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3.9-3a	Core Barrel Submodel
3.9-3b	Internals Submodel
3.9-3c	Reactor Pressure Vessel System Model
3.10-1	Orientation of Cable Tray Axes

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Introduction

The Watts Bar Nuclear Power plant was designed to meet the intent of the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits" published in July, 1967. The Watts Bar construction permit was issued in January, 1973. This UFSAR, however, addresses the NRC General Design Criteria (GDC) published as Appendix A to 10 CFR 50 in July, 1971, including Criterion 4 as amended October 27, 1987.

Each criterion is followed by a discussion of the design features and procedures which meet the intent of the criteria. Any exception to the 1971 GDC resulting from the earlier commitments is identified in the discussion of the corresponding criterion. References to other sections of the UFSAR are given for system design details.

3.1.2 WBNP Conformance with GDCs

3.1.2.1 Overall Requirements

Criterion 1 - Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A Quality Assurance Program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety function. Appropriate records of the design, fabrication, erection and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Compliance

Discussions related to the applicable codes, design criteria and standards used in the design of particular systems are contained in the appropriate SAR sections and in Tables 3.2-1, 3.2-2, 3.2-3, 3.2-4 and 3.2-5.

The Quality Assurance Program conforms to the requirements of 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants." Details of the program are given in Chapter 17.

Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:

1. Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,

2. Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and

3. The importance of the safety functions to be performed.

Compliance

The structures, systems, and components important to safety are designed to either withstand the effects of natural phenomena without loss of capability to perform their safety functions, or to fail in the safest condition. Those structures, systems, and components vital to the shutdown capability of the reactor are designed to withstand the maximum probable natural phenomenon expected at the site, determined from recorded data for the site vicinity, with appropriate margin to account for uncertainties in historical data. Appropriate combinations of normal, accident, and natural phenomena structural loadings are considered in the plant design.

The nature and magnitudes of the natural phenomena considered in the design of the plant are discussed in Sections 2.3, 2.4, and 2.5. Sections 3.2 through 3.10 discuss the design of the plant in relationship to natural events. Seismic and safety classifications, as well as other pertinent standards and information, are given in the sections discussing individual structures and components.

Criterion 3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Compliance

The plant is designed to minimize the probability of fires and explosions, and in the event of such occurrences to minimize the potential effects of such events to plant safety related equipment and personnel. Prime consideration was given to these requirements throughout the design process by providing for the duplication and physical separation of components in plant design and the use of materials classified as noncombustible and/or fire resistant wherever practical in safety-related areas of the plant. Equipment and facilities for fire protection, including detection, alarm, and extinguishment, are provided to protect both plant-equipment and personnel from fire, explosion, and the resultant release of toxic vapors. Fire-fighting systems are designed to assure that their rupture or inadvertent operation will not significantly impair systems important to safety. Portions of the fire-protection systems necessary to protect safety-related equipment in Class I structures are designed to seismic requirements.

The Fire Protection Systems provided are:
1. High pressure water,
2. Carbon dioxide, and
3. Portable extinguishers.

The Fire Protection System is designed such that a failure of any component of the system or inadvertent operation:
1. Does not cause a nuclear accident or significant release of radioactivity to the environment.
2. Does not impair the ability of equipment to safely shutdown and isolate the reactor or limit the releases of radioactivity to the environment in the event of a postulated accident.

The Fire Protection Systems for the Watts Bar Nuclear Plant are discussed in Section 9.5.1. Protection from fire in the control room is discussed in Section 6.4.

Criterion 4 - Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accident (LOCAs). These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Compliance

This criterion has been implemented as amended and published in the Federal Register, Volume 52, Number 207, October 27, 1987, 41288, which added the following:

> "However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The associated environmental parameters are identified and incorporated in the design requirements and specifications. Particular care was given to the extreme environmental conditions associated with major incidents such as LOCAs. Required equipment and instrumentation are identified, environmental conditions such as temperature, pressure, humidity, and irradiation, are calculated, and the effects of the latter on the former were evaluated either analytically or experimentally. The dynamic effects associated with an accident were carefully identified and assurance given that the structures and systems (including engineered safeguards) assumed undamaged in the total assessment of the accident consequences are suitably protected.

Emergency core cooling components are austenitic stainless steel or equivalent corrosion resistant material and hence are compatible with the containment atmosphere over the full range of exposure during the post accident conditions.

Where vital components cannot be located away from potential missiles, protective walls and slabs, local missile shielding, and restraining devices are provided to protect the containment and engineered safety feature components within the containment against damage from missiles generated by the equipment failures associated with the design basis accident (DBA).

The environmental design of safety-related items is discussed in Section 3.8 on the design of structures; Sections 6.2.2 and 6.2.3 on containment heat removal and air purification; and Section 9.4 on ventilation systems. Safety- related systems and components used the input from these sections for design as discussed in Section 3.11. Missile and environmental protection is discussed in Sections 3.5 and 3.11, respectively.

Criterion 5 - Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it is shown that such sharing will not impair significantly their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Compliance

The structures important to safety that are shared are the Auxiliary Building (Section 3.8), Control Building (Section 3.8), Diesel Generator Building (Section 3.8), and the intake pumping station (Section 3.8). Shared safety- related systems include the essential raw cooling water (Section 9.2), component cooling water (Section 9.2), fire protection (Section 9.5), spent fuel cooling (Section 9.1), fuel oil storage tanks (Section 9.5), preferred and emergency electric power (Section 8.2 and 8.3, respectively), chemical and volume control (Section 9.3), radioactive waste (Chapter 11), emergency gas treatment system (Sections 6.2 and 6.5), auxiliary control air system (Section 9.3), and control and Auxiliary Building ventilation systems (Section 6.4). The vital dc power system is shared to the extent that a few loads (e.g., the vital inverters) in one nuclear unit are energized by the dc power channels assigned primarily to power loads of the other unit. In no case does the sharing inhibit the safe shutdown of one unit while the other unit is experiencing an accident. All shared systems are sized for all credible initial combinations of normal and accident states for the two units, with appropriate isolation to prevent an accident condition in one unit from carrying into the other.

If the designated equipment configuration is revised to allow system testing or modification, appropriate action will be taken to ensure that the required system availability for accident mitigation is maintained.

3.1.2.2 Protection By Multiple Fission Product Barriers

Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Compliance

The reactor core with its related coolant, control, and protection systems is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The reactor trip system is designed to actuate a reactor trip for any anticipated combination of plant conditions when necessary to ensure that fuel design limits are not exceeded. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of loss of reactor coolant flow, trip of the turbine-generator, loss of normal feedwater and loss of both normal and preferred power sources.

Chapter 4 discusses the design bases and design evaluation of reactor components. Chapter 5 discusses the reactor coolant system. The details of the reactor trip and engineered safety features actuation system design and logic are discussed in Chapter 7. This information supports the accident analyses presented in Chapter 15.

Criterion 11 - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Compliance

A negative reactivity coefficient is a basic feature of core nuclear design as discussed in Chapter 4.

Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Compliance

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and nonpositive moderator temperature coefficients of reactivity.

Oscillations due to xenon spatial effects in the radial, diametral and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and nonpositive moderator temperature coefficients of reactivity.

Oscillations due to xenon spatial effects in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided as a result of reactor trip functions using the measured axial power imbalance as an input.

Oscillations due to xenon spatial effects in axial modes higher than the first overtone are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

The stability of the core against xenon-induced power oscillations and the functional requirements of instrumentation for monitoring and measuring core power distribution are discussed in Section 4.3. Details of the instrumentation design and logic are discussed in Chapter 7.

Criterion 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Compliance

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, temperatures, pressures, flows, and levels as necessary to assure that adequate plant safety can be maintained. Instrumentation is provided in the reactor coolant system, steam and power conversion system, the containment, engineered safety features systems, radiological waste systems and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room in proximity with the controls for maintaining the indicated parameter in the proper range.

The quantity and types of process instrumentation provided measures safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 6, 7, 8, 9, 11 and 12.

Criterion 14 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

Compliance

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. In addition to the loads imposed on the piping under operating conditions, consideration is also given to abnormal loadings, such as pipe rupture and seismic loadings as discussed in Sections 3.6 and 3.7, respectively. The piping is protected from overpressure by means of pressure relieving devices as required by applicable codes.

Reactor coolant pressure boundary materials selection and fabrication techniques ensure a low probability of gross rupture or significant leakage.

The materials of construction of the reactor coolant pressure boundary are protected by control of coolant chemistry from corrosion which might otherwise reduce the structural integrity of the boundary during its service lifetime.

The reactor coolant pressure boundary has provisions for inspections, testing and surveillance of critical areas to assess the structural and leak tight integrity. The details are given in Section 5.2. For the reactor vessel, a material surveillance program conforming to applicable codes is provided.

Means are provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary with indication in the control room as discussed in Section 5.2.

Criterion 15 - Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Compliance

Transient analyses presented in Section 5.2 lead to the conclusion that design conditions are not exceeded during normal operation. Protection and control set points are based on these transient analyses.

Additionally, reactor coolant pressure boundary components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing and integrated hydrostatic testing of assembled components.

The effect of radiation embrittlement is considered in reactor vessel design, and surveillance samples monitor adherence to expected conditions throughout plant life.

Multiple safety and relief valves are provided for the reactor coolant system. These valves and their set points meet ASME criteria for over-pressure protection. The ASME criteria are satisfactory based on a long history of industry use. Chapter 5 discusses reactor coolant system design.

Criterion 16 - Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Compliance

The reactor containment (Section 6.2) is a freestanding, continuous steel membrane structure housing the reactor and various auxiliary components including the ice condenser. The ice condenser (Section 6.7) limits the initial containment pressure to a value less than design during a large LOCA. A concrete Shield Building surrounding the steel vessel allows for collection of any containment leakage, which is subsequently processed by the emergency gas treatment system (Section 6.5) before release to the environment. The containment also contains a spray system (Section 6.2) which supplements the ice condenser in limiting pressure and which also provides long-term cooling following a LOCA. The design pressure is not exceeded in any pressure transients which result from combining the effects of heat sources with minimal operation of the engineered safety features.

The containment system is designed to provide for protection of the public from the consequences of a LOCA based on a postulated break of the reactor coolant piping up to and including a doubled-ended break of the largest reactor coolant pipe. Periodic containment leak rate measurements ensure that the leaktight barrier is maintained.

Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite power sources, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power sources.

Compliance

The capacity and capability of either the onsite or offsite electric power system is sufficient to assure that (1) specified fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Offsite Electrical Power System

The offsite electrical power source consists of two physically independent circuits which are energized and available. The offsite sources are two independent 161-kV transmission lines terminating at the 161-kV switchyard, providing power to the plant, on demand, via the common station service transformers to the onsite Class 1E distribution system.

Onsite Electrical Power System

The onsite electrical power system serves both nuclear power units and certain common plant equipment. It consists of two independent diesel generator systems, each system containing two diesel generator units, two redundant Class 1E electric power distribution trains, and four redundant vital instrument and control power channels, each provided with an uninterruptible ac power supply and distribution panel. A plant Class 1E dc power system is provided with four redundant divisions, each consisting of a battery, battery charger, and distribution panel. Each redundant onsite power supply, train, and channel has the capability and capacity to supply the required safety loads assuming the failure of its redundant counterpart.

For a detailed description and analysis of the offsite electrical power system and onsite electrical power system, see Sections 8.2 and 8.3, respectively.

Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including
operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Compliance

Inspection

In addition to continuous surveillance by visual and audible alarms for any abnormal condition, the onsite power system is designed to permit inspection and checking of wiring, insulation, connections, and switchboards to the extent that personnel safety is not jeopardized, equipment not damaged, and the plant not exposed to accidental tripping.

On-Line Testing

The onsite power system is designed with provision for periodic testing during normal operation with the unit on line, to the extent that the plant is not exposed to accidental tripping and the reliability of the safety system not degraded. These features include provisions for starting and loading of onsite emergency diesel generators, and starting and loading of individual or groups of engineered safeguards to their respective buses. The system is also designed to permit testing of larger integrated segments of the system during planned cooldown of the reactor coolant system.

Off-Line Testing

The onsite power system is designed with facilities for a complete test of the operability of the system as a whole from initiation of protection system, starting and loading of the diesel generators, transfer of power sources and the full operational sequence of engineered safety features.

Inspection and testing of electrical power systems is further described in Sections 8.3.1.1 and 8.3.2.1.

Criterion 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Compliance

The plant is provided with a separate structure designated as the Control Building. Within the Control Building are located control rooms, auxiliary instrument room, computer room, battery and dc equipment rooms (including space for instrument motor generator, etc), switchyard relay room, plant communications room and service facilities such as shift engineer office, kitchen, instrument shop, toilet facilities, and mechanical equipment room for heating, ventilating, and air conditioning equipment.

The main control room was provided with unit control panels for each of the two units, the switchyard, electrical recording, dc distribution, operation of the diesel generator system, and for those systems shared by the two units. The unit control panels contain those instruments and controls necessary for operation of the unit, functions such as the reactor and its auxiliary system, turbine generator, and the steam and power conversion systems. Selection of loading from the various plant electrical distribution boards such as the startup boards, common service board, shutdown boards, and motor control centers is accomplished from the unit control panels.

The control room is designed and equipped to minimize the effects of possible events such as fire, high radiation levels, and excessive temperature which might preclude occupancy. The main control room is continuously occupied by qualified operating personnel under all operating and accident conditions except in the case of events such as fire or smoke which could necessitate its evacuation. In the unlikely event that control room occupancy becomes impossible, provisions have been made to bring the reactor units to, and maintain them in, a hot shutdown condition, from a location external to the main control room. By use of appropriate procedures and available equipment, the unit can also be brought to cold shutdown conditions.

Sufficient shielding, distance, and containment integrity are provided to assure that under postulated accident conditions control room personnel shall not be subjected to radiation doses which would exceed 5 rem to the whole body, or its equivalent to any part of the body, including doses received during both ingress and egress. Control room ventilation is provided by a system having a large percentage of recirculated air. After an accident, makeup air is automatically routed through a system of HEPA and charcoal filters.

The design of the control room for occupancy during accidents is discussed in Section 6.4. The heating, ventilation, and air conditioning of the Control Building is discussed in Section 9.4. Radiation doses to control personnel following a LOCA are evaluated in Section 15.5.3. Radiation protection design features are discussed in Section 12.3.

3.1.2.3 Protection and Reactivity Control Systems

Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Compliance

A fully automatic protection system (with appropriate redundant channels) is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accord with IEEE Standard 279-1971. The reactor trip system automatically initiates a reactor trip when any appropriate monitored variable or combination of variables exceed the normal operating range. Setpoints are chosen to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all the full length rod cluster control assemblies. This will allow the assemblies to free fall into the core, rapidly reducing reactor power output.

The engineered safety features actuation system automatically initiates emergency core cooling, and other safeguards functions, by sensing accident conditions using redundant process protection system channels measuring diverse parameters. Manual actuation of safeguards is relied upon where ample time is available for operator action. The ESF actuation system also provides a reactor trip on manual or automatic safety injection (S) signal generation.

The response and adequacy of the protection systems is analyzed for all conditions specified by the ANS N18.2 standard through Condition IV.

For further discussion of the reactor trip system and engineered safety features actuation system, see Sections 7.2 and 7.3, respectively.

Criterion 21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of protection function and (2) removal from service of any component or channel does not result in a loss of the required minimum redundancy unless the acceptable reliability of operation of the protection can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Compliance

The protection system is designed for high functional reliability and inservice testability. The design employs redundant logic trains, and measurement and equipment diversity.

The protection system is designed in accordance with IEEE Standard 279-1971. All safety actuation circuitry is provided with a capability for testing with the reactor at power. The protection systems, including the process protection system, nuclear instrumentation system and the engineered safety features test cabinet comply with Regulatory Guide 1.22 on periodic testing of protection system actuation functions. Under the present design, there are protective functions which are not tested at power. The functions can be tested under shutdown plant conditions, so that they do not interrupt power operation, as allowed by Regulatory Guide 1.22. For those process protection functions that may be tested in bypass, alarms are provided in the control room and at the process rack to indicate the bypassed condition. Additional information on the capability of the process protection system to be tested in the bypassed mode is provided in Section 7.2.2.2, Subsections 10, 11, 12, 13 and 14.

In those cases where equipment cannot be tested at power, it is only the actuation device function which is not tested. The logic associated with the actuation devices has the capability for testing at power. Such testing will disclose failures or reduction in redundancy which may have occurred. Removal from service of any single channel or component does not result in loss of minimum required redundancy. For example, a two-of-three function becomes a one-of-two function when one channel is removed. (Note that this is not true for the logic trains which are effectively a one-out-of-two logic).

Semiautomatic testers are built into each of the two logic trains in a protection system. These testers have the capability of testing the major part of the protection system very rapidly while the reactor is at power. Between tests, a number of internal protection system points including the associated power supplies and fuses are continuously monitored. Outputs of the monitors are logically processed to provide alarms for failures in one train and automatic reactor trip for failures in both trains. Self-testing provision is designed into each tester. Additional details can be found in Sections 7.2 and 7.3.

Criterion 22 - Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Compliance

Design of protection systems includes consideration of natural phenomena, normal maintenance, testing and accident conditions such that the protection functions are always available.

Sufficient redundancy and independence is designed into the protection system to assure that no single failure, or removal from service of any component or channel of a system, will result in loss of the protection function. The minimum redundancy is exceeded in each protection function which is active with the reactor at power. Functional diversity and consequential location diversity are designed into the system. For example, loss of one feedwater pump would actuate one pressure reactor trip, one high-level trip, one low-level trip, and two temperature trips. The protective system is discussed in detail in Sections 7.2 and 7.3.

Criterion 23 - Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Compliance

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip and engineered safety features actuation channel (except for containment spray and switchover from injection to recirculation) is designed on the deenergize-to-trip principle so loss of power, disconnection, open-channel faults, and the majority of internal channel short-circuit faults cause the channel to go into its tripped mode. The protection system is discussed in Sections 7.2 and 7.3.

Criterion 24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Compliance

The protection system is separate and distinct from control systems. Control systems may be dependent on the protection system in that control signals are derived from protection system measurements where applicable. These signals are transferred to the control system by isolation devices which are classified as protection components. The adequacy of system isolation has been verified by testing under conditions of postulated credible faults. The failure or removal of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection system leaves intact a system which satisfies the requirements of the protection system. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train. For discussion of details of compliance, see Chapter 7.

Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Compliance

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus, in the postulated accidental withdrawal, (assumed to be initiated by a control malfunction) flux, temperature, pressure, level and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

Analyses of the effects of possible malfunctions are discussed in Chapter 15. These analyses show that for postulated dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution and initiate reboration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Compliance

Two reactivity control systems are provided. These are rod cluster control assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The full length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks along with the full length control banks are designed to shutdown the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCAs are presented in Chapter 4 and the operation is discussed in Chapter 7. The means of controlling the boric acid concentration is described in Chapter 9. Performance analyses under accident conditions are included in Chapter 15.

Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Compliance

Sufficient shutdown capability is provided to maintain the core subcritical for any anticipated cooldown transient, e.g., accidental opening of a steam bypass or relief valve, or safety valve stuck open. This shutdown capability is achieved by a combination of RCCA insertion and automatic boron addition via the emergency core cooling system with the most reactive control rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to supplement the RCCA in maintaining the shutdown margin for the long-term conditions of xenon decay and plant cooldown. For further discussion, see Sections 4.3 and 7.2.

Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Compliance

The maximum reactivity worth of control rods and the maximum rate of reactivity insertion employing control rods and boron removal are limited to values that prevent rupture of the reactor coolant system boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the withdrawal of RCCA and the dilution of the boric acid in the reactor coolant systems are specified in the Technical Specifications for the facility. The specification includes appropriate graphs that show the permissible mutual withdrawal limits and overlap of functions of the several RCCA banks as a function of power. These data on reactivity insertion rates, dilution and withdrawal limits are also discussed in Section 4.3. The capability of the chemical and volume control system to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9. The relationship of the reactivity insertion rates to plant safety is discussed in Chapter 15.

Criterion 29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Compliance

The protection and reactivity control systems are designed to assure an extremely high probability of fulfilling their intended functions. The design principles of diversity and redundancy coupled with a rigorous Quality Assurance Program and analyses support accomplishing this probability as does operating experience in plants using the same basic design. Section 4.2.3 and Sections 7.2 and 7.7 describe design bases and system design.

3.1.2.4 Fluid Systems

Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Compliance

All reactor coolant system components are designed, fabricated, inspected and tested in conformance with ASME Boiler and Pressure Vessel Code, Section III. Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature monitored leakoff between double 0-rings.

Leakage inside the reactor containment is drained to the Reactor Building sump where the level is monitored. Leakage is also detected by measuring the airborne activity and humidity of the containment.

See Section 5.2 for compliance of reactor coolant system components with ASME Boiler and Pressure Vessel Code, Section III.

Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

Compliance

Close control is maintained over material selection and fabrication for the reactor coolant system to assure that the boundary behaves in a nonbrittle manner. The reactor coolant system materials which are exposed to the coolant are corrosion resistant stainless steel or Inconel. The reference temperature RT_{NDT} of the reactor vessel material samples is established by Charpy V-Notch Tensile and 1/2 T compact tension tests. These tests also insure that materials with proper toughness properties and margins are used.

As part of the reactor vessel specification certain requirements which are not specified by the applicable ASME codes are performed, as follows:

1. A complete independent review of the supplier stress analysis is conducted by Westinghouse on the reactor vessel. Independent stress analysis is conducted in selected areas to ascertain that the design conditions imposed by the Westinghouse specification have been adequately accounted for.

2. The reactor vessel received a complete stress analysis, including analysis for cyclic pressure and temperature operation. The ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Class 1 rules to which these components are designed generally exempt them from cyclic analysis by code Paragraph NB-3222.4 (d).

3. Welding Preheat Requirements - On the reactor vessel, the non-mandatory preheat requirements for P1 and P12 material were performed on all weldments.

4. Reactor Vessel Out-of-Roundness Requirements - To ensure uniform coolant flow, the Westinghouse out-of-roundness requirements on the cylindrical region in the area of the thermal shield are above code. Section III, Class 1 out-of-roundness requirements are stated in Paragraph NB-4221.1 of the code. This referenced paragraph states that the difference in inches between the maximum and minimum inside diameters at any cross section shall not exceed the smaller of (D + 50)/200 and D/100, where D is the nominal inside diameter in inches at the cross section under consideration. Westinghouse required the out-of-roundness to be less than O.5 percent of the diameter in the cylindrical section of the vessel in the region of the thermal shield.

