (TS) 3/4.7.5.1, Plant Systems, Ultimate Heat Sink.

DESCRIPTION:

The purpose of the proposed enforcement discretion affects the Davis-Besse Nuclear Power Station (DBNPS) Operating License NPF-3, Appendix A Technical Specifications (TS) and associated Bases. The proposed change would permit deviation from the allowable ultimate heat sink (UHS) average water temperature, as specified in Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.5.1.b, from ≤ 85°F to ≤ 90°F.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The systems and components affected are the: Service Water, Component Cooling Water and Component Cooling Water Loads. The Technical Specification Limiting Condition for Operation 3.7.5.1.b of ≤ 85°F is affected.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES

The SW system provides cooling water from the Forebay to the following Safe Shutdown equipment after either a Safe Shutdown Earthquake (SSE) or a Large Break Loss of Coolant Accident (LOCA): The Containment Air Coolers (CACs), the Component Cooling Water (CCW) heat exchangers, the Emergency Core Cooling Systems (ECCS) room heat exchangers, the Control Room Emergency Ventilation System (CREVS), the seal water to the Hydrogen Dilution System Blowers. The SW system also provides a backup source of water for the Auxiliary Feedwater Pumps (AFPs) in the event of an SSE which renders the condensate storage tanks water supply unavailable. SW is also a backup source of water for the CCW system when all other CCW makeup water sources are not available following an SSE.

The CACs safety function is to remove one half of the post-LOCA heat removal requirement. The CREVS safety function is to maintain Control Room (CTRM) habitability and CTRM Environmental Qualification (EQ) requirements (≤110°F). Each ECCS room has two coolers. The ECCS room heat exchangers safety function is to maintain the ECCS room temperature ≤122°F during a LOCA with only one cooler in operation assuming a SW temperature of 72°F. One heat exchanger in each ECCS room may be isolated if SW is <72°F and still be considered OPERABLE. The Hydrogen Dilution Blowers safety function is to provide Auxiliary Building air to CTMT post-LOCA to decrease the post LOCA hydrogen concentration.

During a Design Basis Accident, the CCW supplies cooling water to the following essential components: High Pressure Injection (HPI) pumps 1 and 2 bearing oil coolers, Decay Heat Pumps/Low Pressure Injection (DH/LPI) 1 and 2 bearing housing coolers, Decay Heat (DH) Coolers 1 and 2, Containment (CTMT) Gas Analyzer heat exchangers 1 and 2, and Emergency Diesel Generator (EDG) jacket cooling water heat exchangers 1 and 2.

The HPI system provides emergency core cooling for small break LOCAs. The HPI system provides makeup to the Reactor Coolant System (RCS) due to contraction during excessive overcooling of the RCS. The HPI system also provides a source of borated water from the BWST for Shutdown Margin (SDM) requirements. The LPI system provides emergency core cooling and refill of the reactor vessel following a Large Break LOCA, post-LOCA CTMT sump recirculation capabilities, long term decay heat removal, and post-LOCA boron dilution. The DH coolers safety function is to provide adequate post-LOCA CTMT sump cooling requirements and normal and emergency shutdown cooling requirements.

The CTMT gas analyzers safety function is to be capable of immediately analyzing CTMT atmosphere to detect the build up of hydrogen in CTMT following a LOCA. This function is required to maintain hydrogen levels to acceptable limits.

The EDG jacket cooling water safety function is to maintain the required temperature of the lube oil and diesel engine. The basis for the safety related cooling water (CCW) source is to ensure that the EDGs will perform their intended safety function during and after an accident.

EFFECTS ON SAFETY:

Because Service Water (SW) serves loads during normal operations as well as during accident conditions, a temperature increase in its water supply will affect any accident analyses initial conditions as well as the post accident response. The discussion below considers each of these aspects.

A. Normal Operation

During normal operation, SW supplies cooling to the Containment Air Coolers (CAC's), the Component Cooling Water (CCW) heat exchangers and the Turbine Plant Cooling Water (TPCW) heat exchangers. An increase in the Intake Canal temperature to 90°F from an assumed maximum of 85°F will increase the normal operating temperatures of the components served by these systems. While TPCW-cooled components do not affect plant safety, they may limit plant operation. The TPCW system temperature is normally maintained at 85°F at the outlet of the heat exchangers. There is no Technical Specification associated with this value. A 90°F SW temperature will not permit this setpoint to be maintained. Therefore TPCW-cooled components will be closely monitored and appropriate action taken before any device is allowed to operate above its design temperature.

During normal operation, there is sufficient cooling capacity in the CCW system to accommodate the changes in the SW supply temperature. The automatic controls on the system maintain the CCW heat exchanger outlet temperature at 95°F. There is no Technical Specification associated with this value. Evaluation of the heat exchangers has determined that the 95°F outlet temperature can be maintained with a 90°F SW inlet temperature. Consequently, all loads served by CCW during normal power operation will be operated at normal conditions and there is no impact on the plant.

The CACs limit the normal containment air temperature, which is a design input to the LOCA and EQ analyses. Technical Specification 3.6.1.5 limits this temperature to 120°F. With SW at approximately 82°F, the containment air temperature is approximately 110°F. Because heat removal through the containment vessel shell supplements heat removal through the CACs, containment temperature increases will be less than any increase in SW inlet temperature. An increase of 8°F in SW temperature would not cause the containment air temperature to exceed 120°F.