Special requirements were imposed by Westinghouse on the quality control procedure for both the basic materials of construction, and on various sub-assemblies and final assembly for the reactor coolant loop components. These requirements supplemented the rules for quality assurance stated in the applicable design codes. Examples of the special quality assurance requirements for the reactor vessel that are beyond code requirements are:

<u>Ultrasonic Examinations</u>

1. A 100-percent shear wave ultrasonic test of plate material.

2. An ultrasonic test of cladding bond.

3. Weld buildup areas to which the core support pads are attached are examined 100 percent.

4. Selected areas of completed vessel are ultrasonically mapped after hydrotest to provide a base for future in-service inspection.

5. Ultrasonic examination of the entire volume of all full penetration welds and heat affected zones in primary pressure boundary welds. The testing was done during fabrication upon completion of the welding and intermediate heat treatment.

Dye Penetrant Testing

1. Dye penetrant test all cladding surfaces and other vessel and head internal surfaces after hydrotest.

2. Dye penetrant examine the weld between the bottom head and instrumentation tubes, after each layer of weld is deposited.

3. Dye penetrant examine weld between CRDM housing and closure head and vent pipe and closure head after the first layer; each 1/4 inch of weld deposited and final surface.

4. Dye penetrant examine weld between the lower core support pad and the vessel shell, after the first layer and each 1/2 inch of weld metal are deposited.

5. Base metal or weld metal surfaces which are exposed to mechanical operations was dye penetrant or magnetic particle inspected.

Magnetic Particle Testing

1. Magnetic particle examination of all exterior vessel and head surfaces after hydrotest.

2. Magnetic particle examination of welds attaching the vessel supports, closure head lifting lugs, and refueling seal ledge to the reactor vessel, after the first layer and each 1/2 inch of weld metal are deposited.

3. Magnetic particle examination of all closure stud surfaces after threading. Continuous circular and longitudinal magnetization was used.

4. Magnetic particle examination of I.D. surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection was performed after forming and prior to cladding.

The fabrication and quality control techniques used in the fabrication of the reactor coolant system were equivalent to those for the reactor vessel. The inspections of reactor vessel, pressurizer, piping and steam generator were governed by ASME code requirements.

The permissible pressure - temperature relationships for selected heatup and cooldown rates were calculated using the methods of ASME Code Section III Non-mandatory Appendix G. The change in RT_{NDT} due to irradiation during plant life was calculated using conservative methods and will be verified periodically by surveillance program irradiated material test data.

See Section 5.2 for further discussion of compliance.

Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Compliance

The design of the reactor coolant pressure boundary provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary component's integrity. The reactor coolant pressure boundary will be periodically inspected under the provision of ASME Code, Section XI.

The RT_{NDT} properties of the reactor vessel core region forging, weldments and associated heat treated zones will be monitored by a surveillance program which is based on ASTM-E-185, Recommended Practice for Surveillance Testing on Structural Materials in Nuclear Reactors. Samples of reactor vessel plate materials will be retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation will be used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below RT_{NDT}, a pressure range was established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit was defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected RT_{NDT}, brittle fracture during normal operation is not considered to be a credible mode of failure. Additional details can be found in Section 5.2.

Criterion 33 - Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Compliance

The chemical and volume control system includes charging pumps and makeup paths that serve the safety function of maintaining reactor coolant inventory during normal operations and in the event of small reactor coolant leakages. The charging pumps can maintain reactor coolant pressure sufficiently high to allow orderly reactor shutdown for small tubing or small pipe breaks. Chapter 5 discusses the reactor coolant system, Section 9.3.4 discusses the chemical and volume control system, and Chapter 15 analyzes charging pump performance and fuel damage in event of postulated accidents. The offsite power system and onsite power system are discussed in Sections 8.2 and 8.3, respectively.

Criterion 34 - Residual Heat Removal (RHR)

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Compliance

The RHR system includes two redundant trains of pumps and heat exchangers each having sufficient heat removal capability to ensure fuel protection. The system is Seismic Category I and is provided electric power by either the preferred power system or the diesel generators of the standby power system. The normal steam and power conversion system is used for the first stage cooldown (i.e., above 350°F and 400 psig). The auxiliary feedwater system provides guaranteed backup of the steam and power conversion system in this function. The systems together accommodate the single-failure criterion.

Section 5.5.7 describes the RHR System.

Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Compliance

The Emergency Core Cooling System (ECCS) design and safety analysis is in accordance with the NRC Acceptance Criterion for Emergency Core Cooling System for Light-Water Power Reactors of December 1973 (10 CFR 50.46).

By combining the use of passive accumulators, centrifugal charging pumps, safety injection pumps, and residual heat removal pumps, emergency core cooling is provided even if there should be a failure of any component in any system. The ECCS employs passive system of accumulators which do not require any external signals or source of power. Two independent and redundant pumping systems are also provided to supplement the passive accumulator system. These systems are arranged sothat the single failure of any active component does not prevent meeting the short-term cooling requirements.

The primary function of the ECCS is to deliver borated cooling water to the reactor core in the event of a LOCA. This limits the fuel-clad temperature and thereby ensures that the core will remain intact and in place and fuel damage will not exceed that stipulated as a basis in the safety analysis (Chapter 15). This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical circumferential rupture of a reactor coolant loop,

2. A loss of coolant associated with a rod ejection accident.

The ECCS is described in Section 6.3. The LOCA, including an evaluation of consequences, is discussed in Chapter 15.

Criterion 36 - Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Compliance

Design provisions facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes and valves for visual or nondestructive inspection.

The components outside the containment are accessible for leaktightness inspection during operation of the reactor.

Details of the inspection program for the reactor vessel internals are included in Section 5.4. Inspection of the ECCS is discussed in Section 6.3.

Criterion 37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Compliance

The design provides for periodic testing of both active and passive components of the ECCS.

Proof tests of the components were performed in the manufacturer's shop. Preoperational system hydrostatic and performance tests demonstrate structural and leaktight integrity of components and proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Design provisions include special instrumentation, testing, and sampling lines to perform the tests during plant shutdown to demonstrate proper automatic operation of the ECCS.

Each active component of the ECCS may be individually actuated on the normal power source at any time during plant operation to demonstrate operability. Components are actuated on the emergency power system during preoperational tests and subsequently during plant shutdown per Technical Specifications.

Details of the ECCS are found in Section 6.3, with periodic testing procedures identified in Section 6.3.4. Performance under accident conditions is evaluated in Chapter 15.

Criterion 38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Compliance

Systems are provided to effect post-accident containment heat removal. The systems are classified as engineered safety features and as such incorporate a large degree of redundancy as well as being provided with multiple power supplies.

Containment heat removal is provided by the ice condenser and by containment sprays. The ice condenser is a passive system consisting of energy absorbing ice on which steam is condensed during and immediately after a LOCA. The condensation of steam on the ice limits the pressure and temperature to values less than containment design.

An air return system is used to circulate the containment gaseous inventory through the upper compartment, lower compartment, and ice condenser after the initial blowdown. This maintains proper mixing of the containment air and steam with the heat removal media, spray and ice, for the necessary heat removal.

The containment spray system sprays coolant automatically into the upper compartment containment atmosphere in the event of a large LOCA, thereby removing containment heat. The recirculation mode allows for a long-term heat removal by means of two spray systems, each of which contains redundant components including spray headers. The containment spray system consists of two completely separate trains consisting of pumps, heat exchangers, valves, and headers. The containment spray system is initiated automatically upon containment high pressure and is later manually realigned for proper operation in the recirculation mode. The residual heat removal spray contains two spray headers which are supplied from separate trains of the residual heat removal system by manual diversion of a portion of the low-pressure safety injection system flow during recirculation.

The loss of a single active component was assumed in the design of these systems. Emergency power system arrangements assure the proper functioning of the air return fan system, and the containment spray system and residual heat removal sprays.

The engineered safety features systems are discussed in Chapter 6; the electric power systems in Chapter 8; the protection systems in Chapter 7.

Criterion 39 - Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, pumps, spray nozzles, and piping to assure the integrity and capability of the system.

Compliance

The ice condenser design includes provisions for visual inspections of the ice bed flow channels, doors, and cooling equipment. The air return fan system provides for visual inspection of the fans and the associated backflow dampers and for duct systems that are not embedded in concrete. The containment spray system and the RHR sprays are designed such that active and passive components can be readily inspected to demonstrate system readiness. Pressure contained systems are inspected for leaks from pump seals, valve packing, flange joints, and relief valves. During operational testing of the containment spray pumps and RHR pumps, the portions of the systems subjected to pressure are inspected for leaks.

System design details are given in Section 6.2.

Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Compliance

The containment heat removal systems described in Section 6.2 are designed to permit periodic testing so that proper operation can be assured. In some cases whole systems can be operated for test purposes. In others, individual components are operated for functional tests so that plant operations are not disrupted.

The ice condenser contains no active components, other than the ice condenser doors, which are required to function during an accident condition. Samples of the ice are taken periodically and tested for boron concentration. The lower inlet door opening force is measured when the reactor is in the shutdown condition. The position of the lower inlet doors is monitored at all times.
Top deck door and intermediate deck doors are tested for operability during the shutdown condition. Air return fans and their associated backflow dampers are tested for operability while the reactor is shut down for refueling.

All active components of the containment spray system and the residual heat removal spray system are tested in place after installation. These spray systems receive initial flow tests to assure proper dynamic functioning. Further testing of the active components is conducted after component maintenance and in accordance with technical specifications. Air test lines, located upstream of the spray isolation valves, are provided for testing to assure that spray nozzles are not obstructed. Testing of transfer between normal and emergency power supplies is also conducted.

Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Compliance

The Shield Building, surrounding the primary containment, serves as a secondary containment. During accident conditions prior to containment isolation, primary and secondary containment purge exhaust is processed by the containment purge system filters prior to release to the atmosphere. The emergency gas treatment system (Section 6.2) maintains this secondary containment at a negative pressure during the entire post-accident period. The emergency gas treatment system also collects and processes the secondary containment atmosphere. After processing, the portion of this processed air necessary to assure a negative pressure is exhausted through the Shield Building exhaust vent. The remainder is recirculated and distributed in the secondary containment.

The Auxiliary Building serves to collect any equipment leakage during the recirculation of containment sump water. The Auxiliary Building ventilation system (Section 9.4) is isolated by the containment Phase A isolation signal. The Auxiliary Building gas treatment system (Section 9.4) then maintains the building at a negative pressure and processes any inleakage prior to release to the environment.

Post-accident hydrogen control within the containment is provided by the hydrogen mitigation system (Section 6.2.5). Distribution of the atmosphere within the containment is provided by the air return fan system (Section 6.8). The air return fan system also takes suction from each compartment to prevent stagnation and excessive accumulation of hydrogen.

Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

Compliance

The emergency gas treatment system (Section 6.2) filtration train and fans and the containment purge filters (Section 9.4.6) are located in the Auxiliary Building and are designed to facilitate inspections. The dampers that control recirculation and exhaust of the emergency gas treatment system effluent are located inside the Shield Building and may be inspected during reactor shutdown.

The entire Auxiliary Building gas treatment system (Section 9.4.3) is located in the Auxiliary Building and is designed to facilitate inspection.

Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Compliance

The containment purge system (Section 9.4) is designed to permit testing to assure leaktightness of the filter trains; functional testing to assure operability of containment isolation valves; and performance testing to assure filter efficiency and to demonstrate the isolation valve closure in response to the accident mode isolation signal.

The emergency gas treatment system (Section 6.2) is designed to permit testing to assure leaktightness of the filtration trains; functional testing to assure operability of the fans, dampers, and instrumentation; and performance testing to assure overall operability of the system and to demonstrate the proper alignment of the system to the accident unit.

The Auxiliary Building gas treatment system (section 9.4) is designed to allow testing to assure the pressure and leaktightness of the filters, adsorbers, and the filtration train housing to assure the operability of the fans and dampers; and to assure the operability of the system as a whole. The system design will permit testing of the actuation signals, the isolation of the normal ventilation system, and the proper alignment of dampers.

The hydrogen mitigation system (Section 6.2.5) is designed to allow testing to assure the operability of the manual controls that place the systems in operation. The system is designed to permit, under conditions as close to design as practical, the operability of the system as a whole.

Criterion 44 - Cooling Water

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Compliance

A Seismic Category I Component Cooling System (CCS) (Section 9.2) is provided to transfer heat from the reactor coolant system reactor support equipment and engineered safety equipment to a Seismic Category I Essential Raw Cooling Water (ERCW) system (Section 9.2).

The CCS serves as an intermediate system and thus a barrier between potentially or normally radioactive fluids and the river water which flows in the ERCW system.

The CCS consists of two independent engineered safety trains, each of which is capable of serving all required loads under normal or accident conditions that are important to safety.

In addition to serving as the heat sink for the CCS, the ERCW system is also used as heat sink for the containment through use of the containment spray heat exchangers, and engineered safety equipment through use of compartment and space coolers. The ERCW system consists of two independent trains, each of which is capable of providing all necessary heat sink requirements. The ERCW system transfers heat to the ultimate heat sink (Section 9.2).

Electric power is discussed in Chapter 8.

Criterion 45 - Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Compliance

The integrity and capability of the component cooling water system (Section 9.2) and essential raw cooling water system (Section 9.2) will be monitored during normal operation by the Surveillance Instruction Program. Nonsafety related systems may be isolated temporarily for inspection. All major components will be visually inspected on a periodic basis.

The component cooling and essential raw cooling water pumps are arranged such that any pump may be isolated for inspection and maintenance while maintaining full plant operational capabilities.

Criterion 46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Compliance

The cooling water systems will be pressurized during plant operations; thus, the structural and leaktight integrity of each system and the operability and performance of their active components will be continuously demonstrated. In addition, normally idle portions of the piping system and idle components will be tested during plant shutdown. The emergency functions of the systems will be periodically tested out to the final actuated device in accordance with the technical specifications.

For details, see the discussions on electric power (Chapter 8), component cooling water (Section 9.2), essential raw cooling water (Section 9.2), and instrumentation and controls (Chapter 7).

3.1.2.5 Reactor Containment

Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Compliance

The containment structure, including access openings and penetrations, is designed with sufficient conservatism to accommodate, without exceeding the design leakage rate, the transient peak pressure and temperature associated with a postulated reactor coolant piping break up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment design consists of a freestanding steel containment vessel and a separate outer reinforced concrete shield wall and roof. The ice condenser concept is used for energy absorption during a LOCA. The annular space between the containment vessel and the exterior shield wall forms a double barrier to fission products and is maintained at less than atmospheric pressure. The ice condenser, which is located inside the steel containment and consists of a suitable quantity of borated ice in a cold storage compartment, provides rapid energy absorption to maintain the containment vessel design pressure at a low level and to reduce the peak duration, thus reducing the potential for escape of fission products from the primary containment vessel.

The functional design of the containment is based upon the following assumptions and conditions:

1. A design basis blowdown energy and mass release.

2. Secondary energy released by safety injection.

3. Carryover energy from zirconium-water reaction.

4. Decay heat from the reactor at rated power.

5. The single failure criterion is accommodated.

The internal design pressure of the containment is greater than the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical double-ended severance of the largest reactor coolant pipe. The design pressure is not exceeded during any subsequent long-term pressure transient.

Refer to Section 3.8 for a description of containment, and to Section 6.2 for design basis details.

Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Compliance

The containment vessel and its penetration sleeves meet the material, design and technical process requirements of ASME Boiler and Pressure Vessel Code, Section III, Class B. Charpy V-notch impact tests were made of the containment vessel material (ASTM A 516, Grade 70) 5/8 inch and greater, weld deposit, and the base metal weld heat affected zone employing a test temperature at least 30° F below minimum service temperature in accordance with ASME Code, Paragraph N-1210. This test measured the ductile to brittle transition with allowable values for energy absorption given in Tables N-421 and N-422. It insures that the material used will not behave in a brittle manner and that rapidly propagating fracture is minimized. The containment boundary design considered uncertainties in material properties, residual, steady-state and transient stresses, and material flaws along with conservative allowable stress levels for all stressed elements of the containment boundary. All material was examined for flaws that would adversely affect the performance of the material in its intended location. See Section 6.2 for further details.

Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Compliance

The reactor containment design permitted overpressure strength testing during construction and permits preoperational integrated leakage rate testing at containment design pressure and at reduced pressure, in accordance with Appendix J, 10 CFR 50. The reactor containment and other equipment which may be subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. All equipment which may be subjected to the test pressure is either vented to the containment, removed from the containment during the test, or designed to withstand the containment design pressure without damage.

The preoperational integrated leak tests at peak pressure verify that the containment, including the isolation valves and the resilient penetration seals, leaks less than the allowable value of 0.25 weight percent per day at peak pressure.

Details concerning the conduct of periodic integrated leakage rate tests are in Section 6.2.

Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Compliance

The reactor containment and the containment isolation system (Section 6.2) are designed so that:

1. Integrated leak rate tests can be run during plant lifetime (see compliance to Criterion 52).

2. Visual inspections can be made of all important areas, such as penetrations.

3. An appropriate surveillance program can be maintained (see Section 6.2).

4. Periodic testing at containment design pressure of the leaktightness of isolation valves and penetrations which have resilient seals and expansion bellows is possible.

5. The operability of the containment isolation system can be demonstrated periodically.

In testing locally the resilient seals and expansion bellows leakages, the guidelines for Type B tests in Appendix J to 10 CFR 50 will be followed.

Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundance, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Compliance

Containment isolation features are classified as Seismic Category I. These components required quality assurance measures which enhance reliability. The containment isolation design provides for a double barrier at the containment penetration in those fluid systems that are not required to function following a design basis event.

All piping systems penetrating the containment, in so far as practical, have been provided with test vents and test connections or have other provisions to allow periodic leak testing as required. Section 6.2.4.4 has further details on testing.

See Section 6.2.4 for general containment isolation details and Section 6.2.4.3 for exceptions to General Design Criteria 54, 55, 56, and 57.

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical, and automatic isolation valves shall be designed to take the position that provides greater safety upon loss of actuating power.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Compliance

The reactor coolant pressure boundary is defined as those piping systems and components which contain reactor coolant at design pressure and temperature. With the exception of the reactor coolant sampling lines, the entire reactor coolant pressure boundary, as defined above, is located entirely within the containment structure. All sampling lines are provided with remotely operated valves for isolation in the event of a failure. These valves also close automatically on a containment isolation signal.

All other piping and components which may contain reactor coolant are low pressure, low temperature systems which would yield minimal environmental doses in the event of failure.

The sampling system and low-pressure systems are described in Section 9.3. An analysis of malfunctions in these systems is included in Chapter 15.

Criterion 56- Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical, and automatic isolation valves shall be designed to take the position that provides greater safety upon loss of actuating power.

Compliance

At least two barriers are provided between the atmosphere outside the containment and the containment atmosphere, the reactor coolant system, or closed systems which are assumed vulnerable to accident forces.

Redundant valving is provided for piping that is open to the atmosphere and to the containment atmosphere. Additional details can be found in Section 6.2.

Criterion 57 - Closed Systems Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Compliance

Those lines that penetrate the containment, that do not communicate with either the reactor coolant pressure boundary or the containment atmosphere, and that are not affected by LOCA forces, are defined as closed systems. All lines penetrating the containment are designed to meet this GDC.

See Section 6.2.4 for a discussion of containment isolation valves.

3.1.2.6 Fuel and Radioactivity Control

Criterion 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Compliance

Liquid, gaseous, and solid radioactive waste processing equipment is provided. The principles of filtration, demineralization, evaporation, solidification and storage for decay are utilized as described in Chapter 11. Process monitoring is provided to control this equipment and regulate releases to the environment as described in Section 11.4.

Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Compliance

The spent fuel pool and cooling system, fuel handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions.

1. Components are designed and located such that appropriate periodic inspection and testing may be performed.

2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Section 12.1.

3. Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems.

4. The spent fuel cooling systems provide cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed for testability to permit continued heat removal.

5. The spent fuel pool is designed such that no postulated accident could cause excessive loss of coolant inventory.

Radioactive waste treatment systems are located in the Auxiliary Building, which contains or confines leakage under normal and accident conditions.

The Auxiliary Building Gas Treatment System (ABGTS) includes charcoal filtration which can be used to minimize radioactive material releases associated with a postulated spent fuel handling accident. The ABGTS system is not required to mitigate the consequences of a spent fuel handling accident.

Fuel storage and handling and spent fuel storage at the ISFSI are discussed in section 9.1, and radioactive waste management in Chapter 11.

Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Compliance

The restraints and interlocks provided for safe handling and storage of new or spent fuel are discussed in Section 9.1.

The center-to center distance between adjacent spent fuel assemblies together with the use of fixed Boral neutron absorber panels in the storage racks and burnup credit administrative controls on fuel assembly placement are sufficient to ensure subcriticality, even if unborated water is used to fill the spent fuel storage pool. Credit for borated water is permitted to maintain subcriticality for inadvertent misplacement of a fuel assembly, e.g., loading of a fresh fuel assembly in a storage cell designated for exposed fuel or placement outside of and adjacent to a rack module.

Layout of the fuel handling area is such that the spent fuel casks will never be required to traverse the spent fuel storage pool during removal of the spent fuel assemblies.

Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling area (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Compliance

Failure in the spent fuel cooling system will result in control room annunciation and local temperature indication.

High radiation in the spent fuel storage (refuel floor) area will produce the following alarms:

1. Main Control Room alarm from the spent fuel pool area accident monitors.
2. Local and Main Control Room alarms from the refuel floor area monitor.
3. Local alarm from a portable continuous air monitor located on the refuel floor.
4. Main Control Room alarm from the Auxiliary Building Ventilation Radiation Monitor for high airborne radiation in the spent fuel storage area.

High radiation in the waste packaging area will result in the following alarms:

1. Main Control Room and local alarms from the waste packaging area monitor.
2. Local alarms from a portable continuous air monitor located in the area.
3. Main Control Room alarm from the Auxiliary Building Ventilation Radiation Monitor for high airborne radiation in the waste packaging area.

See Sections 9.1 and 12.3 and Chapter 11 for further details.