B. Accidents

An increase in SW temperature affects the temperature of the CCW system and the reactor containment building temperature following a loss of coolant accident or a main steam line break accident. A design basis Loss of Coolant Accident (LOCA) imposes the greatest performance requirement on the SW system. In this postulated accident, it is assumed that a loss of offsite power occurs with concurrent failure of an emergency diesel generator. Thus, only one train of service water and ECCS pumps are assumed to be available. In a design basis LOCA, the critical period for refilling and cooling the reactor core occurs within the first few minutes of the accident. Following refill of the reactor vessel, the fuel will be adequately cooled. Service water temperature does not directly impact core cooling during this portion of the event, because the reactor is refilled via the low pressure injection (LPI) pumps using water from the borated water storage tank (BWST). In addition, the available containment spray (CS) pump injects water directly into containment from the borated water storage tank without any cooling supplied by service water. Until the BWST is exhausted, cooling is not required for the decay heat coolers. The initial blowdown is sufficiently quick, that heat removal from containment air coolers is not effective in reducing the initial temperature/pressure spike in containment.

Following consumption of the BWST inventory, containment cooling for preservation of containment equipment qualification must be maintained. At this point, more than 30 minutes into the transient (depending on pump combinations in service), the suction of both the LPI pumps and the CS pumps is transferred to the containment emergency sump, where effluent from the break and CS has accumulated. A CCW heat exchanger, cooled by SW, is required to provide cooling to the Decay Heat Removal (DHR) heat exchangers. An increase in design SW temperature from 85°F to 90°F will cause a similar increase in CCW temperature. Upon making the transfer to the emergency sump (which is at a higher temperature than the BWST), both the LPI injection temperature and the CS injection temperature increase, giving rise to a temporary increase in containment temperature and pressure. This increase peaks at approximately 10,000 seconds and is lower than the initial pressure/temperature peak. The long term containment temperature following a LOCA is controlled by the heat removal through the DHR system and the containment air coolers. The impact of an increase in SW temperature on the heat removal through these systems is discussed below.

The previous containment analysis, as described in references 2a and 3a, used a DHR heat exchanger heat transfer coefficient of 300 BTU/hr-ft2-°F. This value was conservatively used to bound potential degradation from the design heat exchanger performance of 478 BTU/hr-ft2-°F. However, the most recent performance tests (reference 4) indicate that the actual performance of the worst DHR cooler is 407 BTU/hr-ft2-°F. Sensitivity of containment temperatures to DHR cooler performance has been explored in references 2a and 2b for a range between 250 and 478 BTU/hr-ft2-°F. From a graph of calculated containment temperature increase vs. DHR cooler heat transfer coefficients, a coefficient of 400 BTU/hr-ft2-°F would provide a long term containment temperature approximately 7°F below the analyzed temperature. Thus, the margin available in DHR heat exchanger performance will more than compensate for a 5°F increase in service water temperature. This conclusion is confirmed by reference 2c, where a containment temperature response profile was re-run using a heat transfer coefficient of 400 BTU/hr-ft2-°F and a service water temperature of 90°F rather than the original 85°F. The new short term containment temperature profile is unaffected, while the long term profile is lower.

Containment air coolers (CAC) will receive SW water at a temperature of 90°F. In the previous containment analysis, the CACs received 85°F cooling water at an analyzed flow rate of 1150 gpm. Reference 5 indicates that the current CAC flow is greater than 1350 gpm. This flow represents the current flow balance performed under simulated accident conditions. The CACs have a large water side temperature rise in LOCA service, therefore, performance is more affected by flow rate than minor inlet temperature changes. From reference 2c, the CAC heat removal will be greater than that used in the containment analysis for an inlet temperature of 90°F with 1300 gpm SW flow. The greater performance will continue well into the transient, when containment temperatures are no longer high enough to affect equipment qualification. For Main Steam Line breaks, since CAC duty is better with 90°F inlet temperature and 1300 gpm SW flow (than with 85°F inlet temperature and 1150 gpm SW flow), the containment temperatures are also not increased from the previous analyses.

The peak CCW temperature predicted by the containment analysis of reference 2c is less than 108°F at approximately 15000 seconds following the accident. The peak CCW temperature in the existing analysis is slightly over 100°F. The increase is due to both the 5°F increase in SW temperature and the greater credited heat removal by the DHR heat exchanger. A CCW temperature of 108°F is well within the design temperature of the essential components served by CCW. From reference 6a and 6b, the design maximum cooling water temperature supply to the Emergency Diesel Generators and, the LPI bearings is 120°F. The maximum bearing temperature for HPI pump is 165°F. Plant surveillance testing data show that the bearing temperature tracks the CCW temperature; therefore, the HPI bearing temperature will be below 165°F. Therefore, safety related CCW loads will be adequately served even during the time of peak CCW thermal loading. Non-safety related loads (e.g. spent fuel pool cooling, containment loads, etc.) would have been automatically isolated by the safety features actuation system. However, these loads could be re-supplied at a later time, as required.

The ECCS room coolers are directly supplied by service water and are required to operate in design basis events since the normal ventilation system is assumed to be unavailable. Two coolers are provided in each of the two ECCS rooms. At low SW temperatures (less than approximately 72°F), one cooler is sufficient to provide 100 percent of the required cooling capacity. At higher SW temperatures, existing administrative controls require that both ECCS room coolers in each of the two ECCS rooms must be in service. From reference 2d, two ECCS room coolers are adequate up to a SW temperature in excess of 95°F, even with flow rates substantially degraded from normally accepted values.

The containment hydrogen dilution blowers are "Nash" pumps which utilize less than 10 gpm SW to provide seal water. The SW makeup connection to the CCW system will be unaffected. The inlet temperature of SW for these uses is not critical.

The emergency suction for Auxiliary Feedwater is supplied by service water directly. An increase in SW temperature to 90°F represents a very small increase in initial liquid enthalpy when compared to the large increase in enthalpy encountered by the feedwater as it is boiled in the steam generators. Due to the ample flow capacity of the auxiliary feedwater pumps, the increase in service water inlet enthalpy will have negligible effect on system response. Service water also provides cooling water to the motor driven feedwater pump (MDFP) seal water and bearing coolers. The vendor manual states that inlet cooling water is limited to 95°F. Therefore, increasing the SW to 90°F will not adversely affect the MDFP.

Control Room Emergency Ventilation units receive cooling water directly from service water and are required during a LOCA. These units are also provided with an air cooled condensing coil which is designed for ambient temperatures of up to 95°F. The air cooled condensing unit is automatically selected if refrigerant pressure increases due to inadequate water cooled condenser cooling. This is expected to occur at a SW temperature of approximately 110°F. Since the air cooled system backs up the Service Water cooled system, and increase in normal service water temperature does not impact the availability of CREVS.

Thus, existing margin in equipment performance will maintain containment response within the existing profile for all times of importance following a LOCA. All equipment will operate as designed for all transients.

Under normal conditions, the ultimate heat sink is Lake Erie. Lake Erie is connected to the intake canal by a 96" diameter inlet pipe. In the unlikely event of a collapse of the non-seismic portion of the intake canal, the forebay contains sufficient water to provide continuous cooling for more than 30 days. The analyses presently in the USAR assume that the intake structure forebay level is at least 562.0' International Great Lakes Datum (IGLD) and at an initial temperature of 85°F. Currently, the forebay level is approximately 569.5'. The low water datum of Lake Erie is 568.6 feet (I.G.L.D). The maximum variations in the mean monthly level are 4.2 feet above and 1.2 feet below the datum for the 110-year period that the data has been collected. At these levels, the intake canal contains approximately twice the volume of water and 23 percent more initial forebay surface area (USAR figure 2.4-7) for heat transfer than assumed in the original analyses. Therefore, increasing the ultimate heat sink temperature from 85°F to 90°F does not impact the ability to provide continuous cooling for a period of 30 days. The connection between Lake Erie and the intake canal can be reestablished well within the 30 day period.

Low water in the intake canal could also occur due to a maximum probable meteorological event. This consists of a sustained WSW wind of 70 mph for a six hour duration. This could result in lake water level decreasing below the level of the intake crib at 561.85' IGLD. The low level condition from this meteorological event lasts for a maximum period of 12 hours. T.S. 3/4.7.5.1 requires a plant shutdown if the forebay level reaches 562.0' IGLD. Since the low water condition due to a meteorological event is of a limited duration, increasing the ultimate heat sink temperature from 85°F to 90°F will not affect the ability to safely shutdown the plant.

In summary, the proposed enforcement discretion to increase the allowable ultimate heat sink (UHS) average water temperature, as specified in Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.5.1.b, from $\leq$ 85°F to $\leq$ 90°F will not adversely affect plant safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazard exists. No significant hazards are involved if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequence of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. Toledo Edison has reviewed the proposed enforcement discretion and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are significantly affected by the proposed change. The proposed change does not result in the operation of equipment important to safety outside their acceptable operating range.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not change the source term, containment isolation, or allowable releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed change. The proposed change does not result in installed equipment being operated in a manner outside its design operating range. No new or different equipment failure modes or mechanisms are introduced by the proposed change.

3. Not involve a significant reduction in a margin of safety because the proposed change is not a significant change to the initial conditions contributing to accident severity or consequences, consequently there are no significant reductions in a margin of safety.

CONCLUSION:

On the basis of the above, Toledo Edison has determined that the proposed enforcement discretion does not involve a significant hazards consideration. As this proposed enforcement discretion must be reviewed by the Nuclear Regulatory Commission, this does not constitute an unreviewed safety question.

REFERENCES:

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 199.