Criterion 64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Compliance

The facility contains means for monitoring the containment atmosphere and all other important areas during both normal and accident conditions to detect and measure radioactivity which could be released under any conditions. The monitoring system includes area gamma monitors, atmospheric monitors and liquid monitors with full indication in the control room. Alarms are provided to warn of high radioactivity.

Chapter 11 discusses the process and effluent radiation monitoring systems. Chapter 12 discusses the area and airborne radiation monitoring systems.

REFERENCES

None

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 Seismic Classifications

The Watts Bar Nuclear Plant structures, systems, and components which perform a primary safety function have been designed to remain functional in the event of a Safe Shutdown Earthquake (SSE). These structures, systems, and components, designated as Seismic Category I, are those necessary to assure:

1. The integrity of the reactor coolant pressure boundary.

2. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to or in excess of the guideline exposures of 10 CFR Part 100.

These structures, systems, and components are classified in accordance with Regulatory Guide 1.29 unless exception is taken in detailed classification information provided in other sections of the UFSAR, such as Table 3.2-2a, 3.2-6, etc.

Piping, pumps, valves, and other fluid system components which must retain limited structural integrity because their failure could jeopardize to an unacceptable extent the achievement of a primary safety function, because they form an interface between Seismic Category I and non-Seismic Category I plant features, or because they perform a secondary safety function, are designated by TVA as Seismic Category I(L) (i.e., limited requirements). Those fluid containing elements which are included in Seismic Category I(L) are seismically qualified to meet the intent of Position 2 of Regulatory Guide 1.29. For Unit 2 the Seismic Category 1(L) is subdivided into categories 1L(A) Pressure boundary and position retention and 1L(B) position retention.

Where portions of mechanical systems are Category I or I(L) and the remaining portions not seismically classified, the systems have been seismically qualified to a terminating anchor (or other appropriate analysis problem termination) beyond the defined boundary such as a valve, thus meeting Position 3 of Regulatory Guide 1.29.

All Category I safety-related structures, and portions of mechanical and electrical systems and components are listed in Tables 3.2-1, 3.2-2, 3.2-2a, 3.2-2b and 3.2-3. Those Category I(L) portions of mechanical systems are also listed in Table 3.2-2.

3.2.2 System Quality Group Classification

Fluid system components for the Watts Bar Nuclear Plant that perform a primary safety function are identified by TVA Classes A, B, or C (see Section 3.2.2.7 for HVAC Safety Classifications). These piping classes are assigned to fluid systems based on the ANS Safety Classes 1, 2a, and 2b, respectively, which are assigned to nuclear power plant equipment per the August 1970 Draft of ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." Fluid system components whose postulated failure would result in potential offsite doses that exceed 0.5 Rem to the whole body, or its equivalent to any part of the body, are identified as TVA Class D and are based on ANSI N18.2 (Aug., 1970 draft) Safety Class 3 and Regulatory Guide 1.26. The TVA piping classification system for WBNP does not conform strictly to the guidance of Regulatory Guide 1.26 (which was not in effect on the docket date for the Construction Permit). The ANS safety classification of each component has been considered in the various aspects of design, fabrication, construction, and operation.

3.2.2.1 Class A

Class A applies to reactor coolant pressure boundary components whose failure could cause a loss of reactor coolant which would not permit an orderly reactor shutdown and cool down assuming that makeup is only provided by the normal makeup system. Branch piping 3/8 inch inside diameter and smaller, or protected by a 3/8 inch diameter or smaller orifice, is exempted from Class A requirements. The branch piping for the pressurizer steam space instrumentation nozzles (0.83 inch inside diameter, or smaller) is also exempted from Class A requirements.

The components which are within the Reactor Coolant Pressure Boundary (RCPB) and meet all the following requirements may be classified as TVA Class G:

1. Piping and associated components in the RCPB which penetrate containment excluding the actual penetration and its associated components.

2. Piping and associated components which perform no primary safety function.

3. Piping and associated components which are isolated by a normally closed valve off a line in the RCPB that meets the exclusion requirements of 10 CFR Part 50.55a paragraphs (c) (1) and (c) (2). An example would be the ECCS check valve leak test lines.

3.2.2.2 Class B

Safety Class B applies to those components of safety systems necessary to fulfill a system safety function. The classification is specifically applicable to containment and to components of those safety systems, or portions thereof, through which reactor coolant water flows directly from the reactor coolant system or the containment sump.

3.2.2.3 Class C

Class C applies to components of those safety systems that are important to safe operation and shutdown of the reactor but that do not recirculate reactor coolant.

3.2.2.4 Class D

Class D applies to components not in TVA Class A, B, or C whose failure would result in release to the environment of gaseous radioactivity normally held up for radioactive decay. This is being interpreted as those portions of systems whose postulated failure would result in calculated potential offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body.

3.2.2.5 Relationship of Applicable Codes to Safety Classification for Mechanical Components

The applicable codes used for the design, material selection, and inspection of components for the various safety classes are shown in Table 3.2-4. The applicable TVA classification and ANS Safety Classification for each of the fluid systems are tabulated in Table 3.2-2. TVA classifications are also delineated on flow diagrams which have been included as figures in those
sections of the UFSAR where the systems are discussed in detail.

3.2.2.6 Nonnuclear Safety Class (NNS)

Components that are used in Seismic Category I structures whose failure would not result in a release of radioactive products and are not required to function during an accident or malfunction within the reactor coolant pressure boundary have been assigned TVA Classifications G or K. Since these components complement components having a primary safety function during normal operation and may be in close proximity to them, they are seismically qualified as Seismic Category I(L) to the extent necessary to prevent an unacceptable influence on Safety Class equipment during a seismic event. Thus the minimum capability of primary system components is not compromised by the failure of a Class G or K component during a seismic event. Components which are assigned to TVA Class H or L, located inside Seismic Category I structures, are also designed as Seismic Category I(L). The applicable codes, along with the seismic classifications used for the design of the components covered by these classifications, are shown in Table 3.2-5. TVA Class P is assigned to specific sense lines located (in part or totally) in a non-seismic area.

3.2.2.7 Heating, Ventilation and Air Conditioning (HVAC) Safety Classification

Those portions of the HVAC Systems which are safety related have been assigned TVA classifications and have been designed to Seismic Category I and I(L) specifications as applicable. All equipment, components, duct work, etc., in the August 1970 Draft ANSI N18.2 Safety Classes 2a and 2b perform primary safety functions and are designed to Seismic Category I requirements, except as exempted in Table 3.2-6. Portions of systems not performing a safety function may need a degree of seismic qualification because their failure could produce an unacceptable influence on the performance of safety functions. These are designed to Seismic Category I(L) requirements. The applicable codes along with the seismic qualifications used for the design of the HVAC ducting are shown in Table 3.2-6. See Sections 3.7.3.17 and 3.7.3.18 for details of seismic analysis and design of HVAC duct and duct support systems.

3.2.3 Code Cases and Code Editions and Addenda

3.2.3.1 TVA Design and Fabrication

The Code of Record of Section III of the ASME Code applied to systems within TVA's scope is the 1971 Edition with Addenda through Summer 1973. The use of later Edition and Addenda, as permitted by paragraph NA-1140 of the ASME Code, is controlled to ensure the following:

a. Later Edition and Addenda used has been accepted by the NRC through incorporation by reference in 10 CFR 50.55a.

b. Related requirements necessary to support use of later Edition and Addenda are implemented in accordance with NA-1140.

c. Code Cases used have been accepted by the NRC through incorporation by reference in Regulatory Guide 1.84.

d. Additional requirements added by Regulatory Guide 1.84 are implemented.

A listing of Code Cases and provisions of later Code editions and addenda which have been used for design and fabrication is given in Table 3.2-7. A similar listing of Code Cases and provisions of later Code editions and addenda used in analysis of fluid systems is given in Section 3.7.3.8.1. Another similar listing for the RCS is given in Section 5.2.1.4. Code cases and provisions of later Code Editions and Addenda associated with Inservice Inspection and Inservice Testing are found in the Inservice Inspection and Inservice Testing program. Exceptions to the system classification Code requirements associated with Generic Letter 89-09 may be found in notes on the flow diagrams.

3.2.3.2 Purchased Materials and Components

The Code of Record for components ordered by TVA is determined in accordance with 10 CFR 50.55a, footnote 5. Material ordered by TVA and supplied with certification to a later Edition and Addenda is controlled by a comparison of the Edition and Addenda to which it is certified to the Code of Record applicable to the application in which it is used. Deviations from the applicable Code of Record are reconciled prior to use of the material.

Material procured prior to the initiation of the Acceptable Suppliers List (ASL) program (approximately May 1978) has been addressed through an NRC approved alternative to the ASME Code paragraph NA-3451(a)[1]. Material

- procured as ASTM material,
- installed or to be installed in an ASME system,
- whose proof of survey or qualification by TVA of the manufacturer's quality assurance program at the time of procurement cannot be retrieved, and
- whose material specification is identical to the requirements of ASME Section II as stated by the ASME material specification,

is acceptable for use assuming all other attributes of the material and the documentation conform to ASME Code requirements.

REFERENCES

1. Letter from B. D. Liaw, NRC, to O. D. Kingsley, TVA, dated March 15, 1990, "NRC Inspection Report Nos. 50-390/90-02 and 50-391/90-02".

2. Letter from Frederick J. Hebdon, NRR, to Mark O. Medford, TVA, dated February 22, 1993.

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TABLE 3.2-1

CATEGORY I STRUCTURES

1. Reactor Building (Shield Building, Steel Containment Vessel, and Interior Concrete)
2. Auxiliary - Control Building
    a. Auxiliary Building portion
    b. Additional Equipment Building portion
    c. Control bay portion
    d. Waste packaging area
3. Condensate Demineralizer Waste Evaporator Building
4. Class 1E Electrical Systems Structures (Manholes, Handholes, and Conduit Banks)
5. Diesel Generator Building
6. ERCW Pipe Tunnels and RWST Foundations
7. ERCW Structures
8. North Steam Valve Room
9. Intake Pumping Station and Retaining Walls
10. Deleted
11. Refueling Water Storage Tank (RWST).
12. Underground Barrier.
13. ERCW Standpipe Structures I and II and ERCW Discharge Overflow Structure.

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Reactor Vessel	W	A	(13)	X	C	X	I
Full Length CRDM Housing	W	A	III-1	X	C	X	I
CRDM Head Adapter Plugs	W	A	III-1	X	C	X	I
Steam Generators (Tube Side)	W	A	III-1	X	C	X	I
(Shell Side)	W	A(17)	III-1	X	C	X	I
Pressurizer	W	A	III-1	X	C	X	I
Reactor Coolant Pipe	W	A	III-1	X	C	X	I
Reactor Coolant Fittings	W	A	III-1	X	C	X	I
Reactor Coolant Fabricated Piping	W	A	III-1	X	C	X	I
Reactor Coolant Crossover Legs	W	A	III-1	X	C	X	I
Reactor Coolant Thermowell	W	A	III-1	X	C	X	I
Thimble Guide Tubing	W	A	III-1	X	C	X	I
Thimble Guide Couplings	W	A	III-1	X	C	X	I
Incore Instrument Thimble Assembly	W	B	III-2	X	C	X	I
Loop Bypass Line	W	A	III-1	X	C	X	I
Pressurizer Safety Valves	W	A	III-1	X	C	X	I
Power Operated Relief Valves	W	A	III-1	X	C	X	I
Pressurizer Relief Tank	W	G	VIII	X	C	P	I(L)

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TABLE 3.2-2 (Sheet 2 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Reactor Coolant Pump	W	A	III-1	X	C	X	I
RCP Casing	W	A	III-1	X	C	X	I
Main Flange	W	A	III-1	X	C	X	I
Thermal Barrier	W	A	III-1	X	C	X	I
Thermal Barrier Heat Exchanger	W	C	(14)	X	C	X	I
No. 1 Seal Housing Bolts	W	A	III-1	X	C	X	I
Upper Seal Housing	W	B	III-1	X	C	X	I
Pressure Retaining Bolting	W	A	III-1	X	C	X	I
RCP Motor	W	2b	NEMA-MG1	X	C	-	I
Motor Rotor	W	2b	NA	X	C	-	I
Motor Shaft	W	2b	NA	X	C	-	I
Shaft Coupling	W	2b	NA	X	C	-	I
Spool Piece	W	2b	NA	X	C	-	I
Flywheel	W	2b	NA	X	C	-	I
Bearing (Motor Upper Thrust)	W	2b	NA	X	C	-	I
Motor Bolting	W	2b	NA	X	C	-	I
Motor Stand	W	2b	NA	X	C	-	I
Motor Frame	W	2b	NA	X	C	-	I

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Upper Oil Reservoir (UOR)	W	2b	NA	X	C	P	I
Upper Oil Cooler	W	2b	(14)	X	C	P	I
Lube Oil Piping	W	2b	(14)	X	C	P	I
Lower Oil Reservoir (LOR)	W	2b	NA	X	C	P	I
LOR Cooling Coil - CC (supports only)	W	2b	NA	X	C	P	I
Motor Air Coolers (Unit 1)	W	-	-	X	C	-	I(L)B Note 12
Motor Air Coolers (Unit 2)	W	-	-	X	C	-	I(L)A Note 12
Safety Injection System							
Safety Injection Pumps	W	B	III-2	X	AB	X	I
Accumulators (9)	W	B	III-2	X	C	P	I
Boron Injection Tank	W	B	III-2	X	AB	X	I
Refueling Water Storage Tank	T	B	III-2	X	O	P	I
Residual Heat Removal System							
RHR Pumps	W	B	III-2	X	AB	X	I
RHR Heat Exchangers (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I

TABLE 3.2-2 (Sheet 4 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Containment Spray System							
CS Pumps	T	B	III-2	X	AB	X	I
CS Heat Exchangers (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I
CS Nozzles	T	B	-	X	C	P	I
Primary Water Make-Up System							
Pump	T	G	ANSI B31.1	X	AB	-	I(L)
Tank Unit 1	T	G(22)	Note 25	X	O	-	I(L)
Tank Unit 2	T	G(26)	Note 26	X	O	-	I(L)
Chemical and Volume Control System Pumps							
Charging, Centrifugal	W	B	P&V-II	X	AB	X	I
Boric Acid Transfer	W	C	P&V-III	X	AB	-	I
Heat Exchangers							
Regenerative	W	B	III-2	X	C	X	I
Letdown (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I
Excess Letdown (Tube)	W	B	III-2	X	C	X	I
(Shell)	W	B	III-2	X	C	P	I

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TABLE 3.2-2 (Sheet 5 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Seal Water (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I
Tanks							
Volume Control	W	B	III-2	X	AB	X	I
Boric Acid	W	C	III-3	X	AB	P	I
Boric Acid Batching	W	G	VIII	X	AB	-	I(L)
Chemical Mixing	W	G	VIII	X	AB	-	I(L)
Resin Fill	W	G	VIII	X	AB	-	I(L)
Demineralizers							
Mixed Bed	W	D	III-3	X	AB	X	I
Cation	W	D	III-3	X	AB	X	I
Steam Generator Blowdown System							
SG Blowdown Isolation Valves	T	B	III-2	X	AB	P	I
SG Blowdown Heat Exchangers	T	-	VIII	-	TB	P	-
Flash Tank	T	-	VIII	-	TB	P	-
Compressed Air System							
Service & Control Air Subsystem							
Compressors	T	H	-	-	TB	-	-
Receiver Tanks	T	H	VIII	-	TB	-	-
Air Dryers	T	H	VIII	-	TB	-	-
Auxiliary Control Air Subsystem							

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Compressors	T	(19)	-	X	AB	-	I
Receiver Tanks	T	(19)	VIII	X	AB	-	I
Air Dryers	T	C	(14)	X	AB	-	I
Ice Condenser							
Ice Baskets	W	C	-	X	C	-	I
Lower Inlet Doors	W	C	-	X	C	-	I
Lattice Frames	W	C	-	X	C	-	I
Lattice Frame Columns	W	C	-	X	C	-	I
Lower Support Structure	W	C	-	X	C	-	I
Intermediate Deck Doors	W	C	-	X	C	-	I
Wall Panels	W	C	-	X	C	-	I
Floor Structures	W,T	C	-	X	C	-	I
Top Deck Doors	W	C	-	X	C	-	I
Air Handling Unit Supports	W	C	-	X	C	-	I
Top Deck Beams	W	C	-	X	C	-	I
Refrigeration System	W	-	-	X	C,AB	-	I(L)
Ice Machine	W	-	-	X	AB	-	I(L)
Ice Condenser Bridge Crane	W	-	-	X	C	-	I(L)
Floor Drain Gate	W	C	-	X	C	-	I
Containment Isolation System Valves	T	B	III-2	X	C,AB	X,P	I
Air Return Fans	T	(11)	AMCA IEEE	X	C	-	I

TABLE 3.2-2 (Sheet 7 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Component Cooling System							
Pumps	T	C	III-3	X	AB	P	I
Heat Exchangers (Tube)	T	C	III-3	X	AB	-	I
(Shell)	T	C	III-3	X	AB	P	I
Surge Tank	T	C	III-3	X	AB	P	I
Valve (Containment Isolation)	T	B	III-2	X	C	-	I
Valves	T	C	III-3	X	AB,C	-	I
Valves	T	G	B31.1	X	AB	-	I(L)
Valves	T	H	B31.1	-	CDWEB	-	
Seal Leakage Return Unit	T	L	-	X	AB	-	I(L)
Radioactive Waste Disposal System							
Tanks							
Laundry & Hot Shower	W	G	VIII	X	AB	X	I(L)
Chemical Drain	W	G	VIII	X	AB	X	I(L)
Reactor Coolant Drain	W	G	VIII	X	C	X	I(L)
Tritiated Drain Collector	W	G	VIII	X	AB	X	I(L)
Waste Condensate (See Note 23)	W	H	VIII	X	AB	X	I(L)
Spent Resin Storage	W	D	III-3	X	AB	X	I
Gas Decay	W	D	III-3	X	AB	X	I
Floor Drain Collector	W	G	VIII	X	AB	X	I(L)
CVCS Monitor	W	G	III	X	AB	P	I(L)
Cask Decontamination Collector	T	G	----	X	AB	P	I(L)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Pumps							
Reactor Coolant Drain Tank Pumps	W	G	B31.1	X	C	X	I(L)
Chemical Drain Tank Pump	W	G	B31.1	X	AB	X	I(L)
Laundry & Hot Shower Tank Pump	W	G	B31.1	X	AB	X	I(L)
Tritiated Equipment Drain Sump Pumps	W	G	B31.1	X	AB	X	I(L)
Waste Condensate Pumps (See Note 23)	W	H	B31.1	X	AB	X	I(L)
Tritiated Drain Collector Tank							
Discharge Pump	W	G	B31.1	X	AB	X	I(L)
Floor Drain Collector Tank Discharge Pump	W	G	B31.1	X	AB	X	I(L)
Aux. Condensate Demin Waste Evap Feed Pump (See Note 23)	W	G	B31.1	X	AB	X	I(L)
CVCS Monitor Tank Pump	W	G	B31.1	X	AB	P	I(L)
Cask Decon Collector Tank Pump	T	G	B31.1	X	AB	P	I(L)
Component							
Containment Pit Sump Pumps	W	G	B31.1	X	C	X	I(L)
AB Floor & Equip Drain Sump Pumps	W	G	B31.1	X	AB	P	I(L)
RB Floor & Equip. Drain Sump Pump	T	G	B31.1	X	C	X	I(L)
RB Floor & Equip Drain Pocket Sump Pump	T	G	B31.1	X	C	X	I(L)
AEB Floor & Equip. Drain Sump Pump	T	G	B31.1	X	AEB	P	I(L)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Miscellaneous							
Waste Gas Compressor Pkg.	W	D	III-3	X	AB	X	I
Waste Gas Filter	T	G	VIII	X	AB	X	I(L)
Nitrogen Manifold	T	G	-	X	AB	-	I(L)
Hydraulic Compactor	W	-	-	-	SB	P	-
Laundry Tank Basket Strainer	W	G	VIII	X	AB	P	I(L)
Cond Demin Waste Evaporator (See Note 23)	T	H	III-3	X	CDWEB	P	-
Fire Protection System							
Valves - Flood Mode & CI	T	C	III-3	X	C,AB,O,B,P	-	I
Valves - Balance of System	T	G	B31.1, UL/FM	X	C,AB,O,B,P,CB DB,SB,CDWEB	-	I(L)
Fire Pumps (vertical turbine)	T	C	III-3	X	O	-	I
Station Ventilation System							
Containment Ventilation							
Containment Purge							
Fans (excluding Inst. Rm. Fan)	T	(11)	AMCA	X	AB	-	I(L)
Filters	T	(11)	-	X	AB	X	I
Dampers	T	(11)	-	X	AB	X	I
Ductwork	T	(11)	SMACNA	X	C,AB	X	I/I(L) (See Note 20)

TABLE 3.2-2 (Sheet 10 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Upper Compartment, CRDM & Instrument Room Cooling							
Fan	T	-	AMCA	X	C	P	I(L)
Supply Air Filters	T	(11)	-	X	AB	-	I(L)
Upper Containment Coolers							
Coil Units (Unit 1)	T	-	-	X	C	P	I (See Note 12)
Coil Units (Unit 2)	T	-	-	X	C	P	I(L)A (See Note 12)
CRDM Coolers							
Coil Units (Unit 1)	T	-	-	X	C	P	I(L)A (See Note 12)
Coil Units (Unit 2)	T	-	-	X	C	P	I(L)B (See Note 12)
Instrument Room A/C Water Chiller							
Coil Units (Unit 1)	T	(11)	-	X	AB	P	I(L)A (See Note 12)
Coil Units (Unit 2)	T	-	-	X	AB	P	I(L)B (See Note 12)

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TABLE 3.2-2 (Sheet 11 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Lower Compartment Cooling Units							
Fans	T	(11)	AMCA	X	C	-	I
Motors	T	(11)	IEEE	X	C		I
Coils	T	C	III-3	X	C	P	I (See Note 12)
Backdraft Dampers	T	(11)	ASME	X	C	-	I
Instrumentation/Controls	T	(11)	IEEE	X	C/CB	-	I
Ductwork/Accessories	T	(11)	SMACNA	X	C	P	I
Auxiliary Bldg. Ventilation							
Fan/Coil Units	T	(11)	AMCA	X	AB	-	I(L)
Filters	T	(11)	-	X	AB	-	I(L)
ESF Room Coolers	T	(11)	-	X	AB	P	I
Auxiliary Board Rooms Air-conditioning System	T	(11)	AMCA,ARI	X	AB	-	I
Shutdown Board Rooms Air-conditioning System	T	(11)	AMCA,ARI	X	AB	-	I
Other Air Conditioning Systems	T	(11)	AMCA,ARI	X	AB	-	I(L)
Control Bldg. Ventilation							
Fan	T	(11)	-	X	CB	-	I
Filters	T	(11)	-	X	CB	X	I
Air Conditioning Unit (MCR)	T	(11)	AMCA,ARI	X	CB	-	I
Air Conditioning Unit (Elec. Bd. Rm.)	T	(11)	AMCA,ARI	X	CB	-	I
RB Inst Rm Air Conditioning System	T	(11)	AMCA,ARI	X	RB	-	I(L) (See Note 21)

TABLE 3.2-2 (Sheet 12 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Diesel Bldg. Ventilation							
Exhaust System	T	(11)	AMCA	X	DB	-	I
Battery Hood Exhaust System	T	(11)	AMCA	X	DB	-	I
Elec Board Room Exhaust System Fans	T	(11)	AMCA	X	DB	-	I
Main Steam System							
Relief Valves	T	B	III-2	X	AB	-	I
Safety Valves	T	B	III-2	X	AB	-	I
MSIVs	T	B	III-2	X	AB	-	I
Isolation Bypass Valves	T	B	III-2	X	AB	-	I
Feedwater System							
MFIVs	T	B	III-2	X	AB	-	I
Auxiliary Feedwater System							
Auxiliary Feedwater Pumps							
Motor Driven	T	C	III-3	X	AB	-	I
Steam Turbine Drive	T	C	III-3	X	AB	-	I
Steam Dump Systems							
Turbine Bypass	W	-	-	-	TB	-	-

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Spent Fuel Pit							
Spent Fuel Pit Heat Exch.							
(Tube)	W	C	III-3	X	AB	X	I
(Shell)	W	C	III-3	X	AB	X	I
Spent Fuel Pit Pump	W	C	III-3	X	AB	X	I
Spent Fuel Pit Filter	W	G	VIII	X	AB	X	I(L)
Spent Fuel Pit Demineralizer	W	G	VIII	X	AB	X	I(L)
Spent Fuel Pit Strainer	W	G	-	X	AB	X	I(L)
Spent Fuel Pit Skimmer Pump	W	G	-	X	AB	X	I(L)
Spent Fuel Pit Skimmer Strainer Assembly	W	G	-	X	AB	X	I(L)
Spent Fuel Pit Skimmer Filter	W	G	VIII	X	AB	X	I(L)
Purification Pumps	T	G	ANSI B31.1	X	AB	X	I(L)
Purification Filter	T	G	ANSI B31.1	X	AB	X	I(L)
Fuel Handling System							
Refueling Machine	W	-	-	X	C	-	I(L)
Reactor Vessel Head Lifting Device	W	-	-	X	C	-	-
Reactor Internals Lifting Device	W	-	-	X	C	-	-

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Spent Fuel Pit Bridge & Hoist	W	-	-	X	AB	-	I(L)
Rod Cluster Cont. Handling Fixture	W	-	-	X	-	-	-
Reactor Vessel Stud Tensioner	W	-	-	X	-	-	-
Spent Fuel Handling Tool	W	-	-	X	-	-	-
Fuel Transfer System							
Fuel Transfer Tube & Flange	W	B	-	X	C,AB	P	I
Conveyor System & Controls	W	-	-	X	C,AB	P	-
New Fuel Storage Racks	W	-	-	X	AB	-	I
Spent Fuel Storage Racks	T	-	-	X	AB	X	I
Emergency Diesel Fuel Oil System							
To 7 Day Tanks							
Transfer Pumps	T	G	B31.1	X	DB	-	I(L)
Fuel Oil Tanks (7 Day)	T	I	VIII	X	DB	-	I
Raw Cooling Water System							
Pumps	T	-	-	-	O	-	-
Strainers	T	-	-	-	TB	-	-
Sampling System							
Sample Heat Exchanger	T	-	VIII	-	AB	X	-
Sample Vessel	T	-	VIII	-	AB	X	-
Delay Coil	T	B	III-2	X	C	X	I

TABLE 3.2-2 (Sheet 15 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Equipment Decontamination System							
Pump	W	-	-	-	AB	-	-
Tank	T	-	D100	-	AB	-	-
Filters							
Reactor Coolant	W	B	III-2	X	AB	X	I
Seal Water Return	W	B	III-2	X	AB	X	I
Seal Water Injection	W	B	III-2	X	AB	X	I
Boric Acid	W	C	III-3	X	AB	-	I
Miscellaneous							
Letdown Orifices	W	B	III-2	X	C	X	I
Boric Acid Blender	W	C	III-3	X	AB	-	I
Boron Recovery System (See Note 24)							
Pumps							
Holdup Tank Recirc.	W	D	P&V-III	X	AB	X	I
Gas Stripper Feed	W	D	P&V-III	X	AB	X	I
Monitor Tank	W	G	VIII	X	AB	P	I(L)
Tanks							
Holdup	T	D	III-3	X	AB	X	I
Monitor	T	G	VIII	X	AB	P	I(L)

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TABLE 3.2-2 (Sheet 16 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Demineralizers							
Evaporator Feed Ion Exch	W	G	VIII	X	AB	X	I(L)
Evaporator Condensate	W	G	VIII	X	AB	X	I(L)
Filters							
Evaporator Feed Ion Exch	W	G	VIII	X	AB	X	I(L)
Evaporator Condensate	W	G	VIII	X	AB	X	I(L)
Concentrates	W	G	VIII	X	AB	X	I(L)
Emergency Gas Treatment System							
Fans	T	(11)	AMCA	X	AB	P	I
Filters	T	(11)	-	X	AB	P	I
Moisture Separator	T	(11)	-	X	AB	P	I
Dampers	T	(11)	-	X	AB,C	P	I
Ducting	T	(11)	-	X	AB,C	P	I
Auxiliary Bldg. Gas Treatment System							
Fans	T	(11)	AMCA	X	AB	P	I
Filters	T	(11)	-	X	AB	P	I
Moisture Separator	T	(11)	-	X	AB	P	I
Dampers	T	(11)	-	X	AB,C	P	I
Ducting	T	(11)	-	X	AB,C	P	I

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Essential Raw Cooling Water System Pumps	T	C	III-3	X	P	-	I
ERCW Pump Motor Bearing							
Cooling Coils	T	(18)	-	X	P	-	I
Containment Isolation Valves	T	B	III-2	X	C	-	I
Valves	T	C	III-3	X	AB	-	I
Valves (yard)	T	C	III-3	X	B	-	I
Valves (Station Air Compressor)	T	H	B31.1	-	TB	-	-
Valves (Auxiliary Air Compressor)	T	C	III-3	X	AB	-	I
Valves (Screen Wash Supply)	T	(15)	B31.1	X	P	-	I(L)(15)
Screen Wash Pumps	T	(15)	B58.1	X	P	-	I(15)
Automatic Backwashing Strainers	T	C	III-3	X	P	-	I
Valves (Discharge Header Air Release & Piping)	T	K	C512/B31.1	X	AB	-	I(L)
Flood Mode Boration Makeup System							
Aux. Boration Makeup Tank	T	C	III-3	X	AB	X	I
Aux. Charging Booster Pumps	T	H	B31.1	X	AB	X	I(L)
Flood Mode Boration Demineralizer	T	H	B31.1	X	AB	X	I(L)
Flood Mode Boration Filters	T	H	B31.1	X	AB	X	I(L)
Aux. Charging Pump	T	H	B31.1	X	AB	X	I(L)
Valves	T	H	B31.1	X	AB	X	I(L)
Valves	T	B	III-2	X	AB	X	I

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TABLE 3.2-2 (Sheet 18 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Valves	T	C	III-3	X	AB	X	I
Flood Mode Boration Filters	T	H	B31.1	X	AB	X	I(L)
Aux. Charging Pump	T	H	B31.1	X	AB	X	I(L)
Valves	T	H	B31.1	X	AB	X	I(L)
Valves	T	B	III-2	X	AB	X	I
Valves	T	C	III-3	X	AB	X	I

Notes:

(1) T = Tennessee Valley Authority

W = Westinghouse

(2) A = TVA Safety Class A

B = TVA Safety Class B

C = TVA Safety Class C

D = TVA Safety Class D

G = TVA Safety Class G

H = TVA Safety Class H

1, 2a, 2b, or 3 = ANS N18.2 Safety Class

I = Seismic Category I, part of structure

K = TVA Safety Class K

P = TVA Safety Class P

(3) The code class listed for an item is the minimum required. An item may have been obtained to a higher code level than that listed.

III = ASME Boiler and Pressure Vessel Code - Section III

III-1 = ASME Boiler and Pressure Vessel Code - Section III, Class 1

III-2 = ASME Boiler and Pressure Vessel Code - Section III, Class 2

III-3 = ASME Boiler and Pressure Vessel Code - Section III, Class 3

IIIa9 = ASME Boiler and Pressure Vessel Code - Section III, Article 9 "Protection Against Overpressure"

VIII = ASME Boiler and Pressure Vessel Code - Section VIII

P&V-I = ASME Code for Pumps and Valves for Nuclear Power, Class I

P&V-II = ASME Code for Pumps and Valves for Nuclear Power, Class II

P&V-III = ASME Code for Pumps and Valves for Nuclear Power, Class III

D100 = American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage, AWWA, D100

B31.1 = ANSI B31.1 1973 Edition through summer 1973 Addenda

ACI = American Concrete Institute

AMCA = Air Moving and Conditioning Association

ARI = Air Conditioning and Refrigeration Institute

HIS = Hydraulic Institute Standards

IEEE = Institute of Electrical and Electronics Engineers

NFPA = National Fire Protection Association

B58.1 = ANSI B58.1 Vertical Turbine Pumps

B73.1 = ANSI B73.1M Horizontal end Suction Centrifugal Pumps

UL/FM = Underwriters Laboratory or Factory Mutual

SMACNA = Sheet Metal and Air Conditioning Contractors National Association

C512 = American Waterworks Association, Air-Release, Air/Vacuum, and Combination Air Valves for Waterworks Services ANSI/AWWA, C512.

(4) Quality assurance required:

X = Yes, - = No

(5) C=Containment

AB =Auxiliary Building

AEB = Additional Equipment Building

CB = Control Building

DB = Diesel Generator Building

SB = Service Building

CDWEB = Condensate Demineralizer

Waste Evaporator Building

O = Outdoors above ground

B = Buried in ground

P = ERCW Pumping Station

TB = Turbine Building

(6) X = Source of radiation

\- = No source of radiation

P = Possible source of radiation

(7) I = Seismically qualified

I(L) = Limited seismic qualification

I(L)A = Limited seismic qualification - pressure boundary integrity and position retention

I(L)B = Limited seismic qualification - position retention

\- = Not seismically qualified

(8) AMCA Class III and performance tested in accordance with AMCA Standard air moving devices.

(9) Performance test required.

(10) Deleted by FSAR Amendment 79

(11) Those components of the heating, ventilating, and air conditioning system (HVAC), which are not covered directly by the TVA piping classifications of Subsection 3.2.2, have been designed and constructed to standards and specifications which are equivalent to ANS Safety Class 2b.

(12) The following HVAC cooling coils perform no safety related cooling function but they are seismically designed or qualified as indicated to provide ERCW pressure boundary integrity to ensure the ERCW can perform its primary safety functions:

- Unit 1 and 2 Lower Compartment Cooler cooling coils Seismic Class I
- Unit 1 CRDM Cooler cooling coils Seismic Class I(L)A
- Unit 1 Incore Instrument Room Chiller coils Seismic Class I(L)A

TABLE 3.2-2 (Sheet 22 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

▪ Unit 2 RCP Motor Air Coolers	Seismic Class I(L)A
▪ Unit 2 Upper Containment Cooler coils	Seismic Class I(L)A
▪ Unit 1 Upper Compartment Cooler cooling coils	Seismic Class I
▪ Unit 2 Upper Compartment Cooler cooling coils	Seismic Class I(L)A

The Unit 2 CRDM Cooler coils, Unit 2 Incore Instrument Room Chiller coils, Unit 1 RCP motor air coolers are not required for plant safety and are designed to Seismic Class I(L)B. The differences in Seismic Classification for the HVAC equipment supplied by ERCW are considered in the analyses that demonstrate the acceptable capability of the ERCW system.

(13) Vessel was built to the requirements of ASME code but does not have code stamp.

(14) Acceptable for use within Regulatory Guide 1.26 Quality Group C system (ASME Section III, Class 3.) For the auxiliary air system, see also Note 1 of Table 3.2-2a.

(15) Although the screen wash pumps, piping and valves are required for plant safety, they were not purchased to TVA Class C standards. The pumps are seismically qualified, have limited QA, and were the best commercially available product for the service. The piping and valves are designed to TVA Class G and Seismic Category I(L) for pressure boundary integrity. For this application, this level of qualification meets the intent of TVA Class C. Criteria requires that any future modifications or repair to the ERCW screen wash pumps, piping or valves are made to the requirements of TVA Class C.

(16) This component is actually a system containing many components. Those parts of the system that contain component cooling water are Safety Class C with design code of ASME III, Class 3. The remainder of the system is Safety Class G with design codes as identified in Table 3.2-5.

(17) The secondary chamber of the steam generators (shell side) are built to the ASME B&PV Code Section III, Division 1, Class 1, and applicable code interpretations and/or rulings. Although the shell side of the steam generator functions only dictate a TVA Class B, they were procured to comply with ASME Section III, Class 1. Therefore, repairs, modifications and/or additions shall be in accordance with the original contract specifications and drawing requirements.

(18) The ERCW pump motor bearing cooling coils, required for plant safety, were not purchased to TVA Class C standards. The vendor-supplied cooling coils have been seismically qualified and are considered safety-related and suitable for the intended service. For their application, the level of qualification meets the intent of TVA Class C at the motor interface.

(19) Although not purchased and stamped in accordance with ASME Section III Code Requirements, this equipment meets the highest available commercial quality standards.

TABLE 3.2-2 (Sheet 23 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

(20) All purge air ductwork (supply and exhaust) inside the annulus and exhaust air ductwork from the Shield Building isolation valves 2-FCV-30-61 and -62 to 2–FCV-30-213 and -216 is Seismic Category I. Supply air ductwork from the ABSCE isolation valves 2-FCV-30-294 and -295 to the Shield Building isolation valves 2-FCV-30-2 and -5 and all purge air ductwork (supply and exhaust) inside primary containment up to the inboard containment isolation valves is Seismic Category I(L).

(21) All piping between the containment isolation valves is Seismic Category I. The piping up to the containment isolation valves on each side is Seismic Category I(L).

(22) This tank was procured to ASME III - 3 requirements.

(23) Not used for Unit 2 operation.

(24) The boron recycle (recovery) system is not required for the operation of Unit 2. See FSAR Section 9.3.7. The portions of this system which are used for the operation of Unit 2 are discussed in FSAR Section 9.3.4.

(25) The Primary Water Storage Tank (PWST) meets the ASME Section III, Class 3, design by analysis requirements with Seismic Class I(L) Forcing Functions for atmosphere tanks.

(26) The Unit 2 PWST was initially procured to ASME Section III-3 requirements and was subsequently downgraded to API-650. However, the Unit 2 PWST bottom plate and associated nozzles are Seismic Category I and were procured to the requirements of ANSI N45.2 (Safety Related). The remainder of the Unit 2 PWST is classified as Seismic Category IL(B).

TABLE 3.2-2a (Sheet 1 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Auxiliary Control Air	Portions of the System necessary for containment isolation. (See Note 5)	2a	B	I
	Balance of system. (See Note 1 and Note 5). System boundary is considered to exist to the upstream side of the filters which tie the non-essential control air systems to the auxiliary control air lines.	2b	C	I
Boron Recycle (See Note 12)	Equipment used to provide a ready supply of concentrated boric acid (boric acid tanks, boric acid transfer pumps, boric acid filters, and associated pipes and valves).	2b	C (See Note 11)	I (See Note 11)
	Processing and Waste Holdup Equipment whose failure could result in a site boundary dose of 0.5 rem or more. (See Note 8) (gas stripper feed pumps, holdup tanks and holdup tank recirculation pumps, and associated piping and valves).	3	D	I
	Other equipment which carries minimal or no radioactive wastes and/or has no safety function to perform (monitor tank and pumps, evaporator condensate demineralizers, batching tank, gas stripper and boric acid evaporator packages, evaporator feed ion exchangers, condensate filters, concentrate filters, evaporator feed ion exchange filters, and associated pipes and valving).	-	G	I(L)

TABLE 3.2-2a (Sheet 2 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Chemical and Volume Control	Equipment that circulates reactor coolant normally or during an accident (charging, letdown, excess letdown, seal water return lines; centrifugal charging pumps; volume control tank; and, miscellaneous associated lines and valves). (See Note 8)	2a	B	I
	Equipment necessary for boric acid addition (boric acid tanks, boric acid blender, lines and valves).	2b	C (See Note 11)	I (See Note 11)
	Equipment associated with radwaste cleanup whose failure could result in a 0.5 rem offsite dose (Mixed bed and cation demineralizers, associated piping and valves).	3	D	I
	Balance of equipment (resin fill and chemical mixing tanks, piping and valves).	-	G	I(L)
Component Cooling	Portions of the system necessary for containment isolation.	2a	B	I
	Major pressure boundary components.	2b	C	I
	Equipment inside the CDWE Building. (See Note 13)	-	H	-
	Sample heat exchangers, drains, and vents.	-	G or L	I(L)
Containment Spray	Major pressure boundary components.	2a	B	I

TABLE 3.2-2a (Sheet 3 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Essential Raw Cooling Water	Portions of the system piping necessary for containment isolation.	2a	B	I
	Portions of the system piping required for plant safety. (See Note 4, 10).	2b	C	I
	Portions of the system piping not required for plant safety, but in Seismic Category I structures.	-	G	I(L)
	Portions of system discharge headers located between the two Hydraulic Gradients and the associated Valve Boxes 1 and 2.	-	H	I(L)
	Portions of the system piping not required for safety and not in Seismic Category I structures.	-	H	-
	Auxiliary control air compressors (see Note 1).	-	C	I
	HVAC equipment required for plant safety (see Note 2).	-	-	I
	HVAC equipment not required for plant safety	-	-	See Note 3
	Portions of the strainer backwash/backflush piping (see Note 6).	-	G	I(L)
Feedwater	Downstream of and including the anchors in the valve room exterior walls.	2a	B	I
	Flow transmitter sensing lines	-	P	-
	Upstream of the anchors	-	H	-
Feedwater, Auxiliary	Downstream of and including the first anchor which is immediately up- stream of the check valve closest to and outside of containment.	2a	B	I
	Portions of the system not in SC-2a but required after a seismic event.	2b	C	I
	Condensate supply and other piping not required after a seismic	-	G	I(L)

TABLE 3.2-2a (Sheet 4 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Fire Protection High Pressure (HPFP)	Piping necessary to provide water to AFW system in the event of a flood above plant grade. Equipment necessary to provide makeup to the primary and spent fuel cooling systems in the event of a flood above plant grade.	2b	C	I
	Balance of equipment within Seismic Category I structures.	-	G	I(L)
	Remainder	-	H	-
Flood Mode Boration and Makeup (Auxiliary Charging)	Portion of the system necessary for containment isolation.	2a	B	I
	Piping essential for makeup and boration in the event of a flood above plant grade.	2b	C	I
	Balance of system.	-	H	I(L)
Fuel Oil	Equipment necessary to assure continuous, full power operation of the emergency diesel-generator sets for seven days following a loss of offsite power.	2b	C	I
	Balance of equipment in Seismic Category I structures which performs no safety function but needs to maintain pressure boundary.	-	G or H	I(L)
	Remainder	-	H	-

TABLE 3.2-2a (Sheet 5 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Fuel Pool Cleaning and Cooling	Portions of the system required for containment isolation.	2a	B	I
	Portions of the system required to cool the spent fuel (heat exchangers, pumps, associated piping and valves).	2b	C	I
	Makeup water loop from the RWST through RWP pumps to isolation valve downstream of pumps SFP skimmer piping.	-	G (See Note 7)	I(L)
	Line from isolation valve downstream of RWP pumps to the SFPC loop	2b	C	I
	Balance of system.	-	G	I(L)
Heating, Ventilation, and Air Conditioning (HVAC)	System containment isolation valves and piping between valves.	2a	B	I
	HVAC components, ductwork, and piping located in the Reactor, Auxiliary, Control, and Diesel Generator Buildings that perform safety related air cooling and heating operations or essential air filtration and purification processes or that supply life supporting air (see Note 2).	2b	M,Q, or S	I
	Balance of system (see Note 3)	-	M,Q,S,U,V	I(L)
Hydrogen Analyzer	Portion of H_2 Analyzer system which supplies pure O_2 as reagent gas to H_2 Analyzer Panels and Vacuum Trap Assemblies located on sample tubing low points for H_2 Analyzer System. (see Note 9). (Unit 1 Only)	-	-	I

TABLE 3.2-2a (Sheet 6 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Ice Condenser	Portions of the system required to function during a DBA.	2b	C	I
	Portions of the system that provide a containment isolation function	2a	B	I
	System refrigeration piping	-	N	I(L)
	System refrigeration piping in Reactor Building.	-	M	I(L)
	Balance of system not classified as refrigeration piping (i.e. drains).	-	G or H	I(L)
Main Steam	Upstream of, and including the flued-head anchors in the valve room exterior walls.	2a	B	I
	Turbine impulse pressure transmitter sensing lines	-	P	-
	Downstream of the flued-head anchors.	-	H	-
Reactor Coolant	Equipment within the reactor coolant system boundary, failure of which could cause a Condition III or IV loss-of-coolant accident. (Components downstream of a 3/8 inch or smaller orifice are excluded).	1	A	I
	Portions of the system protected from reactor coolant pressure by a 3/8 inch or smaller orifice, reactor vent head ventilation system, and branch piping for the pressurizer steam space instrumentation nozzles (0.83 inch inside diameter, or smaller).	2a	B	I
	Portions of the system that provide a containment isolation function.	2a	B	I
	Safety and relief valve discharge piping.	-	G	I(L)
	Equipment that does not provide a safety function (pressurizer relief tank (PRT), primary water supply inside containment, nitrogen supply and vent headers to the PRT).	-	G	I(L)

TABLE 3.2-2a (Sheet 7 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Residual Heat Removal (RHR)	Major pressure boundary components.	2a	B	I
Safety Injection (SI)	Balance of system that recirculates reactor coolant after an accident or prevents leakage of reactor coolant to points external to the system. (See Note 8).	2a	B	I
	Refueling water storage tank and SIS accumulators.	2a	B	I
	Piping from the SIS accumulators to the accumulator isolation valves, and from the RWST to SC-2a valves in the safety injection, RHR, charging pump, and containment spray pump suction lines.	2a	B	I
	Piping to CVCS holdup tanks.	2b	C	I
	Accumulators N_2 fill line.	-	G	I(L)
Steam Generator Blowdown	Piping and valves from the steam generators to and including the containment isolation valves	2a	B	I
	Piping and valves down stream of the containment isolation valves to Column U	-	G	I(L)
	Piping and valves down stream of Column U.	-	H	-
Waste Disposal	Portions of system that provide a containment isolation function.	2a	B	I
	Equipment whose failure could cause a site boundary dose of 0.5 rem or more (per RG 1.26). (See Note 8).	3	D	I
	Balance of system in Seismic Category I structures for which a component failure may cause damage to safety-related equipment.	-	G	I(L)
	Equipment not within Seismic Category I structures.	-	H	-

TABLE 3.2-2a (Sheet 8 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS
RELATED TO A PRIMARY SAFETY FUNCTION

Note 1: Although not purchased and stamped in accordance with ASME Code Section III requirements, the auxiliary control air system compressors and dryers meet the highest commercial quality standards.