2. Calculations:
   a. C-NSA-049.02-012, rev. 0, Long Term Containment Response Following a LOCA w/DHR Cooler U=300 BTU/hr-ft^2-°F

   b. C-NSA-049.02-011, rev. 0, Effect of Degraded DHR Cooler 1-1 on Containment P/T Response Following a LOCA

   c. C-NSA-60.05-006 Rev 0

   d. C-NSA-032.02-003, rev. 3, Maximum Service Water Temperature

3. DBNPS Updated Safety Analysis Report through Revision 19.

   USAR

   a. Section 6.2, Containment System
   b. Section 2.4.11, Low Water Considerations
   c. Section 9.4.1, Control Room HVAC
   d. Section 9.2.5, Ultimate Heat Sink
   e. Section 9.2.1, Service Water System

4. 1991 DHR cooler performance tests DB-PF-04727 (09/06/91) DB-PF-04703 (09/03/91)

5. SW Flow Balance Test DB-SP-4019, DB-SP-4020 November 1994

6. System Descriptions:

   a. SD-042, Revision 0, Decay Heat Removal System.
   b. SD-003, Revision 3, Emergency Diesel Generators.
   c. SD-029B, Revision 1, Control Room Emergency Ventilation System.
   d. SD-18, Revision 1, Service Water
   e. SD-16, Revision 3, Component Cooling Water
   f. SD-23, Revision 2, Hydrogen Control
   g. SD-22B, Revision 1, Containment Air Coolers
   h. SD-38, Revision 2, High Pressure Injection

7. M480N-21, Vendor Manual for Motor Driven Feedwater Pump

8. Regulatory Guide 1.27, Ultimate Heat Sink for Nuclear Power Plants.

9. NUREG-0136, Operating License NPF-3, Safety Evaluation Report, December 1976.

Docket Number 50-346
License Number NPF-3
Serial Number 2318
Attachment 2

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action

This proposed action involves the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Operating License Number NPF-3, Appendix A, Technical Specifications (TS). Enforcement discretion is proposed to allow a temporary increase in the allowable ultimate heat sink (UHS) average water temperature, as specified in TS Limiting Condition for Operation (LCO) 3.7.5.1.b, from $\leq$ 85°F to $\leq$ 90°F. The current TS Action statement requires that the plant be placed in Hot Standby (Mode 3) within 2.5 hours and Cold Shutdown (Mode 5) within the following 30 hours, in the event that UHS average temperature exceeds 85°F.

Need for the Proposed Action

The changes proposed are needed to allow continued plant operation in the event that UHS average temperature exceeds 85°F before TS LCO 3.7.5.1.b can be revised.

Environmental Impact of the Proposed Action

Toledo Edison has determined that the structures, systems and components which could be affected by the proposed increase in allowable UHS average temperature will continue to be capable of performing their safety functions.

The proposed enforcement discretion will reduce the potential for unduly requiring cooldown and heatup transitions of plant equipment, thus preserving the cycling margin between plant design and actual operating history.

The proposed enforcement discretion involves a change to a requirement with respect to the use of facility components located within the restricted area as defined in 10CFR Part 20. As discussed in the Significant Hazards Consideration, this proposed enforcement discretion does not involve a significant hazards consideration. The proposed change to allow continued plant operation in the event the UHS average temperature exceeds 85°F does not alter source terms, containment isolation or allowable releases. In addition, the proposed change does not involve an increase in the amounts, and no change in the types, of any radiological effluents that may be allowed to be released offsite. Furthermore, there is no increase in the individual or cumulative occupational radiation exposure.

With regard to potential non-radiological impacts, the proposed enforcement discretion involves no increase in the amounts or change in types of any non-radiological effluents that may be released offsite, and has no other environmental impact.

Based on the above, Toledo Edison concludes that there are no significant radiological or non-radiological environmental impacts associated with the proposed enforcement discretion.

Alternatives to the Proposed Action

Since Toledo Edison has concluded that the environmental effects of the proposed action are not significant, any alternatives will have only similar or greater environmental impacts. The principal alternative would be not to grant the enforcement discretion. This would not reduce the environmental impacts attributable to the facility. Furthermore, it would force a shutdown of the facility in accordance with the present TS in the event UHS average temperature exceeds 85°F.