Note 2: Although not purchased under ASME Section III requirements, the HVAC equipment is required for plant safety and does meet the highest commercial quality standards. Safety Class 2b round flexible and triangular duct board ducting installed as part of the ceiling air delivery system in the main control room above the suspended ceiling is qualified to limited seismic requirements, analyzed to ensure that the ducting will remain in place, the physical configuration will be maintained such that flow will not be impeded, the ducting pressure boundary will not be lost, and the ducting is constructed of standard commercial grade materials.

Note 3: The following equipment performs no safety related cooling function but it is seismically designed or qualified as indicated to provide ERCW pressure boundary integrity to ensure the ERCW can perform its primary safety functions:

- Unit 1 and 2 Lower Compartment Cooler cooling coils — Seismic Class I
- Unit 1 CRDM Cooler cooling coils — Seismic Class I(L)A
- Unit 1 Incore Instrument Room Chiller coils — Seismic Class I(L)A
- Unit 2 RCP Motor Air Coolers — Seismic Class I(L)A
- Unit 1 Upper Containment Cooler cooling coils — Seismic Class I
- Unit 2 Upper Compartment Cooler cooling coils — Seismic Class I(L)A

The Unit 2 CRDM Cooler coils, Unit 2 Incore Instrument Room Chiller coils and Unit 1 RCP Motor Air Coolers are not required for plant safety and are designed to Seismic Class I(L)B. The differences in Seismic Classification for the HVAC equipment supplied by ERCW are considered in the analyses that demonstrate the acceptable capability of the ERCW system.

Note 4: Although the screen wash pumps, piping and valves are required for plant safety, they were not purchased to TVA Class C standards. The pumps are seismically qualified, have limited QA, and were the best commercially available product for the service. The piping and valves are designed to TVA Class G and Seismic Category I(L) for pressure boundary integrity. For this application, this level of qualification meets the intent of TVA Class C. Criteria requires that any future modifications or repair to the ERCW screen wash pumps, piping or valves are made to the requirements of TVA Class C.

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

NRC Bulletin 83-06 "Nonconforming Materials Supplied By Tube-Line Corporation" has been evaluated for this system; Carbon Steel and Stainless Steel Program Plans were developed, presented to the NRC, verbally approved, initiated, completed, and reported to the NRC in the NRC Bulletin 83-06 report. The NRC Bulletin 83-06 and NCR GENMEB 8301 were closed and approved by the NRC Inspection Reports 50-390/84-03 and 50-391/84-03. The fittings that were installed and found to be acceptable are identified in Tables 3.1-6 and 3.1-7 of WB-DC-40-36 (Reference DIM WB-DC-40-36-18). The potential effect of unacceptable indications in radiographs of Tube-Line fittings welded with filler material has also been evaluated. The radiographs were supplied by Tube-Line as required by the material specification (ASME SA-403). Piping stress analysis was reviewed for Condition Adverse to Quality Report WBP890546. The review showed that stresses in the fittings are within ASME Section III allowable stresses even if the worst radiographic indications for each size fitting were to be transposed to the highest-stressed fitting. Authorization to use an alternative to the testing requirements of Section III Subsection ND-2000 of the ASME Code was provided by the NRC through a SER dated September 23, 1991.

Note 5: Portions of the control air system were not pneumatic tested to the correct pressure. NRC Inspection Report Nos. 50-390/90-04 and 50-391/90-04 has approved an alternate acceptance to pneumatic test criterion.

Note 6: The strainer backwash/backflush piping has been upgraded from Class G to Class G Seismic Category I(L) for pressure boundary integrity.

Note 7: Class G piping labeled with PBQA is analyzed for Seismic Category I(L) pressure boundary retention and is within the scope of the hydrostatic QA program (the valve seat terminates the PBQA boundary). All remaining Class G piping is seismically supported for position retention only.

Note 8 Manual block valves exist in the discharge piping of the relief valves which provide overpressure protection for the volume control tank, the boron injection tank, and the waste gas compressors. ASME Code, Section III, Subsection NC/ND, Paragraph 7153 prohibits the placement of a block valve in the discharge of pressure relief devices unless the block valve is installed with positive controls and interlocks and means are provided such that the operation of the controls and interlocks can be verified. The following design and administrative features, evaluated and approved by NRC[2], are an acceptable alternative to the ASME code requirements because they provide reasonable assurance that both stop valves would not be left closed during plant operation. Redundant flowpaths exist in the relief valve discharge piping, the locked-open block valves are installed in a controlled access location, and administrative procedures are in place to assure the locked-open position of the block valves.

Note 9 Although not purchased under ASME Section III requirements, the O2 supply bottles, related manifolds and vacuum trap assemblies are required for post-LOCA conditions and meet the highest commercial quality standards, and are qualified to WBNP Seismic Category I classification. (Unit 1 only)

TABLE 3.2-2a (Sheet 10 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

Note 10 Some welds in the buried portion of the ERCW System did not receive a Section III hydrostatic test visual examination. They did, however, receive a vacuum box examination after welding and the code required NDE. Additionally, a pressure test was performed and held for 1 hour, then a Section XI VT-2 test was performed using 1 psi/min pressure drop or a 2 gal/min loss for 10 min. This was approved by the NRC in Safety Evaluation Report, Supplement 12.

Note 11 Although not originally purchased to ASME Section III and Seismic Category I requirements, the replacement immersion heater assemblies for boric acid tank A have been non-destructive tested and evaluated to be acceptable for the classification.

Note 12 The boron recycle system is not required for operation of Unit 1 or Unit 2. See UFSAR Section 9.3.7. The portions of this system which are used for the operation of Unit 1 or Unit 2 are discussed in UFSAR Section 9.3.4.

Note 13 Not used for Unit 1 or Unit 2 operation.

TABLE 3.2-2b

CLASSIFICATION OF SYSTEMS NOT HAVING MAJOR DESIGN CONCERNS
RELATED TO A PRIMARY SAFETY FUNCTION
(See Note 2 below)

System Subsection	Safety Class ANS. N-18.2	TVA Class	Seismic Category
Portion of system necessary for primary containment isolation.	2a	B	I
Portion of system in Seismic Category I structures and not in a higher safety class (except refrigeration piping).	---	G, H or L (See Note 1)	I(L)
Balance of system (except refrigeration piping).	---	H, J, or L	---

Note 1:In special applications, where the code requirements for Class G are not appropriate, Class K may be used.

Note 2:All Entries Above Apply to the Following Systems:

Auxiliary Boiler	Hydrogen Cooling
Carbon Dioxide Storage, Fire Protection, and Purge	Layup Water Treatment
Condensate	Lube Oil
Condenser Circulating Water	Makeup Water
Condenser Tube Cleaning	Potable Water
Demineralized Water	Primary Water
Extraction Steam	Raw Cooling Water
Feedwater Treatment and Secondary Chemical Feed	Raw Service Water
Gland Seal Insulating Oil	Station
Gland Seal Water	Drainage
Heater Drains and Vents	Turbine Drains and Misc. Piping, Vacuum primer

TABLE 3.2-3 (Sheet 1 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
6.9-kV Auxiliary Power System		
6.9-kV Shutdown Boards	2/4	Yes (2)
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
6.9-kV Shutdown Logic Relay Panels	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
6.9-kV/480V Shutdown Board Transformers	6/12	Yes
(2000 kVA) (Unit 1) 1A1-A, 1A-A, 1A2-A, 1B1-B, 1B-B, 1B2-B (Unit 2) 2A1-A, 2A-A, 2A2-A, 2B1-B, 2B-B, 2B2-B		
6.9-kV/480V Pressurizer Heater Backup Group Transformers (500 kVA)	2/4	Yes (3)
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
480V Auxiliary Power System		
480V Shutdown Boards	4/8	Yes
(Unit 1) 1A1-A, 1A2-A, 1B1-B, 1B2-B (Unit 2) 2A1-A, 2A2-A, 2B1-B, 2B2-B		

TABLE 3.2-3 (Sheet 2 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
480V Reactor MOV Boards	4/8	Yes
(Unit 1) 1A1-A, 1A2-A,		
1B1-B, 1B2-B		
(Unit 2) 2A1-A, 2A2-A,		
2B1-B, 2B2-B		
480V Reactor Vent Boards	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
480V Control and Auxiliary		
Bldg. Vent Boards	4/8	Yes
(Unit 1) 1A1-A, 1B1-B,		
1A2-A, 1B2-B		
(Unit 2) 2A1-A, 2B1-B,		
2A2-A, 2B2-B		
480V Diesel Auxiliary Boards	4/8	Yes
(Unit 1) 1A1-A, 1B1-B, 1A2-A, 1B2-B (Unit 2) 2A1-A, 2B1-B, 2A2-A, 2B2-B		
480V Distribution Panelboards for Pressurizer Heater Backup Groups	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
480V Transfer Switch for Component Cooling System Pump C-S	-/1	Yes

TABLE 3.2-3 (Sheet 3 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
120V AC Vital Plant Control Power System		
Static Inverter System Components	4/12	Yes
a. Auctioneer unit b. A transformer rectifier power supply c. A single phase static inverter with associated equipment for control, voltage, regulation, filtering, and instrumentation d. Regulated transformer bypass source with static and manual bypass switches (Not applicable to spare inverters)		
(Unit 1) 1-I, 1-II, 1-III, 1-IV		
(Unit 2) 2-I, 2-II, 2-III, 2-IV		
(Spare) 0-I, 0-II, 0-III, 0-IV		
120V AC Vital Instrument Power Boards	4/8	Yes
(Unit 1) 1-I, 1-II, 1-III, 1-IV		
(Unit 2) 2-I, 2-II, 2-III, 2-IV		
125V DC Vital Plant Control Power System		
480V AC Vital Transfer Switches	-/4 Note (4)	Yes
Transfer SW I, II, III, IV		
125V DC Vital Battery Chargers	-/9 Note (4) (5)	Yes
Chgrs I, II, III, IV, V		
Chgr 6-S, 7-S, 8-S, and 9-S		
Transfer Devices for Spare 125V DC		
Vital Battery Chargers	-/8 Note (4)	Yes
DC Transfer Switch 68DC1-S & 68DC2-S DC Transfer Switch 79DC1-S &		

TABLE 3.2-3 (Sheet 4 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
79DC2-S AC Transfer Switch 68AC1-S & 68AC2-S AC Transfer Switch 79AC1-S & 79AC2-S		
480V AC Vital Disconnect Panels		
	-/4 Note (4)	Yes
125V DC Vital Batteries	-/4 Note (4)	
Batteries I & II		
Batteries III & IV		
125V DC Vital Battery Boards	-/4 Note (4)	Yes
I, II,		
III, IV		
Electrical Penetrations		
High Voltage Power Penetrations	4/8	Yes
Nuclear Instrument System Penetrations	4/8	Yes
Control Rod Position Indication Penetrations	1/2	Yes
Low Voltage, Power, Control, and Indication Penetrations	41/82	Yes
Thermocouple Penetrations	3/6	Yes
Onsite Electrical Power Source Components		
Diesel Generator Protective Relay Panels	2/4	Yes
(Unit 1) 1A, 1B		
(Unit 2) 2A, 2B		
Diesel Control Panels	2/4	Yes

TABLE 3.2-3 (Sheet 5 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
125V Diesel Generator Batteries and Battery Racks	2/4	Yes
DC Distribution Panels	2/4	Yes
125V DC Dual Battery Charger Assemblies	2/4	Yes
Standby Diesel Generators	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		

1. Those equipment items procured prior to publication of IEEE 344-1971 were purchased under specifications which TVA believes conform to the intent of that document. Equipment procurement, modification, and evaluation activities after September 1, 1974 applied the IEEE 344-1975 standard for seismic qualification.

2. The 6.9-kV shutdown boards are qualified under Section 3.2.2.4.3 of IEEE 344-1971. The test unit withstood higher accelerations than shown on the frequency response spectrum for resonance at 1 percent damping.

3. The 500-kVA transformers were shown analytically to have lower stress under seismic loading conditions than the 2000-kVA transformers which were tested. The 500-kVA transformers are similar in design and construction to the 2000-kVA transformers.

4. The 125-V DC Vital Control Power System is not unitized therefore, numbers shown are on a per plant basis.

5. The discussion of the number of chargers in the vital battery system in certain UFSAR sections (i.e., 1.2.2.7, “Plant Electrical System,” 8.3.2.2, “Analysis of Vital 125V DC Control Power Supply System”) indicate there are eight chargers. The eight chargers are the normal supplies for the four vital channels (I, II, III and IV). The ninth charger is the charger used to maintain Vital Battery V. Additional information regarding this charger is provided in UFSAR Section 8.3.2.2.

TABLE 3.2-4

SUMMARY OF CODES AND STANDARDS
FOR SAFETY CLASS COMPONENTS OF THE WATTS BAR NUCLEAR PLANT

Code Requirements

Safety Class ANS N-18.2	TVA Class	Seismic Category	Code Classification Piping, Pumps, Valves, and Vessels	Remarks
1	A	I	ASME Code, Sec. III, Class 1	Note 1
2a	B	I	ASME Code, Sec. III, Class 2	Note 1
2b	C	I	ASME Code, Sec. III, Class 3	Notes 1,2,3
3	D	I	ASME Code, III, Class 3	Notes 1 & 3

NOTE: 1) Equipment designated "Vendor-Supplied Safety-Related Equipment Packages" on the drawing meet the following requirements:

a. The vendor-supplied equipment packages (component and piping) contained within TVA piping systems classified as A, B, C, or D which do not meet the requirements of ASME Section III are installed and documented using the rules of ASME Section III, and manufacturer's instruction manuals, as requirements, except that the materials and equipment are not certified to Section III, and N-5 data report is not required. 10 CFR 50 Appendix B applies. In some cases there may be portions of these packages which are not safety-related (i.e., drains and vents past the first normally closed isolation valve) and do not require installation to these requirements.

b. Any substitute material used or repairs performed by construction shall be in accordance with the original contract specification and drawing requirements.

c. TVA Class D components are those whose postulated failure would result in potential offsite doses that exceed 0.5 Rem to the whole body, or its equivalent. Class D components do not perform a primary safety function.

d. Exception is taken to Note 1.a above for the auxiliary systems supplied on the diesel generator skid. The fuel oil, engine cooling water (except the ASME Section III, Class 3 heat exchangers), starting air and lubricating oil systems are designed per ANSI B31.1. They are designed to Seismic Category I and are within the 10CFR50 Appendix B QA program. Criteria requires that any modifications to this piping are performed to meet the intent of ASME Section III Class 3 (TVA Class C).

2) ANSI B31.1 code is an acceptable substitute for the ASME code for installation of piping and valves on Class C instrument lines attached to TVA Class M, Q, and S systems. 10 CFR 50 Appendix B applies.

3) Condition Adverse to Quality Report WBP 900336SCA and NRC Violation 390/90-15-02 identified lack of penetration and/or lack of fusion in ASME Code Section III, Class 3 butt welds made prior to September 26, 1990.

TABLE 3.2-5

NON-NUCLEAR SAFETY CLASSIFICATIONS

TVA Class	Seismic Category	Piping	Code Classification Pumps	Valves	Vessels
G	I(L)	ANSI B31.1	Manufac-turers Standards	ANSI B31.1 B16.5, or MSS-SP-66	ASME Code, Sec. VIII, Div. 1
H	Note 1	ANSI B31.1	*	ANSI B31.1, B16.5, or MSS-SP-66	*
J	N/A	*	*	*	*
K	I(L)	*	*	*	*
L	Note 1	*	*	*	*
P	Note 2	*	*	*	*

* Code used is determined by the design requirements of the equipment.

Note 1: Those portions of TVA Class H and L systems located inside Seismic Category I structures and those portions of TVA Class H yard piping located between the two ERCW Hydraulic Gradients and the associated Valve Boxes 1 and 2 are Seismic Category I(L). The balance of these systems are not designed for seismic loading.

Note 2: This class applies to specific sensing lines which meet ASME Code Section III, Class 3 requirements except that inertia effects need not be used for design of lines in non seismic areas and independent verification by an Authorized Nuclear Inspector is not required for fabrication and installation. 10 CFR 50 Appendix B applies. N-5 data report is not required. Portions of sensing lines in Seismic Category I structures meet Seismic Category I(L) pressure boundary requirements.

Note 3 :The Unit 2 Primary Water Storage Tank (PWST) shall be classified as API-650 in lieu of ASME Code Section VIII, Div. 1. The Unit 2 tank bottom plate and nozzles shall be qualified to Seismic Category I requirements and shall meet the requirements of 10 CFR 50, Appendix B; i.e., the Unit 2 PWST bottom plate and associated nozzles were procured to the requirements of ASTM N45.2 (Safety Related). The connecting tank piping located within the ABSCE shall be qualified to Seismic Category 1L(A) (pressure boundary & position retention) requirements. The remainder of the tank and piping shall be qualified to Seismic Category 1L(B) (position retention) requirements.

TABLE 3.2-6

TVA HEATING, VENTILATION AND AIR CONDITIONING CLASSIFICATIONS

TVA Class	ANS Safety Class	Code Jurisdiction	Seismic Category
M	2b*	ANSI B31.5	I or I(L)
N	None	ANSI B31.5	Note 1
Q	2b*	Round Duct, Steel, Spiral or Longitudinal Seam, Locked Seam or Welded (ASTM A211) and SMACNA High Velocity Duct Construction Standards, 2nd Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3). ANSI/ASME N509 (see Note 2).	I or I(L)
S	2b*	Rect. Duct, Steel, Locked Seam or Welded, SMACNA High Velocity Duct Construction Standards, 2nd Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3). ANSI/ASME N509 (see Note 2).	I or (L) Note 4
U	None	Round Duct, Steel, SMACNA Low Velocity Duct Construction Standards, 4th Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3).	I(L)
V	None	Rect. Duct, Steel, SMACNA Low Velocity Duct Construction Standards, 4th Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3).	I(L)

*TVA Class M, Q, and S designations are also used on heating, ventilation, and air-conditioning systems which have no ANS safety class requirements if seismic requirements are invoked.

Note 1: Those portions of TVA Class N systems located inside Seismic Category I structures are Seismic Category I(L). The balance of these systems are not designed for seismic loading.

Note 2: Those portions of TVA Classes Q and S Category I duct which are of welded construction, that are fabricated or repaired after January 12, 1987, meet the welding requirements of ANSI/ASME N509 1976. The workmanship samples are not required to have Penetrant Testing (PT) or Magnetic Testing (MT).

Note 3: *Historical Information* - All duct installations prior to December 21, 1990, (except those discussed in Note 4 below) were evaluated and qualified to meet the requirements of WB-DC-40-36.1.

All duct installations after December 21, 1990, shall be in accordance with specifications G-95, N3M-914, N3C-942, and Design Criteria WB-DC-40-31.8.

Note 4 Safety Class 2b round flexible and triangular duct board ducting installed as part of this ceiling air delivery system in the main control room above the suspended ceiling is qualified to Seismic I(L) requirements, analyzed to ensure that the ducting will remain in place, the physical configuration will be maintained such that flow will not be impeded, the ducting pressure boundary will not be lost, and is constructed of standard commercial grade materials.

TABLE 3.2-7 (SHEET 1 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
I. CODE CASES				
A. DESIGN/MATERIAL RELATED				
N/A	N-192	Provides rules for use of flexible metal hose	Regulatory Guide 1.84, Rev 26 imposes the following addition to the requirements of Code Case N-192: The applicant should indicate system application, design and operating pressure/temperature rating of the flexible hose. Data to demonstrate compliance of the flexible hose with NC/ND-3649 particularly NC/ND-3649.4(e), are required to be furnished with the application.	Referenced in data report for some of the flexible metal hose assemblies.
N-224-1	N/A	Provides rules for the use of ASTM A-500 Grade B and ASTM A-501 as integrally welded attachments.		
N/A	N-304	Provides for use of other materials not listed in the appendices.	None	Code Case N-304 was originally approved by the NRC in Regulatory Guide 1.84, Rev. 20 dated November 1982. Subsequent revisions of the Code Case have continued to retain unrestricted approval.
NX-2000	N-188-1	Provides rules for using alloy 625 or 825 tubing that is welded without filler metal.	None	Materials to be used for flexible metal hose assemblies
N/A	N-514	Provides alternate analysis for pressure/temperature curves and low temperature overpressure protection (LTOP) system.	None	Pressure and Temperature Limits Report (PTLR)
B. FABRICATION/EXAMINATION/TESTING RELATED				
N-32-4	NA	Provides for alternative testing of inaccessible or embedded	Regulatory Guide 1.84, Rev. 26, accepts Code Case N-32-4 based on the following clarification	Used in G-29 Process Specification 3.M.9.1

TABLE 3.2-7 (SHEET 2 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		welded joints in piping.	and interpretation. Code Case N-32-4 does not replace paragraph NC/ND-6129, "Provisions For Embedded Or Inaccessible Welded Joints In Piping," of the Code. The intent of the Code Case is to provide additional testing above code requirements and permit liquid penetrant or magnetic particle testing in place of radiographic testing for Class 3 piping with 3/8 inch nominal wall or less. Paragraph 1 contains an additional requirement to the Code. It was, therefore, acceptable but unnecessary to include in the Code Case. Paragraph 2 is a variation in the volumetric examination technique and was acceptable as written. Paragraph 3 contains an additional requirement and is not a relaxation of the Code. It was, therefore, acceptable but unnecessary to include in the Code Case.	
N-127	NA	Provides for an alternative examination requirement for Class 1 and 2 welds made by an automatic welding process.	None	Used on the 47B333 drawings.
N-237-2	NA	Permits acceptance of open ended Class 2 or 3 piping within Class 2 or 3 vessels or tanks or into the gaseous atmosphere of Class MC vessels without hydrostatic testing	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-240	NA	Permits acceptance of open ended piping without hydrostatic testing.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-241	NA	Safety and safety relief valve piping submerged in a suppression pool inside a Class MC or CC vessel may be exempted from hydrostatic	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.

TABLE 3.2-7 (SHEET 3 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		testing.		
N-292	NA	Provides rules for addition of weld metal at weld end prep to meet dimensional and minimum wall considerations.	Regulatory Guide 1.84, Rev. 26 imposes the following requirement in addition to the requirements in Code Case N-292: Class 3 piping that is longitudinally welded and that has a weld efficiency factor of 1.0 as selected from Table ND-3613.4-1 should receive a 100% volumetric examination (RT or UT) of the deposited weld metal in accordance with the requirements of ND-5000.	Used to disposition NCR W-4-P.
N-316	NA	Provides for socket weld sizes <1.09 TNOM.	None	Used to disposition NCRs 3702R R0, W-427-P R0, 3555R R0, 4135R R0, 4345R R1, 4739-0, 2217R, 4114, 5305-0, and 5833 R0.
N/A	N-341	Provides for a five year period of certification of Level III NDE personnel.	None	
C.	QUALITY ASSURANCE RELATED			
N-272	NA	Allows information to be cross referenced or depicted on an attached drawing rather than physically attached to the data report.		
N-282	NA	Provides alternatives when Code data plates are removed.	None	Used in disposition of NCRs 1464R R1 and 3750.
II.	CODE EDITION AND ADDENDA			
A.	DESIGN/MATERIAL RELATED			
Sect II SA-36	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-515	74W74	Bend testing made an optional	None	See Interpretation III-1-83-275.

TABLE 3.2-7 (SHEET 4 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		(supplemental) requirement		Used in G-62.
Sect II SA-516	74W74	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-283	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-479	74S76	Decreased the required elongation to 30% and reduction in area to 40%.	None	Used in G-62.
NCA-1140(e)	80W81	Permits the use of material certified to editions and addenda earlier than the Code of Record providing certain requirements are met.	None	Used in G-62
NE-4430	77W78	Deletes the minimum size of attachment pads	None	Used in disposition of NCR 3723R
ND-4435	74ED	Provides for certain minor permanent attachments to be made of uncertified material	None	Used to disposition NCR 5409 R1
NA	80S81	Deleted the paragraph limiting instrument take off size	None	Used to disposition NCR 2001R. Also involved in change number NC/ND3676-1
NE-3213.10	74ED	Provides for acceptance of locally over-stressed conditions	None	Used to disposition NCR 3250 R1
NC/ND-3612.4A	74W76	Provides for use of intervening isolation valves	None	
NA	80S81	Deletes requirements formerly in paragraph NC/ND-3676 affecting lagging of steam instrument lines	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not been used at WBN.
NC/ND-3643.1	77S77	Adds provisions for making branch connections by means of complete penetration pipe to pipe welds.	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not

TABLE 3.2-7 (SHEET 5 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
				been used at WBN.
NC/ND-2311	74W76	Lists materials exempted from impact testing. This provision exists in Section NB 2311 Summer 1972 Addenda which is referenced in Section NC/ND.	None	Requirements of later Code exempting certain materials from impact test for Class 2 and Class 3 systems.
ASME III, Appendix XVII, 2461.1	80ED	Bolting material allowable stress of non-pressure retention parts (i.e. motor actuator mounting screws).	None	For the analysis of non-pressure retention parts of 1-FCV-72-13 and -34 for Sect. increased valve closure thrust loops. SQN calc. SCG-4M-00786 was reviewed for WBN application. Since the 1971 ASME III NC Codes do not contain stress allowables for non-pressure retention parts, thus ASME III NC and Appendix XVII of 1980 Codes, Sect. 2461.1 rules will be used for the analysis. Ref. DCN-S-33722-A and MEB Calc. EPM-CDM-071092 and -071192.
B.	FABRICATION/EXAMINATION/TESTING RELATED			
NX-4436	80W81	Permits limited welding of attachments to a piping system after it has been hydrostatically tested providing certain conditions are met.	None	Used in G-29 Process Specification 3.M.9.1.
NC-6129	77W78	Provides for alternative testing of inaccessible or embedded welding joints in piping.	None	Used in G-29 Process Specification 3.M.9.1.
NB-6128	77S78	Piping systems which serve as spray systems shall be hydrostatically tested to the rules of this section, except that the	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.

TABLE 3.2-7 (SHEET 6 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		test may be performed with the spray nozzle attachment connections plugged. The spray nozzles and any connection beyond the run connections need not be hydrostatically tested.		
NX-6211	80S80	Air pockets in components or systems shall be minimized during the conduct of the hydrostatic test by providing vents at high points, or by flushing the system or by providing calculations to show that the entrapped air is dissolved at the pressure/temperature conditions existing during the test.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
NB-4622.1-1 (TBL)	74ED	PWHT holding time is specified to be: For T less than or equal to 1/2 inch, 30 minutes; For T greater than 1/2 inch but less than or equal to 2 inches (P number 1 & 3), 1 hour per inch; For T greater than 2 inches (P number 1 & 3), 2 hours plus 15 minutes per inch over 2 inches; For T greater than 1/2 inch but less than or equal to 5 inches (P number 4,5,6,7,11,11A), 1 hour per inch; For T greater than 5 inches (P number 4,5,6,7), 5 hours plus 15 minutes per inch over 5 inches. For T greater than 5-inches (P Number 11, 11A) one hour per inch.	None	Used in G-29 Process Specification 2.M.1.1.
NB-4623	74ED	For PWHT above 800°F, the rate	None	Used in G-29 Process

TABLE 3.2-7 (SHEET 7 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.		Specification 2.M.1.1.
NB-4624.3	74ED	For PWHT, the minimum width of controlled band at each side of the weld on the face of the greatest weld width shall be the thickness of the weld or 2 inches whichever is less. (2.M.1.1 uses the minimum from 71S73 and the maximum from 74).	None	Used in G-29 Process specification 2.M.1.1. G-29 uses 2 times the thickness or 2 inches whichever is less. This is more conservative than the Code requires.
NC-4623	74ED	For PWHT above 800°F, the rate of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.	None	Used in G-29 Process Specification 2.M.1.1.
NA	74ED	Deleted requirement of Code of Record Paragraph NB-4623.1 that furnace temperature be less than 600°F when the component is placed in it for PWHT.	None	Used in G-29 Process Specification 2.M.1.1.
NX-4453.1	83S83	Relative to examination of defect removal area (adds the following) this examination is not required where defect removal removes the full thickness of the weld and where the backside of the weld is not accessible for removal of examination materials.	None	Used in G-29 Process Specification 1.M.1.2.

TABLE 3.2-7 (SHEET 8 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4231.2	74ED	Deletes Code of Record requirements for MT/PT examination of temporary attachment removal sites on Class 3 components. Permits temporary attachments but deletes the NDE examinations of removal areas.	None	Used in G-29 Process Specification 1.M.1.2.
NC/ND-6322	80S81	(By reference to NC/ND-6222) If the minimum test pressure of NC/ND-6221(a) or (d) is exceeded by 6% at any location, the upper limit shall be established by analysis using all loadings that may exist during the test.	NC/ND-6221(a) NC/ND-6221(d)	Used in G-29 Process Specification 3.M.12.1
NX-5110	74ED	(By reference to Section V T-732.2(B) which refers to T-733.2) Direct or rectified current shall be used at 700/N to 900/N amperes per inch OD up to 5 inches and 500/N to 700/N amperes per inch OD from 5 to 10 inches and 300/N to 500/N amperes per inch OD over 10 inches where N = # of turns.	T-733.2	Used in G-29 Process Specification 3.M.2.1.
NB-5110	74ED	(By reference to Section V T-733.2.) Direct or rectified current shall be used at 700 to 900 amperes per inch OD up to 5 inches and 500 to 700 amperes per inch OD from 5 to 10 inches and 300 to 500 amperes per inch OD over 10 inches.	None	Used in G-29 Process Specification 3.M.2.1.
NX-4427-1 (FIG)	80S80 80W81	Minimum size for socket weld fitting fillet weld (CX) = 1.09 T	None	Used in G-29 Process Specification 3.M.5.1.

TABLE 3.2-7 (SHEET 9 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
	Errata	nominal.		
NX-5110	74S75	(By Reference to Section V T-732.2) For encircling coils, direct or rectified current at 35,000 ampere-turns divided by the sum of 2 plus the L/D ratio of the test part shall be used for magnetization.	T-732.2(A) T-732.2(B)	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74ED	(By reference to Section V T-731.3) Direct or rectified magnetizing current shall be used at a minimum of 100 and a maximum of 125 amperes per inch of prod spacing for sections greater than or equal to 3/4 inch. For sections less than 3/4 inch, amperage shall be 90-110 ampere per inch of prod spacing.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5112	74ED	All NDE performed under this section shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74W74	(By reference to Section V T-630(B)). The penetrant materials are acceptable if the residue does not exceed 0.005 grams or the total sulfur or halogen content shall not exceed 1% of the residue by weight.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5110	74ED	(By reference to Section V T-662). A groove may be machined across the center of each face approximately 1/16	None	Used in G-29 Process Specification 3.M.1.1.

TABLE 3.2-7 (SHEET 10 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		inch deep and 3/64 inch wide, or some other means should be provided to permit side by side comparison without interfering cross contamination between sides.		
NX-5110	77W79	(By reference to Section V T-660) Permits qualification to be performed by the entire block at each temperature, photographing the results and comparing the photographs.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.7.1.
Section IX, QW-203	74ED	Allows welding procedure qualification in any position to qualify for all position production welds except as noted.	QW-405.2 which requires qualification in the vertical position with upward progression if fracture toughness (impact testing) is a requirement.	Used in G-29 Process Specification 1.M.1.2.
NC/ND-2578	74ED	Unacceptable surface defects may be removed by grinding or machining provided: The remaining thickness is not reduced below the minimum required; the depression after defect elimination, is blended uniformly.	None	Used in G-29 Process Specification 4.M.5.1.

TABLE 3.2-7 (SHEET 11 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4130	74W76	Radiography is not required for welded repairs in material used in components provided that the welds joining these materials are not required to be radiographed, the extent of the welded repair does not exceed 10 square inches of surface area and the magnetic particle or liquid penetrant examination of the repair is made as required by ND-2539.4.	None	Used in G-29 Process Specification 4.M.5.1.
NB-4131 NC-4130	74W74	The time of examination of the weld repairs to weld edge preparations shall be in accordance with NB-5130. (NC-5130).	Since NB/NC-5130d did not exist in summer 1973, these two paragraphs from winter 1974 must be added.	Used in G-29 Process Specification 4.M.5.1.
NX-5110	74ED	(By reference to Section V T-535.1) The primary reference shall be equalized at 50% full screen height.	None	Used in G-29 Process Specification 3.M.7.1.
NB-4620 NB-4651	74ED	Ferritic alloy steel pipe that has been heated for bending shall receive a heat treatment in accordance with NB-4620. Exemptions are listed in Table NB-4622.3-1	None	Used in G-29 Process Specification 4.M.2.1.
Sect V T-263.3	83W83	Provides for not using shims if the radiographic density requirements can be met.	None	Used in G-29 Process Specification 3.M.3.1.
NX-5112	74ED	All nondestructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.3.1.

TABLE 3.2-7 (SHEET 12 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NC/ND-3641.1(a)	71W73	Revise the definition of certain terms in Equation 3 and 4 to permit other means than pipe wall thickness to provide for mechanical strength.		Used in the following Calculations and NCRs: B26850423089, B26850508012, B26850710009, B26850710010, B26850808014, B26860611003, MEB850228009, WBP850211062, NCR4145RRO, NCR2217R. Also involved in Code changes NC-3641(A)-1-3, NC-3641(A)-1-4, NC-3641(A)-1-5, ND-3641(A)-1-3, ND-3641(A)-1-4, and ND-3641(A)-1-6.
NX-4311.3	80W81	Provides for use of capacitor discharge welding of thermocouples and strain gauges.	None	Used in G-29 Process Specification 1.M.4.3.
NC-4244(e)	74ED	Provides alternative for attachment of internally threaded bosses.	None	Used in Calculation B26850613072.
ND-5321	77W77	(By reference to Appendix VI, VI-1132) Provides minimum size for relevant RT indications.	None	Used to disposition CAQR WBP890600.
NX-5342	77ED	Defines nonrelevant condition as any indication with a major dimension of less than 1/16 of an inch. (MT examination).	None	
NB-4622.7	74ED	Provides for exemptions to otherwise mandatory post weld heat treatment based on P number, size, carbon content, and preheat.	None	Also invoked by NC/ND-4000.
TBL NX-4622.3-1	74S76	Provides for substitution of 200°F preheat for PWHT when carbon >0.30% and/or tensile strength >70 ksi for certain material applications.	None	Used in disposition of NCR 1146R and in the 47B333 drawings.
NX-4622.3-1 (TBL)	74ED	Provides additional exemptions to PWHT under conditions as described in Table.	New Paragraph NX-4622.3 - defines term “nominal thickness” as used in Table NX-4622.3-1.	Used on the 47B333 Drawings.

TABLE 3.2-7 (SHEET 13 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-5320 & NC-5320	77W77	(By reference to Appendix IV) Provides relevant size and spacing clarification for rounded indications for RT.	None	Provides acceptance criteria for rounded indications for RT
NB-5250 and NC-5261	77W78	Exempts welds of non-structural attachments from liquid penetrant or magnetic particle examination. Does not exempt the removal area of non-structural or temporary attachments.	None	WBPER910208
NB/NC/ND 6114.2	80S81	Provides exemption to requirement to re-hydro following repair by welding if repair weld is not required to be radiographed per NB/NC/ND-4453.4.	None	Used in G-29 process specification 3.M.9.1
NB-2510(a)	83S83	Seamless pipe, tube, and fittings 1"NPS and less need not be examined by the rules of this subarticle.	None	Use of examination requirements for ASME Section III, Class 1, 1" nominal pipe and smaller from a later edition. Invoke for 1" NPS and less, only.
Table NC-4622.3-1	74S76	Exempts PWHT in Pl materials less than 1 1/2" thick provided 200 degrees F preheat is used.	None	Used to disposition WBP 900419PER
NX-4427	80W82	Allows fillet welds to be undersize 1/16 inch for 10% of length of weld.	None	Weld Project
NCA 1273	80S80	Exempts orifice plates not exceeding 1/2" nominal thickness which are clamped between flanges and used for flow measuring service only from the code.	None	Orifice Plates
NB-2538(a)4	77ED	Areas ground to remove oxide scale or other mechanically caused depressions...need not be PT or MT examined.	None	Provisions for not NDE examining impressions caused by mechanical means.

TABLE 3.2-7 (SHEET 14 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4622.9	89 ED	Increases the area which can be repaired by temper bead to 100 in2 and up to 1/3 of base material thickness.	None	Steam generator manway repair or other locations as required.
C.	QUALITY ASSURANCE RELATED			
NCA-4000	80W81	Provided more detailed QA program requirements.	None	Used in the NCM
NCA-8240	83S83	Provides alternatives when code data plates are removed.	None	Used in disposition of WBP880052, NCR 5577R0, NCR 5611R0, and NCR 5619R0.
NCA-8240(b)	80S81	Provides alternatives when code data plates are removed.	None	Use in disposition of NCR 3951R.
NX-2610b and c	77ED	Makes provision to exempt manufacturers of small products from certain QA program` requirements.	None	Used in Disposition NCR GENMEB8402, NCR 5146-0, WBP880431, WBP889432, WBP880433, WBP880437, and WBP880438.
NCA 3800	89ED	Metallic material manufacturers and material suppliers Quality System program provides for non certificate holders QA Manual revision and date to be placed on the documentation and also provides documentation requirements for 1" bar stock.	None	QA programmatic requirements for vendors supplying materials and components.

TABLE 3.2-7 (SHEET 1 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
I. CODE CASES				
A. DESIGN/MATERIAL RELATED				
N-192	NA	Provides rules for use of flexible metal hose	Regulatory Guide 1.84, Rev 26 imposes the following addition to the requirements of Code Case N-192: The applicant should indicate system application, design and operating pressure/temperature rating of the flexible hose. Data to demonstrate compliance of the flexible hose with NC/ND-3649 particularly NC/ND-3649.4(e), are required to be furnished with the application.	Referenced in data report for some of the flexible metal hose assemblies.
N-224-1	NA	Provides rules for the use of ASTM A-500 Grade B and ASTM A-501 as integrally welded attachments.		
N-304	NA	Provides for use of other materials not listed in the appendices.	None	Code Case N-304 was originally approved by the NRC in Regulatory Guide 1.84, Rev. 20 dated November 1982. Subsequent revisions of the Code Case have continued to retain unrestricted approval.
N-188-1	NA	Provides rules for using alloy 625 or 825 tubing that is welded without filler metal.	None	Materials to be used for flexible metal hose assemblies
N-514	NA	Provides alternate analysis for pressure/temperature curves and low temperature overpressure protection (LTOP) system.	None	Pressure and Temperature Limits Report (PTLR)

TABLE 3.2-7 (SHEET 2 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
B.	FABRICATION/EXAMINATION/TESTING RELATED			
N-32-4	NA	Provides for alternative testing of inaccessible or embedded welded joints in piping.	Regulatory Guide 1.84, Rev. 26, accepts Code Case N-32-4 based on the following clarification and interpretation. Code Case N-32-4 does not replace paragraph NC/ND-6129, "Provisions For Embedded Or Inaccessible Welded Joints In Piping," of the Code. The intent of the Code Case is to provide additional testing above code requirements and permit liquid penetrant or magnetic particle testing in place of radiographic testing for Class 3 piping with 3/8 inch nominal wall or less. Paragraph 1 contains an additional requirement to the Code. It was, therefore, acceptable but unnecessary to include in the Code Case. Paragraph 2 is a variation in the volumetric examination technique and was acceptable as written. Paragraph 3 contains an additional requirement and is not a relaxation of the Code. It was, therefore, acceptable but unnecessary to include in the Code Case.	Used in G-29 Process Specification 3.M.9.1
N-127	NA	Provides for an alternative examination requirement for Class 1 and 2 welds made by an automatic welding process.	None	Used on the 47B333 drawings.
N-237-2	NA	Permits acceptance of open ended Class 2 or 3 piping within Class 2 or 3 vessels or tanks or into the gaseous atmosphere of Class MC vessels without hydrostatic testing	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.

TABLE 3.2-7 (SHEET 3 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
N-240	NA	Permits acceptance of open ended piping without hydrostatic testing.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-241	NA	Safety and safety relief valve piping submerged in a suppression pool inside a Class MC or CC vessel may be exempted from hydrostatic testing.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-292	NA	Provides rules for addition of weld metal at weld end prep to meet dimensional and minimum wall considerations.	Regulatory Guide 1.84, Rev. 26 imposes the following requirement in addition to the requirements in Code Case N-292: Class 3 piping that is longitudinally welded and that has a weld efficiency factor of 1.0 as selected from Table ND-3613.4-1 should receive a 100% volumetric examination (RT or UT) of the deposited weld metal in accordance with the requirements of ND-5000.	Used to disposition NCR W-4-P.
N-316	NA	Provides for socket weld sizes <1.09 T_{NOM}.	None	Used to disposition NCRs 3702R R0, W-427-P R0, 3555R R0, 4135R R0, 4345R R1, 4739-0, 2217R, 4114, 5305-0, and 5833 R0.
N-341	NA	Provides for a five year period of certification of Level III NDE personnel.	None	
C.	QUALITY ASSURANCE RELATED			
N-272	NA	Allows information to be cross referenced or depicted on an attached drawing rather than physically attached to the data report.		
N-282	NA	Provides alternatives when Code data plates are removed.	None	Used in disposition of NCRs 1464R R1 and 3750.

TABLE 3.2-7 (SHEET 4 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
N-520-3	NA	Under the provisions Watts Bar Unit 2 will complete documentation and Code Data report and Stamping for those portions of the plant which are suitable for use		Provide documentation to the N-Certificate Holder having overall responsibility to support WBN2 Completion.
II.	CODE EDITION AND ADDENDA			
A.	DESIGN/MATERIAL RELATED			
Sect II SA-36	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-515	74W74	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-516	74W74	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-283	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-479	74S76	Decreased the required elongation to 30% and reduction in area to 40%.	None	Used in G-62.
NCA-1140(a)	80W81	The use of the provisions of NCA-1140(a) from 1980 Edition with Addenda through Winter 1981 will allow the development of design specifications for various components without requiring these components to be in accordance with latest ASME Code Edition. This will allow for consistency in specifying the Code of Record for these components to be as required by the existing design specifications..	None	Preparation of Certified Design Specifications for Watts Bar Unit 2.