Alternative Use of Resources

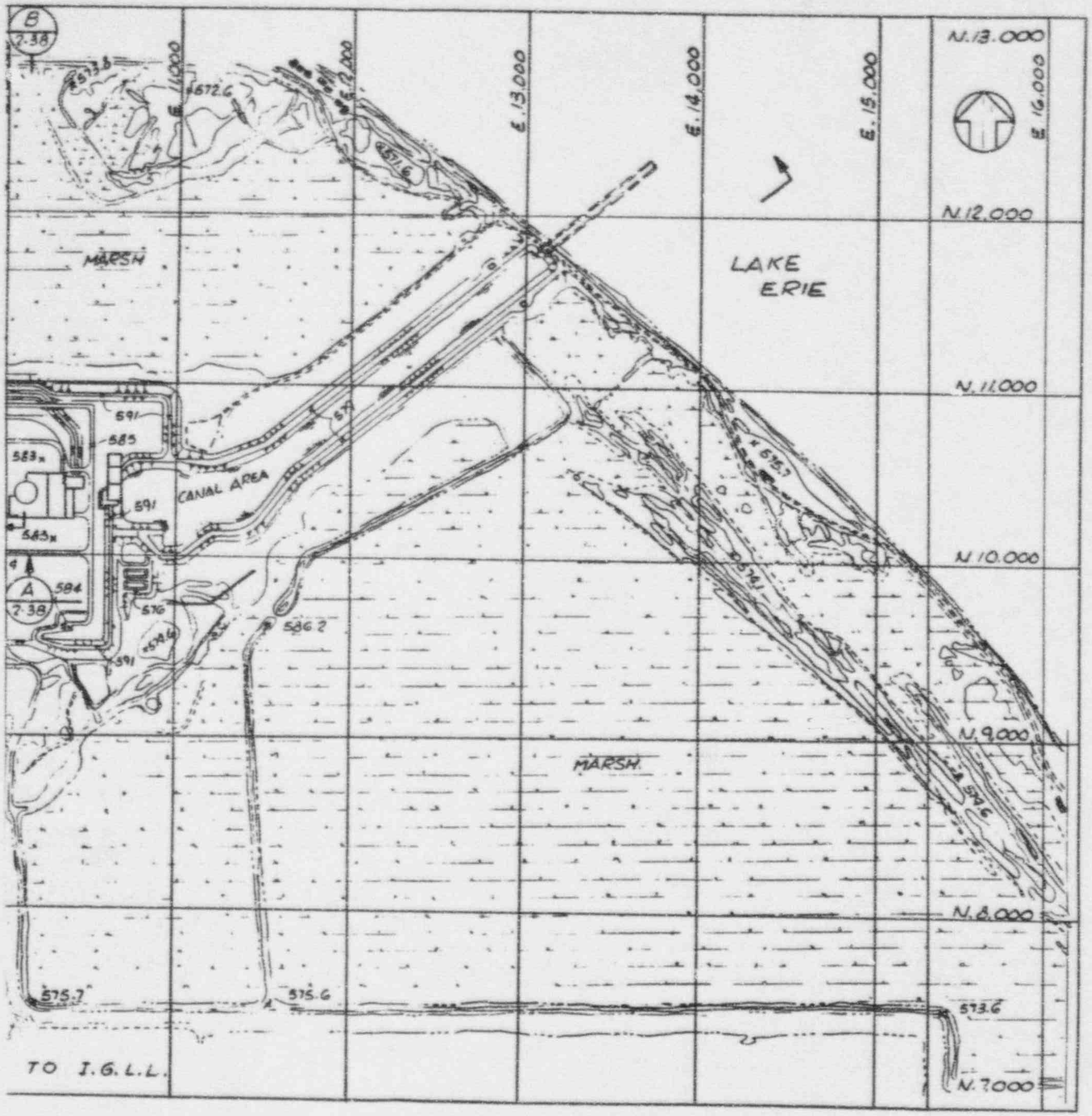
This action does not involve the use of resources not previously considered in the Final Environmental Statement Related to the Operation of the Davis-Besse Nuclear Power Station, Unit Number 1 (NUREG 75/097).

Finding of No Significant Impact

Toledo Edison has reviewed the proposed enforcement discretion against the criteria of 10CFR51.30 for an environmental assessment. As demonstrated above, the proposed enforcement discretion does not involve a significant hazards consideration, does not increase the types or amounts of effluents that may be released offsite, and does not increase individual or cumulative occupational radiation exposures. Accordingly, Toledo Edison finds that the proposed enforcement discretion, if approved by the Nuclear Regulatory Commission, will have no significant impact on the environment and that no Environmental Impact Statement is required.

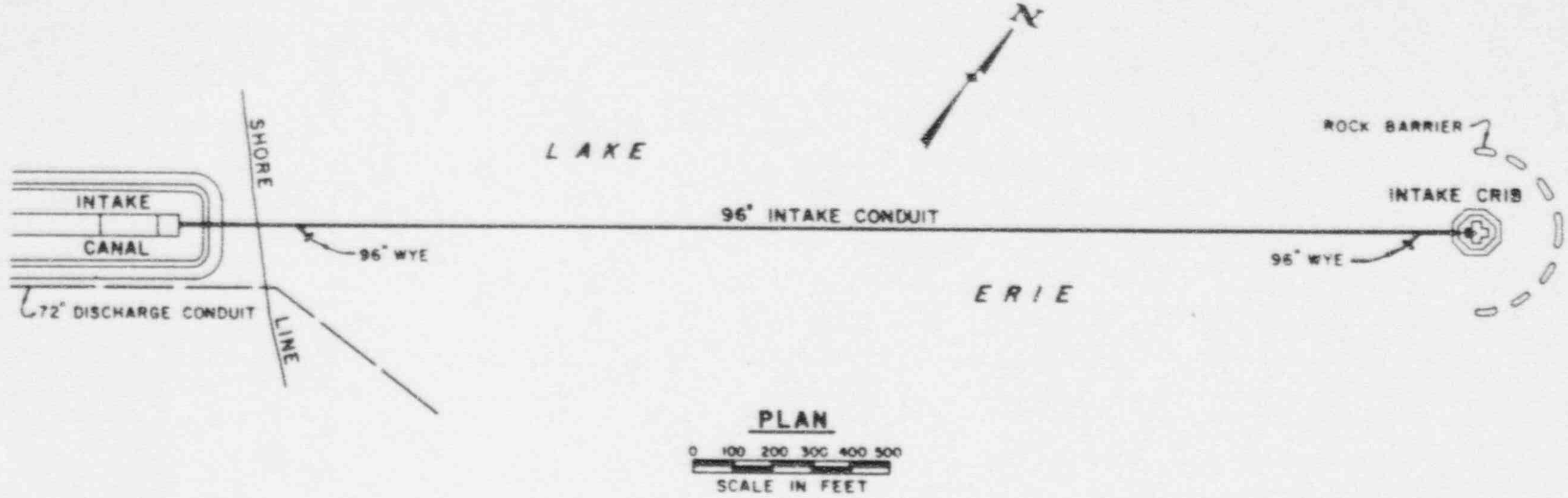
Docket Number 50-346
License Number NPF-3
Serial Number 2318
Attachment [illegible]

FIGURES

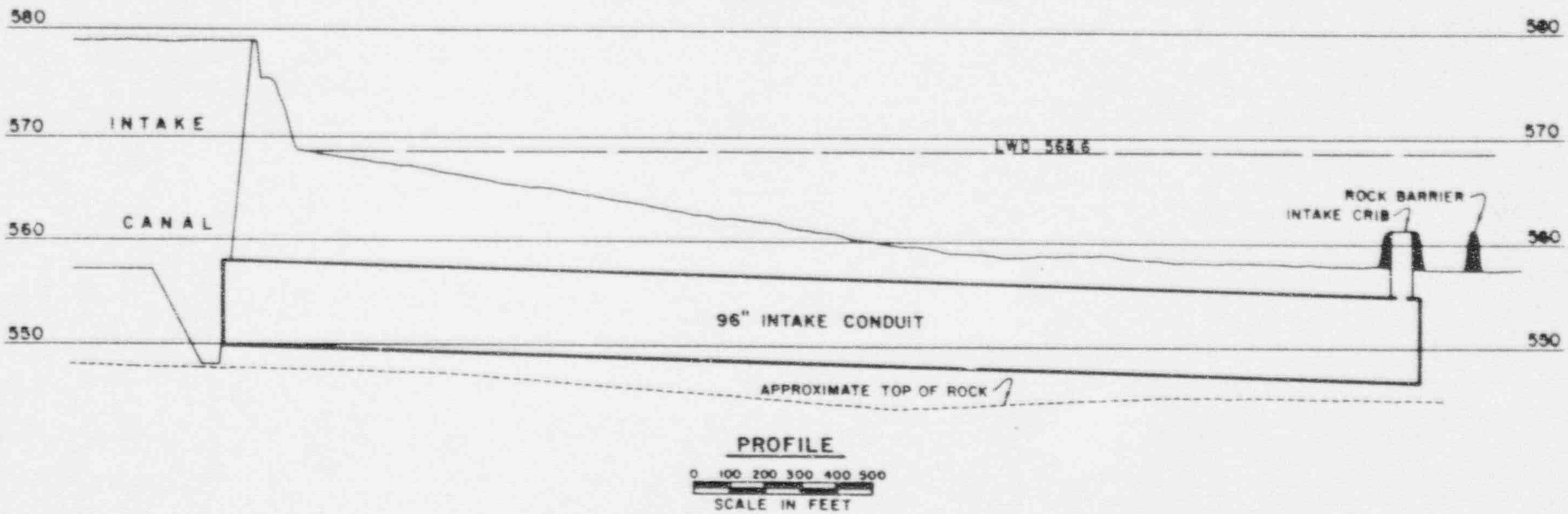


DAVIS-BESSE NUCLEAR POWER STATION
FINISHED SITE TOPOGRAPHY
FIGURE 2.4-2

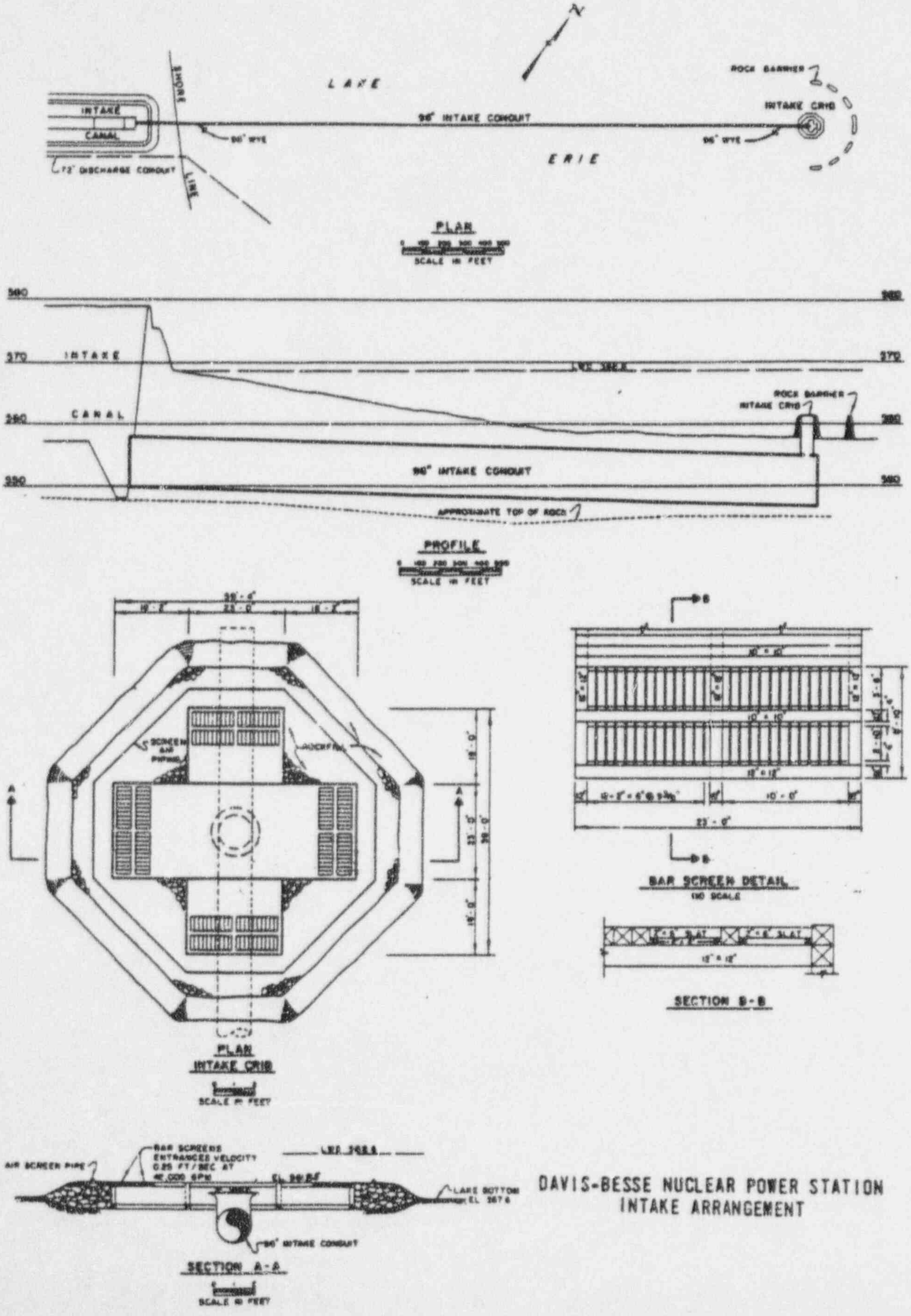
REVISION 0
JULY 1982



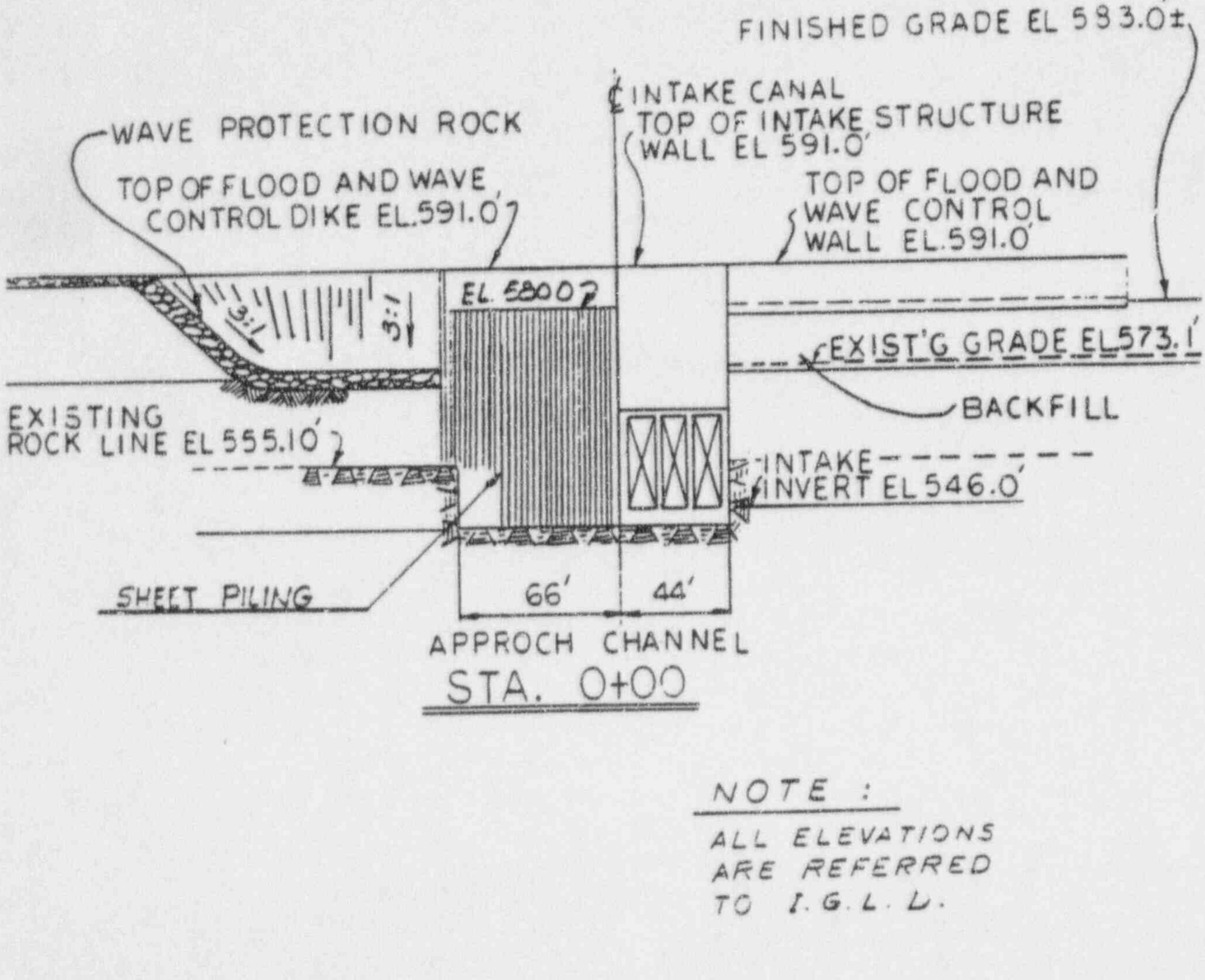
PLAN



PROFILE



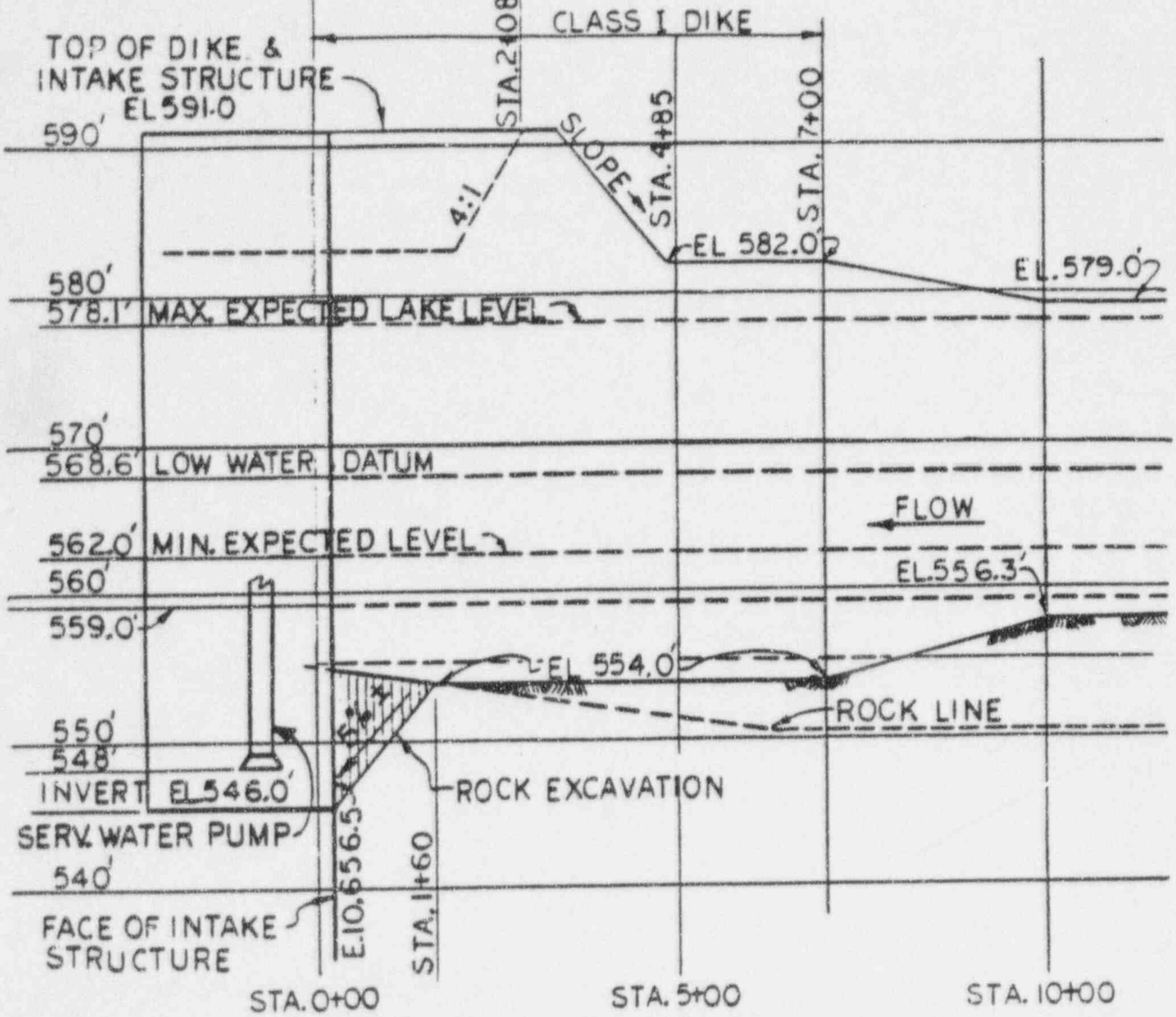
DAVIS-BESSE NUCLEAR POWER STATION
INTAKE ARRANGEMENT



NOTE :

ALL ELEVATIONS ARE REFERRED TO I.G.L.D.

DAVIS-BESSE NUCLEAR POWER STATION
CROSS SECTION @ STA. 0+00
INTAKE CANAL



WATER EL.	VOL. (GALS)	SURF. AREA (S.F.)
568.6'	19,780,000	216,900
562.0	10,119,000	175,400
560.0'	7,559,000	167,000
559.0'	6,325,500	162,800
557.0'	3,954,000	154,400
555.0'	1,692,000	141,800
554.0'	631,300	19,400
553.0'	494,800	17,200
548.0'	59,100	6,200

NOTE :

ALL ELEVATIONS ARE REFERRED TO I.G.L.D.

DAVIS-BESSE NUCLEAR POWER STATION
PROFILE ALONG THE CENTER
LINE OF THE INTAKE CANAL