TABLE 3.2-7 (SHEET 5 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NCA-1140(e)	80W81	Permits the use of material certified to editions and addenda earlier than the Code of Record providing certain requirements are met.		Used in G-62
NE-4430	80S81	Deletes the minimum size of attachment pads	None	Used in disposition of NCR 3723R
ND-4435	74ED	Provides for certain minor permanent attachments to be made of uncertified material	None	Used to disposition NCR 5409 R1
NA	80S81	Deleted the paragraph limiting instrument take off size from NC/ND-3676.3	None	Used to disposition NCR 2001R.
NE-3213.10	74ED	Provides for acceptance of locally over-stressed conditions	None	Used to disposition NCR 3250 R1
NC/ND-3612.4A	74W76	Provides for use of intervening isolation valves	None	
NA	80S80	Deletes requirements formerly in paragraph NC/ND-3676 affecting lagging of steam instrument lines	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not been used at WBN.
NC/ND-3643.1	77S78	Adds provisions for making branch connections by means of complete penetration pipe to pipe welds.	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not been used at WBN.

TABLE 3.2-7 (SHEET 6 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NC/ND-2311	74S75	Lists materials exempted from impact testing. This provision exists in Section NB 2311 Summer 1972 Addenda which is referenced in Section NC/ND.	None	Requirements of later Code exempting certain materials from impact test for Class 2 and Class 3 systems.
ASME III, Appendix XVII, 2461.1	80ED	Bolting material allowable stress of non-pressure retention parts (i.e. motor actuator mounting screws).	None	For the analysis of non-pressure retention parts of 1-FCV-72-13 and -34 for Sect. increased valve closure thrust loops. SQN calc. SCG-4M-00786 was reviewed for WBN application. Since the 1971 ASME III NC Codes do not contain stress allowables for non-pressure retention parts, thus ASME III NC and Appendix XVII of 1980 Codes, Sect. 2461.1 rules will be used for the analysis. Ref. DCN-S-33722-A and MEB Calc. EPM-CDM-071092 and -071192.

B. FABRICATION/EXAMINATION/TESTING RELATED

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4121.3	98ED/ 99AD	The use of provisions of NB-4121.3 is desired to support restoration of various Unit 2 components.	None	Completion of Watts Bar Unit 2
NX-4436	80W81	Permits limited welding of attachments to a piping system after it has been hydrostatically tested providing certain conditions are met.	None	Used in G-29 Process Specification 3.M.9.1.
NC-6129	77W78	Provides for alternative testing of inaccessible or embedded welding joints in piping.	None	Used in G-29 Process Specification 3.M.9.1.

TABLE 3.2-7 (SHEET 7 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-6128	77S78	Piping systems which serve as spray systems shall be hydrostatically tested to the rules of this section, except that the test may be performed with the spray nozzle attachment connections plugged. The spray nozzles and any connection beyond the run connections need not be hydrostatically tested.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
NX-6211	80S80	Air pockets in components or systems shall be minimized during the conduct of the hydrostatic test by providing vents at high points, or by flushing the system or by providing calculations to show that the entrapped air is dissolved at the pressure/temperature conditions existing during the test.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
NB-4622.1-1 (TBL)	74ED	PWHT holding time is specified to be: For T less than or equal to 1/2 inch, 30 minutes; For T greater than 1/2 inch but less than or equal to 2 inches (P number 1 & 3), 1 hour per inch; For T greater than 2 inches (P number 1 & 3), 2 hours plus 15 minutes per inch over 2 inches; For T greater than 1/2 inch but less than or equal to 5 inches (P number 4,5,6,7,11,11A), 1 hour per inch; For T greater than 5 inches (P number 4,5,6,7), 5 hours plus 15 minutes per inch over 5 inches. For T greater than 5-inches(P Number 11, 11A) one hour per inch.	None	Used in G-29 Process Specification 2.M.1.1.

TABLE 3.2-7 (SHEET 8 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4623	74ED	For PWHT above 800°F, the rate of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.	None	Used in G-29 Process Specification 2.M.1.1.
NB-4624.3	74ED	For PWHT, the minimum width of controlled band at each side of the weld on the face of the greatest weld width shall be the thickness of the weld or 2 inches whichever is less. (2.M.1.1 uses the minimum from 71S73 and the maximum from 74).	None	Used in G-29 Process specification 2.M.1.1. G-29 uses 2 times the thickness or 2 inches whichever is less. This is more conservative than the Code requires.
NC-4623	74ED	For PWHT above 800°F, the rate of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.	None	Used in G-29 Process Specification 2.M.1.1.
NA	74ED	Deleted requirement of Code of Record Paragraph NB-4623.1 that furnace temperature be less than 600°F when the component is placed in it for PWHT.	None	Used in G-29 Process Specification 2.M.1.1.
NX-4453.1	83S83	Relative to examination of defect removal area (adds the following) this examination is not required where defect removal removes the full thickness of the weld and where the backside of the weld is not accessible for removal of examination materials.	None	Used in G-29 Process Specification 1.M.1.2.

TABLE 3.2-7 (SHEET 9 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4231.2	74ED	Deletes Code of Record requirements for MT/PT examination of temporary attachment removal sites on Class 3 components. Permits temporary attachments but deletes the NDE examinations of removal areas.	None	Used in G-29 Process Specification 1.M.1.2.
NC/ND-6322	80S81	(By reference to NC/ND-6222) If the minimum test pressure of NC/ND-6221(a) or (d) is exceeded by 6% at any location, the upper limit shall be established by analysis using all loadings that may exist during the test.	NC/ND-6221(a) NC/ND-6221(d)	Used in G-29 Process Specification 3.M.12.1
NX-5110	74ED	(By reference to Section V T-732.2(B) which refers to T-733.2) Direct or rectified current shall be used at 700/N to 900/N amperes per inch OD up to 5 inches and 500/N to 700/N amperes per inch OD from 5 to 10 inches and 300/N to 500/N amperes per inch OD over 10 inches where N = # of turns.	T-733.2	Used in G-29 Process Specification 3.M.2.1.
NB-5110	74ED	(By reference to Section V T-733.2.) Direct or rectified current shall be used at 700 to 900 amperes per inch OD up to 5 inches and 500 to 700 amperes per inch OD from 5 to 10 inches and 300 to 500 amperes per inch OD over 10 inches.	None	Used in G-29 Process Specification 3.M.2.1.
NX-4427-1 (FIG)	80S80 80W81 Errata	Minimum size for socket weld fitting fillet weld (CX) = 1.09 T nominal.	None	Used in G-29 Process Specification 3.M.5.1.

TABLE 3.2-7 (SHEET 10 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NX-5110	74S75	(By Reference to Section V T-732.2) For encircling coils, direct or rectified current at 35,000 ampere-turns divided by the sum of 2 plus the L/D ratio of the test part shall be used for magnetization.	T-732.2(A) T-732.2(B)	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74ED	(By reference to Section V T-731.3) Direct or rectified magnetizing current shall be used at a minimum of 100 and a maximum of 125 amperes per inch of prod spacing for sections greater than or equal to 3/4 inch. For sections less than 3/4 inch, amperage shall be 90-110 ampere per inch of prod spacing.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5112	74ED	All NDE performed under this section shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74W74	(By reference to Section V T-630(B)). The penetrant materials are acceptable if the residue does not exceed 0.005 grams or the total sulfur or halogen content shall not exceed 1% of the residue by weight.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5110	74ED	(By reference to Section V T-662). A groove may be machined across the center of each face approximately 1/16 inch deep and 3/64 inch wide, or some other means should be provided to permit side by side comparison without interfering cross contamination between sides.	None	Used in G-29 Process Specification 3.M.1.1.

TABLE 3.2-7 (SHEET 11 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NX-5110	77W79	(By reference to Section V T-660) Permits qualification to be performed by the entire block at each temperature, photographing the results and comparing the photographs.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.7.1.
Section IX, QW-203	74ED	Allows welding procedure qualification in any position to qualify for all position production welds except as noted.	QW-405.2 which requires qualification in the vertical position with upward progression if fracture toughness (impact testing) is a requirement.	Used in G-29 Process Specification 1.M.1.2.
NC/ND-2578	74ED	Unacceptable surface defects may be removed by grinding or machining provided: The remaining thickness is not reduced below the minimum required; the depression after defect elimination, is blended uniformly.	None	Used in G-29 Process Specification 4.M.5.1.

TABLE 3.2-7 (SHEET 12 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4130	74W76	Radiography is not required for welded repairs in material used in components provided that the welds joining these materials are not required to be radiographed, the extent of the welded repair does not exceed 10 square inches of surface area and the magnetic particle or liquid penetrant examination of the repair is made as required by ND-2539.4.	None	Used in G-29 Process Specification 4.M.5.1.
NB-4131 NC-4130	74W74	The time of examination of the weld repairs to weld edge preparations shall be in accordance with NB-5130. (NC-5130).	Since NB/NC-5130d did not exist in summer 1973, these two paragraphs from winter 1974 must be added.	Used in G-29 Process Specification 4.M.5.1.
NX-5110	74ED	(By reference to Section V T-535.1) The primary reference shall be equalized at 50% full screen height.	None	Used in G-29 Process Specification 3.M.7.1.
NB-4620 NB-4651	74ED	Ferritic alloy steel pipe that has been heated for bending shall receive a heat treatment in accordance with NB-4620. Exemptions are listed in Table NB-4622.3-1	None	Used in G-29 Process Specification 4.M.2.1.
Sect V T-263.3	83W83	Provides for not using shims if the radiographic density requirements can be met.	None	Used in G-29 Process Specification 3.M.3.1.
NX-5112	74ED	All nondestructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.3.1.

TABLE 3.2-7 (SHEET 13 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NC/ND-3641.1(a)	71W73	Revise the definition of certain terms in Equation 3 and 4 to permit other means than pipe wall thickness to provide for mechanical strength.		Used in the following Calculations and NCRs: B26850423089, B26850508012, B26850710009, B26850710010, B26850808014, B26860611003, MEB850228009, WBP850211062, NCR4145RRO, NCR2217R. Also involved in Code changes NC-3641(A)-1-3, NC-3641(A)-1-4, NC-3641(A)-1-5, ND-3641(A)-1-3, ND-3641(A)-1-4, and ND-3641(A)-1-6.
NX-4311.3	80W81	Provides for use of capacitor discharge welding of thermocouples and strain gauges.	None	Used in G-29 Process Specification 1.M.4.3.
NC-4244(e)	74ED	Provides alternative for attachment of internally threaded bosses.	None	Used in Calculation B26850613072.
ND-5321	77W77	(By reference to Appendix VI, VI-1132) Provides minimum size for relevant RT indications.	None	Used to disposition CAQR WBP890600.
NX-5342	77ED	Defines nonrelevant condition as any indication with a major dimension of less than 1/16 of an inch. (MT examination).	None	

TABLE 3.2-7 (SHEET 14 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4622.7	74ED	Provides for exemptions to otherwise mandatory post weld heat treatment based on P number, size, carbon content, and preheat.	None	Also invoked by NC/ND-4000.
TBL NX-4622.1-1	74S76	Provides for substitution of 200°F preheat for PWHT when carbon >0.30% and/or tensile strength >70 ksi for certain material applications.	None	Used in disposition of NCR 1146R and in the 47B333 drawings.
NX-4622.3-1 (TBL)	74ED	Provides additional exemptions to PWHT under conditions as described in Table.	New Paragraph NX-4622.3 - defines term “nominal thickness” as used in Table NX-4622.3-1.	Used on the 47B333 Drawings.
NB-5320 & NC-5320	77W77	(By reference to Appendix IV) Provides relevant size and spacing clarification for rounded indications for RT.	None	Provides acceptance criteria for rounded indications for RT
NB/NC/ND-4452	01ED / 03AD	The use of NB/NC/ND-4452 allows defects detected by visual or volumetric method and located on an interior surface need only by reexamined by the method which initially detected the defect when the interior surface is inaccessible for surface examination.		Used in disposition of PER 934436
NB-5250 and NC-5261	77W78	Exempts welds of non-structural attachments from liquid penetrant or magnetic particle examination. Does not exempt the removal area of non-structural or temporary attachments.	None	WBPER910208
NB/NC/ND 6114.2	80S81	Provides exemption to requirement to re-hydro following repair by welding if repair weld is not required to be radiographed per NB/NC/ND-4453.4.	None	Used in G-29 process specification 3.M.9.1

TABLE 3.2-7 (SHEET 15 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-2510(a)	83S83	Seamless pipe, tube, and fittings 1"NPS and less need not be examined by the rules of this subarticle.	None	Use of examination requirements for ASME Section III, Class 1, 1" nominal pipe and smaller from a later edition. Invoke for 1" NPS and less, only.
Table NC-4622.3-1	74S76	Exempts PWHT in Pl materials less than 1 1/2" thick provided 200 degrees F preheat is used.	None	Used to disposition WBP 900419PER
NX-4427	80W82	Allows fillet welds to be undersize 1/16 inch for 10% of length of weld.	None	Weld Project
NCA 1273	80S80	Exempts orifice plates not exceeding 1/2" nominal thickness which are clamped between flanges and used for flow measuring service only from the code.	None	Orifice Plates
NB-2538.4	77S78	Areas ground to remove oxide scale or other mechanically caused depressions...need not be PT or MT examined.	None	Provisions for not NDE examining impressions caused by mechanical means.
NB-4622.9	89ED	Increases the area which can be repaired by temper bead to 100 in2 and up to 1/3 of base material thickness.	None	Steam generator manway repair or other locations as required.

C. QUALITY ASSURANCE RELATED

NCA-4000	80W81	Provided more detailed QA program requirements.	None	Used in the NCM
NCA-8240	83S83	Provides alternatives when code data plates are removed.	None	Used in disposition of WBP880052, NCR 5577R0, NCR 5611R0, and NCR 5619R0.

TABLE 3.2-7 (SHEET 16 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NCA-8240(b)	80S81	Provides alternatives when code data plates are removed.	None	Use in disposition of NCR 3951R.
NX-2610b and c	77ED	Makes provision to exempt manufacturers of small products from certain QA program` requirements.	None	Used in Disposition NCR GENMEB8402, NCR 5146-0, WBP880431, WBP889432, WBP880433, WBP880437, and WBP880438.
NCA 3800	89ED	Metallic material manufacturers and material suppliers Quality System program provides for non certificate holders QA Manual revision and date to be placed on the documentation and also provides documentation requirements for 1" bar stock.	None	QA programmatic requirements for vendors supplying materials and components.
N-801	NA	Rules for repair of N-stamped Class 1, 2 and 3 components by organization other than the N-certificate holder that originally stamped the component being repaired Section III, Division 1.	None	This code case will allow N-certificate holders at WBN Unit 2 to perform repairs of N-stamped components in support of N-5 code data report completion. These components were installed between 1971 to the present. Many of the older components require refurbishment to meet or exceed design or code requirements. Refurbishment may require replacement of code and non-code material or repair of pressure boundary material. Because the original N-certificate holder that stamped the component may no longer exist or maintain an N-certificate, relief is necessary to perform the required work. Code Case N-801 will be applied at WBN Unit 2 to the repairs accomplished by the N-certificate holders at WBN Unit 2.

TABLE 3.2-7 (SHEET 17 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
N-802	NA	Rules for repair of stamped components by the N-certificate holder that originally stamped the component Section III, Division 1.	None	This code case will allow N-certificate holders that stamped components to perform repairs of these components in support of N-5 code data report completion. These components were installed between 1971 to the present. The original N-certificate holder may need to refurbish these components to meet or exceed design or code requirements. Refurbishment may require replacement of code and non-code material or repair of pressure boundary material. Code Case N-802 will be applied at WBN Unit 2 to the repairs accomplished by the N-certificate holder that originally stamped the component. This includes stamped components shipped to the N-certificate holder's facility or those repaired on site if the holder's certificate extends to the WBN Unit 2 site.

3.3 Wind and Tornado Loading

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The Seismic Category I structures are designed for a 95-mile per hour wind, 30 feet above grade, with a 100 year recurrence interval. The wind was determined from Figure 1, ASCE paper 3269, "Wind Forces on Structures."[3] ANSI A58.1-1972, 'Building Code Requirements for Minimum Design Loads in Building and Other Structures'[4] is an acceptable alternative for determining design wind velocities and forces.
The wind was applied for the full height of the structure. A gust factor of 1.1 is included for all wind loads and combinations of loads where wind is involved as recommended in ASCE paper 3269.[3]

3.3.1.2 Determination of Applied Force

The pressure and pressure distribution of wind loads on structures were determined by the methods described in ASCE Paper 3269.[3] The dynamic wind pressure, q, is defined as $q = 0.00256V^2$, where q is in psf and V is in mph. A gust factor of 1.1 is applied which redefines q as $q = 0.00256\ (1.1V)^2 = 0.00310V^2$. The wind pressure, p, in psf, is defined as $p = Cq$, where C is the pressure distribution coefficient (Cpe or Cpi) or the shape coefficient (C_D) determined from Table 4 in ASCE Paper 3269.[3]

For the analysis of box-shaped structures, a shape coefficient (C_D) of 1.3 is used which defines the wind pressure as p = 1.3q. Of the total pressure (p = 1.3q), 0.8q is applied to the windward wall, and 0.5q is applied to the leeward wall. Concurrently the end walls receive 0.7q negative pressure and the roof receives 0.5q uplift.

For the analysis of cylindrical structures, such as the Shield Buildings and storage tanks, the shape coefficients and pressure distribution coefficients are obtained from Table 4(f) of ASCE Paper 3269.[3]

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

All Category I structures except for the additional Diesel Generator Building are designed for a "funnel" of wind moving with a translational velocity of 60 miles per hour and having a rotational velocity of 300 miles per hour. Category I structures are also designed for an external depressurization of 3 psi occurring in 3 seconds.

The tornado loading for the Additional Diesel Generator Building and structures initiated after July 1979 is discussed in Section 2.3.1.

Information about the spectrum and pertinent characteristics of tornado-generated missiles is in Section 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

The pressures and pressure distribution of wind forces on Category I structures due to tornado wind loadings were determined by following the recommendations of ASCE Paper 3269, 'Wind Forces on Structures'.[3] ANSI A.58.1-1972, 'Building Code Requirements for Minimum Design Loads in Building and Other Structures'[4] is used to provide an alternate method to determine tornado wind loads. The provisions for gust factors and variations of wind velocity with height are not applied. The dynamic wind pressure, q, is defined as $q = 0.00256V^2$, where q is in psf and V is in mph. The wind pressure, p, in psf, is defined as p = Cq, where C is the shape coefficient (C_D).

A 1.3 shape coefficient is included for box-shaped structures with vertical walls normal to the wind direction. The dynamic pressure load, p = 1.3q, due to tornadoes is applied to the structure walls and roof in the same manner as the wind loads in Section 3.3.1.

Cylindrical structures and tanks have the same shape coefficients applied as for wind loads in Section 3.3.1. The pressures are applied over the structures as shown in Table 4(f) of ASCE Paper 3269.[3]

The loadings of the wind force and the depressurization are considered to act concurrently. Coincident wind velocities and pressure drops for the design tornado are shown in Figure 3.3-1. The relationship between wind velocity and pressure in the design tornado shown in Figure 3.3-1 was developed based on Hoecker's studies of the Dallas tornado of 1957.[1,2]

Venting, when used as a design procedure for reducing the tornado-generated differential pressure, is accomplished by using blowoff panels that fail at a lower differential pressure. Upon relief of the differential pressure by the blowoff panel from the exterior wall of a room, the interior walls and slabs of these rooms are designed for the 3 psi pressure differential.

The effective loads on Category I structures due to tornado-generated missiles were determined using the procedures described in Section 3.5.3.

The effect of various combinations of tornado loadings were studied with respect to each Category I structure. The most adverse combination was selected individually for the design basis of each structure.

The tornado loadings are not considered to be coincident with accident or earthquake loadings.

Venting is utilized to reduce the effective tornado-generated differential pressure in portions of the Auxiliary Building. Four hundred square feet of relief panel area are provided in the roof over the spent fuel pool room and cask loading room at Elevation 814.75 for venting purposes during the tornado. The relief panels are held in place by gravity. An upward pressure of 0.25 psi is sufficient to offset the weight of the panels and cause them to be lifted from their nominal positions. Two corners of each panel are chained to the roof to prevent the panel from becoming a missile after it relieves.

The shutdown board room and, in general, the area between columns q and u at Elevation 757.0 is not part of that portion of the Auxiliary Building vented by design; however, the remainder of the building is considered to depressurize due to the vent area provided by the air intake openings and through ventilation penetrations. In addition, the Diesel Generator Building and the Intake Pumping Station are designed to depressurize due to the vent areas provided by the ventilation openings in those buildings.

The roof and exterior walls of the spent fuel pool room and cask loading area were evaluated for the effective tornado-generated pressure differential and were found to be within allowable stress limits. Air velocity induced by venting is expected to be high at the vent opening, but decrease rapidly within a few feet of the opening. No hazard to equipment is foreseen since the vents are located in the Auxiliary Building roof, well away from any essential equipment.

No hazard to equipment in these areas is foreseen due to the small pressure differential and low air velocities. Walls, ceilings, and floors separating areas experiencing depressurization during a tornado from areas not experiencing depressurization are designed to withstand the total tornado-generated pressure differential of 3 psi.

The analytical model employed in determining the effective differential pressures utilizes isentropic, perfect gas relations in a step-wise, steady-state first law analysis. The analysis determined pressure and temperature variations within the structure induced by the design base tornado defined in Section 3.3.2.1.

Pressure differentials and assorted air velocities are expected in all areas which depressurize due to the vented design of the building. In these areas, the partition walls have been checked for the differential pressure from depressurization. In the room(s) where the differential pressure exceeds the wall design, administrative operating instructions ensure that the doors will remain open during a tornado event to reduce the differential pressure to an acceptable value.

3.3.2.3 Ability of Category I Structures to Perform Despite Failure of Structures Not Designed for Tornado Loads

An investigation of the effect of tornado loading on the Turbine Building was made to determine the extent of failure of the structure as to collapse or to the possibility of generating missiles that could damage Category I structures and impair their ability to perform their intended design function.

The following information was determined:

1. The metal siding panels will fail at loads considerably below the design tornado loading and will become missiles that could affect the Control Building. The siding will fail before the main girts are overloaded enough to cause failure. The failure of the parapet girts is likely, resulting in the release of 6WF15.5 in 4-foot lengths, 8C11.5 in 8-foot lengths, 18-inch x 3/8-inch plate in varying lengths, and 4ST8.5 in 7-foot lengths.

   The roof of the Control Building was investigated for the above missiles and found to be adequately designed to resist the missiles.

2. Following the failure of the siding, the structural steel framing of the building will be exposed to tornado forces acting upon the steel structure, equipment, piping, and other items of wind resistance. The resistance of the structure at this point will be sufficient to prevent collapse onto the Control Building.

3. The turbine room cranes, if not anchored, could possibly be blown from the crane girders, either falling on the operating floor or out the end of the building onto the Control Building roof.

   To preclude the occurrence of this event, the cranes will be anchored to stops at one end of the runway during tornado alerts, watches, and tornadoes.

4. The potable water tanks and gland seal water tanks at Elevation 796.0 floor could be blown to the Control Building roof along with air intake hoods, auxiliary boiler stack, and heating and vent equipment on the Elevation 796.0 floor.

   The Control Building roof was determined to be adequately designed to resist the described events.

The methods used to convert the tornado wind loadings into loads acting on the structures, as well as their distributions across the structures, were determined by following the recommendations of ASCE Paper 3269.[3]

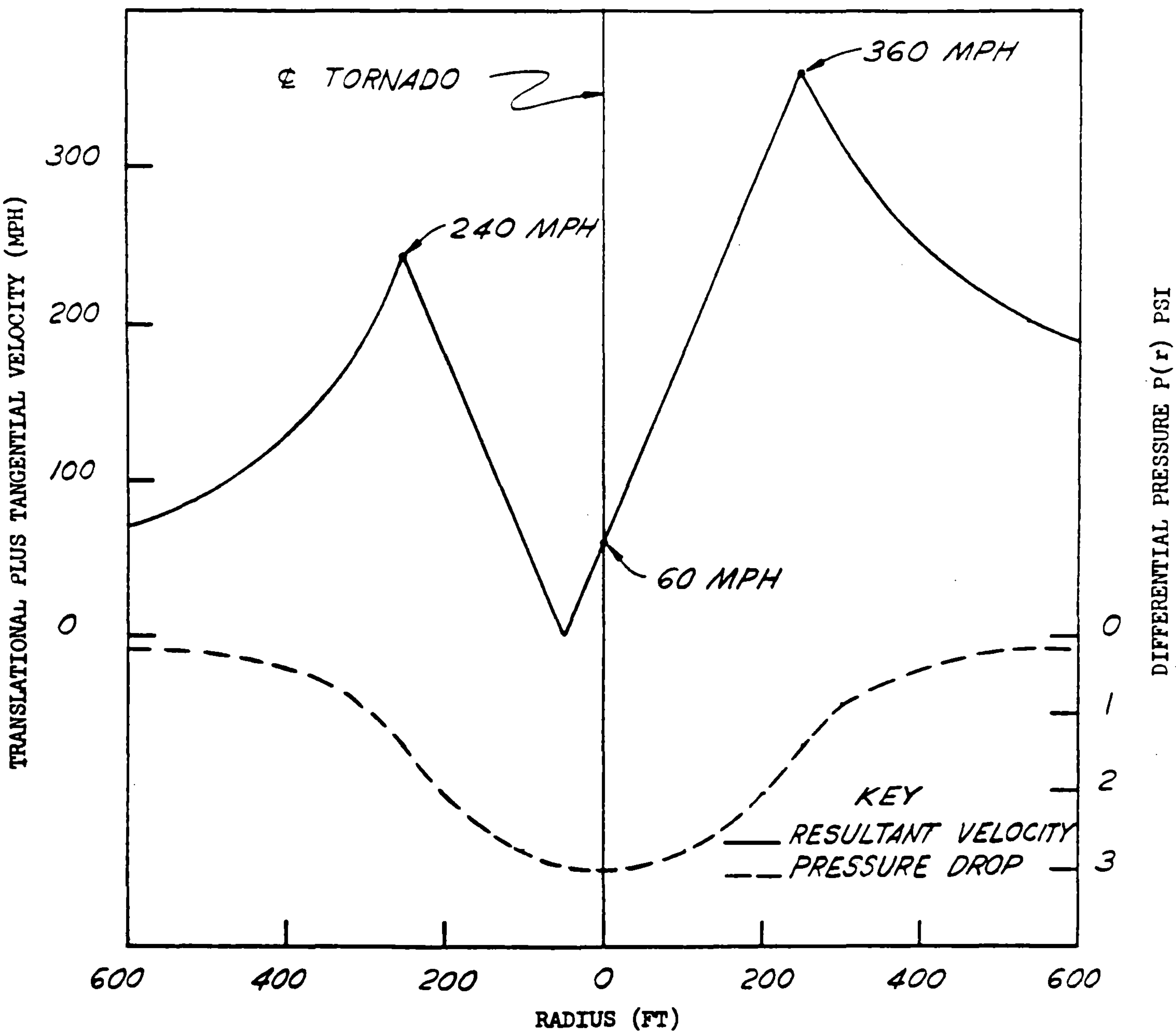
REFERENCES

1. Hoecker, W. H., 'Wind Speed and Air Flow Patterns in the Dallas Tornado and Some Resultant Implications', Monthly Weather Review, May 1960.

2. Hoecker, W. H., 'Three Dimensional Pressure Pattern of the Dallas Tornado and Some Resultant Implications,' Monthly Weather Review, December 1961.

3. 'Wind Forces on Structures', Final Report, Task Committee On Wind Forces, Committee on Loads and Stresses, Structural Division, Transactions, American Society of Civil Engineers, Publication Number 3269, Volume 126, Part II, (1961).

4. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Building and Other Structures," Committee A58.1, American National Standards Institute, 1972.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Variations of Differential Pressure and
Tangential Plus Translational Velocity
as a Function of the Distance
from the Center of a Tornado

FIGURE 3.3-1

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

The flood protection requirements and provisions for Category I structures are discussed in Sections 2.4.1.1, 2.4.2.2, 2.4.2.3, and 2.4.10.

3.4.2 Analysis Procedure

The methods and procedures by which the static and dynamic effects of the design basis flood conditions are applied to Category I structures are discussed in Sections 2.4.3.6 and 2.4.13.5.

REFERENCES

None.

3.5 MISSILE PROTECTION

Category I structures have been analyzed and designed to be protected against a wide spectrum of credible missiles. Failure of certain rotating or pressurized components of equipment is credible and would presumably lead to generation of missiles. In addition, noncredible missiles are identified and justification is given for their not being a credible source of missiles. Tornado-generated missiles and missiles resulting from activities peculiar to the site are also discussed in this section. It is shown that the missile protection criteria to which the plant has been analyzed and protected comply with the intent of Criterion 4 of 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants.

A very basic premise for protection is to design components and equipment so that they will have a low potential for generation of missiles. In general, the design that results in reduction of missile generation potential promotes the long life and usability of a component, and is well within permissible limits of accepted codes and standards. The following general methods are used in the design, manufacture, and inspection of equipment:

1. Pressurized equipment and sections of piping that from time to time may become isolated under pressure have been provided with pressure relief valves. (Relief valves are in accordance with ASME Section III or the appropriate industry standards.) These valves are present to ensure that no pressure buildup in equipment or piping sections will exceed the design limits of the materials involved.

2. Components and equipment of the various systems have been designed and built to the standards established by the ASME or other equivalent industrial standard. A stringent quality control program has been enforced during manufacture, testing, and installation.

3. Volumetric and ultrasonic testing where required by code, coupled with periodic inservice inspections of materials used in components and equipment, adds further assurance that any material flaws that could permit the generation of missiles will be detected.

The design bases to which the plant has been designed in order to meet the intent of the criterion are listed below.

Design Bases

1. Protection shall be provided against potential missiles that could cause a loss-of-coolant accident (LOCA).

2. Protection shall be provided against potential missiles that could result in the loss of ability to control the consequences of a LOCA, including both the necessity for core cooling and for retention of containment integrity.

3. Protection shall be provided against potential missiles that could jeopardize functions necessary to bring the reactor to a safe shutdown condition during normal or abnormal conditions.

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

The structures that are to be protected against damage from internally-generated missiles outside containment are listed in Table 3.2-1. The systems and components that are to be protected against damage from internally generated missiles outside containment include the ANS Safety Class equipment listed in Tables 3.2-2a, 3.2-2b, and 3.2-3.

A discussion of the individual structures and the systems and components inside these structures is provided below. In general, the systems and components important to safety located in the structures of Table 3.2-1 rely on redundancy and separation for protection from internal missiles generated by failure of high pressure system components. Missiles which result from pipe breaks are not considered. Missiles that are associated with overspeed failures of rotating components are considered a greater safety hazard than those generated by failure of high-pressure system components and are evaluated in more detail.

3.5.1.1.1 Shield Building

No rotating components which might generate missiles are installed between the primary containment and the Shield Building. No high-pressure system components whose failure could generate missiles are located in this area.

3.5.1.1.2 North and South Steam Valve Rooms

No rotating components are installed in these rooms. The barriers provided to limit pipe whip and jet impingement in these rooms also provide protection from any potential missiles generated by failure of any high-pressure system components. The main steam isolation valve stems and the main feedwater isolation valve stems are not postulated as credible missiles in these rooms as explained below.

Numerous features of the main steam isolation valve (MSIV) design and construction serve to prevent the valve stem from being ejected as a missile. The MSIV's are 32 inch wye, bi-directional globe valves that are air-opened, spring-closed, and qualified to ASME Section III, Class 2. Several components of the valve and operator would have to fail concurrently before the valve stem could be ejected. First, the pilot poppet is fastened onto the bottom end of the valve stem and backseated against the poppet cap. The lower segment of the valve stem has a larger diameter than the opening provided in the bonnet, the bottom spring seat, and the bottom bead of the air cylinder that the upper segment of the valve stem and the piston rod normally operate through. The worst case failure tending to cause MSIV stem ejection would be the complete severance of the valve stem itself at the valve bonnet. This failure would eliminate the components and design features described thus far as barriers to valve stem ejection. The internal area of the stem, tending to eject it. However, the valve operator is designed to withstand the transmitted forces acting through the stem during the normal operation of the valve. These thrust forces are at least as great as those that would act on a broken stem. Thus, several additional components of the valve and operator would still have to fail before even this fractured segment of the valve stem could be ejected. First, the top of the valve stem is fastened into the bottom spring seat cap which is in turn bolted to the bottom spring seat. The air cylinder piston rod is located directly over the valve on the same axis and is inserted through the opening in the air cylinder bottom bead. The air cylinder piston and top head (each 3 inch plate) are also located over the valve stem, perpendicular to its axis. The combination of these valve and operator features precludes the MSIV stem (or a segment of it) from being postulated as a credible missile.

Similarly, various features of the main feedwater isolation valve (MFWIV) design and construction serve to prevent its valve stem from being ejected as a missile. The MFWIV's are 16 inch bolted bonnet, wedge gate valves with motor operators and are qualified to ASME Section III, Class 2. Several components of the valve and operator would have to fail concurrently before the stem could be ejected. First, the wedge gate holds the bottom end of the valve stem in a close-fitting ball-and-socket type arrangement. The lower segment of the valve stem itself is backseated against the valve bonnet. This segment has a larger diameter than the openings provided in the bonnet, the upper and lower yoke plates, and the motor operator that the upper segment of the valve stem normally operates through. The worst case failure tending to cause MFWIV stem ejection would be the severance of the stem itself above the backseat segment just described. For the same reasons as given previously for the MSIV, additional components of the MFWIV operator would still have to fail before even this fractured segment of the valve stem could be ejected. First, the upper stem segment is geared into the motor-driven worm shaft of the operator. Then, a pipe cap is bolted on top of the operator unit over the valve stem and perpendicular to its axis. This combination of valve and operator features precludes the MFWIV stem (or a segment of it) from being postulated as a credible missile.

3.5.1.1.3 Auxiliary Building

The only rotating component which was considered for overspeed condition is the auxiliary feedwater steam-driven pump. All other pumps are electrically driven and incapable of achieving an overspeed condition. The manufacturer of the steam turbine (Terry Turbine) has indicated that they have tested their solid wheel turbine under overspeed conditions and no missiles are generated. The pump itself (manufactured by Ingersoll-Rand) may develop missiles under overspeed conditions but its potential for damage is small because of the small size of any missiles postulated. The room containing the pump is oriented in such a manner as to minimize the potential for damage caused by postulated pump missiles.

Consideration of missiles associated with failure of high-pressure system components is handled by redundancy and separation of safety-related systems.

The internal walls and floors of the Auxiliary Building are constructed of reinforced concrete which limit the range of any potential internal missiles. In particular, the spent fuel pool is protected by at least one wall or floor of reinforced concrete from internal missiles generated in other parts of the Auxiliary Building.

The portions of the CVCS and SIS outside of containment are physically separated and protected by concrete barriers of sufficient strength to contain any postulated internally generated missiles.

3.5.1.1.4 Control Building

There are no credible potential internal missiles in this building. There are no rotating components which could have an overspeed failure and no high-pressure systems. The carbon dioxide fire protection system inside the Control Building is not pressurized until it is actuated.

3.5.1.1.5 ERCW Structures

At the Intake Pumping Station the essential raw cooling water (ERCW) pump motors are exposed to the atmosphere. A structural steel grillage system, discussed in Section 3.8.4, provides protection to the pumps from tornado missiles. A concrete shield wall separates the four motors of Train A from those of Train B. These components are arranged in a straight line over a distance of about 100 feet. An overspeed failure is not postulated for these pumps. Even if a failure were postulated, no credible trajectory of any resultant missile could damage enough components to reduce the number available to less than four. No credible failure of any high-pressure component could create a missile which could reduce the availability of pumps on the opposite power train.

No credible, potential internal missile sources are installed in the remainder of the ERCW structure.

3.5.1.1.6 ERCW Pipe Tunnels and RWST Foundations

No credible potential internal missile sources are installed in these structures.

3.5.1.1.7 Diesel Generator Building

Four emergency diesel generators, which are required to supply emergency power to certain engineered safety features, are each located inside a separate room in the Diesel Generator Building. Interior walls of reinforced concrete separate these generators.

There is a mechanical governor on the diesel engine of each diesel-generator unit which is designed to assume control of the engine when there is a tendency to overspeed. In addition, the diesel generators have an overspeed trip which cuts off fuel to the diesel engine upon an overspeed condition. Consequently, no missiles are postulated for overspeed conditions of the generator. The diesel generator units are protected from the effects of a postulated failure of the carbon dioxide storage tank by an 18-inch thick reinforced concrete wall. Therefore, any missiles or pressure build-up generated by a rupture of the carbon dioxide storage tank would not damage essential equipment.

The vent path for the carbon dioxide storage tank compartment is through one set of standard double doors into a stairwell. If additional pressure relief is required, the vent path is through another set of standard double doors which open to the atmosphere from the stairwell.

3.5.1.2 Internally Generated Missiles (Inside Containment)

3.5.1.2.1 Missile Selection

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, and piping leading to generation of missiles is not considered credible. Massive and rapid failure of these components is not postulated because of the material characteristics and inspections; quality control during fabrication, erection, and operation; conservative design; and prudent operation as applied to the particular component. The reactor coolant pump flywheel is not considered as a source of missiles for the reasons discussed in Section 5.2.6.

Nuclear steam supply components, which nevertheless are considered to have a potential for missile generation inside the reactor containment, are the following:

1. Control rod drive mechanism housing plug, drive shaft, and the drive shaft and drive mechanisms latched together.

2. Certain valves.

3. Temperature sensor assemblies.

4. Pressurizer instrument well and heaters.

Gross failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

1. Full length control rod drive mechanisms are shop tested at 4105 psig.

2. The mechanism housings are individually hydro-tested to 3107 psig as they are installed on the head adapters of the reactor vessel and they are checked during the hydro-test of the completed reactor coolant system.

3. Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops.

4. The mechanism housings are made of type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

However, it is postulated that the top plug on the control rod drive mechanism will become loose and be forced upward by the water jet. The following sequence of events is assumed: The drive shaft and control rod cluster are forced out of the core by the differential pressure of 2500 psi across the drive shaft; the drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts; after approximately 12 feet of travel, the rod cluster control spider hits the underside of the upper support plate; upon impact, the flexure arms in the coupling joining the drive shaft and control cluster fracture, completely freeing the drive shaft from the control rod cluster. The control cluster would be completely stopped by the upper support plate; however, the drive shaft would continue to be accelerated upward to hit the missile shield provided. This analysis is summarized in Table 3.5-1. For a description of the missile shield see Section 3.5.1.2.6.

Valve stems are considered to be extremely unlikely sources of missiles because of the design, construction qualifications, and testing of the valves. The isolation valves installed in the reactor coolant system have stems with a back seat. This effectively eliminates the possibility of ejecting valve stems even if the stem threads fail. Analysis shows that the back seat or the upset end would not penetrate the bonnet. Additional interference is encountered with air and motor operated valves.

Valves with nominal diameter larger than 2 inches have been designed against bonnet body connection failure and subsequent bonnet ejection by means of:

1. Using the design practice of ASME Section VIII which limits the allowable stress of bolting material to less than 20% of its yield strength;

2. Using the design practice of ASME Section VIII for flange design; and

3. By controlling the load during the bonnet body connection stud tightening process.

The pressure containing parts are designed per code Class 1 requirements established by the ASME Section III Code.

The proper stud torquing procedures and the use of a torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the ASME Code.

This stress level is far below the material yield. The complete valves are hydrotested per the ASME Section III Code. The stainless steel bodies and bonnets are volumetrically and surface tested to verify soundness.

Valves with a nominal pipe size of 2 inches or smaller are forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the ASME III Code specification.

Whereas valve missiles are not generally postulated as outlined in the above discussion, it was decided to conservatively review valves as though their bonnets or stems could become missiles. Representative of these are the valves in the region where the pressurizer extends above the operating deck. Valves in this region are the pressurizer safety valves, the motor-operated isolation valves in the relief line, the air-operated relief valves, and the air-operated spray valves. Although failure of these valves is unlikely, provisions are made to assure protection of safety-related equipment, including the containment liner, from hypothetical missiles from these valves (see Table 3.5-2).

The potential for other valve missiles in the lower compartment to damage safety equipment is assessed to be extremely limited. The same measures taken to prevent damage from other postulated non-valve missiles will be effective against damage from these hypothetical valve missiles. These measures include layout of the basic plant arrangement utilizing the design philosophy of physical separation of equipment and components through distance or by barriers. In consideration of the postulation that valve bonnet fragmentation may occur resulting in the ejection of a valve stem, valves are oriented such that any missile will strike such a barrier.

The only other postulated jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is that represented by the temperature sensor assemblies, as listed in Table 3.5-3. The resistance temperature sensor assemblies are of two types: 'with well' and 'without well'. Two rupture locations have been postulated: around the weld (or thread) between the temperature element assembly and the boss for the 'without well' element, and the weld (or thread) between the well and the boss for the 'with well' element.

A temperature sensor is installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end of a steel plate. In evaluating missile potential, it is assumed that this plate could break and the pipe plug on the external end of the hole could become a missile.

In addition, it is assumed that the weld between the instrumentation well and the pressurizer wall could fail and the well and sensor assembly could become a jet-propelled missile.

Finally, it is assumed that the pressurizer heaters could become loose and become jet-propelled missiles. Adequate barriers are provided for the missiles above to protect safety-related equipment.

3.5.1.2.2 Missile Description

The postulated control rod drive mechanism (CRDM) missiles are summarized in Table 3.5-1. The velocities of the missiles have been calculated using the method shown in Appendix 3.5A. The reactor coolant discharge rate from the break has been calculated using the Burnell equation.[1] The coolant pressure has been assumed constant at the initial value. No spreading of the water jet has been assumed.

The missile characteristics of the bonnets of the typical valves in the region where the pressurizer extends above the operating deck are given in Table 3.5-2.

The missile characteristics of the postulated piping temperature sensor assemblies are given in Table 3.5-3. A 10-degree expansion half-angle water jet has been assumed. The missile characteristics of the piping pressure element assemblies are less severe than those of Table 3.5-3.

The characteristics of other missiles postulated within reactor containment are given in Table 3.5-4. A 10-degree expansion half-angle water jet has been assumed.

3.5.1.2.3 Electrical Cables

Electrical cables are not protected against damage from internal missiles. However, separation and redundancy of vital cables are such that any single failure within the protection system will not prevent proper protective action at the system level when required.

3.5.1.2.4 Upper Compartment

The generation of internal missiles inside the upper compartment is not postulated. Piping in this area is not used during normal operation. The upper compartment is protected from missiles generated in the lower compartment by the steel reinforced divider deck and missile shield and by the reinforced concrete walls of the upper portions of the steam generator and pressurizer compartments.

3.5.1.2.5 Ice Condenser Compartment

The generation of internal missiles inside the ice condenser is not postulated. Low energy refrigeration piping is not considered to be a potential missile source. Missiles generated in portions of the lower compartment will be prevented from entering the ice condenser compartment by the crane wall. The trajectory of missiles generated within the bottom regions of the lower compartment is such that the missiles will not pass through the inlet door openings in the lower crane wall except by ricochet. The potential for damage of such a ricocheting missile is considered negligible. This situation is shown in Figure 3.5-1. As can be seen in the figure, the location of the main portions of the reactor coolant system and of the other systems which connect to it are below Elevation 718 feet-0 inches, whereas the openings for the ice condenser lower inlet doors are between Elevation 746 feet-5 inches and Elevation 753 feet-9inches.

3.5.1.2.6 Lower Compartment

The spectrum of missiles generated within the lower compartment is discussed in Sections 3.5.1.2.1 and 3.5.1.2.2. These missiles will not cause failure of vital systems inside the lower compartment. Particular attention was paid to the potential missile damage to the steel containment structure, the emergency core cooling system, and the containment isolation system.

Any missile generated within the lower compartment will not impair the integrity of the steel containment structure. Protection against the postulated missiles in the lower compartment was accomplished by locating a reinforced concrete wall (crane wall), a steel-reinforced concrete slab (divider deck), and steel-reinforced concrete removable blocks (control rod drive mechanism missile shield) between the primary reactor coolant system and the containment structure. Additionally, since there are openings in the crane wall, protection for the containment structure is enhanced by orienting potential missile sources, especially valve components, so that their anticipated trajectory will not permit them to pass through these openings.

The control rod drive mechanism missile shield has been located above the reactor vessel and will prevent the postulated missiles of the control rod drive system from striking the inside surface of the containment structure or the containment spray headers (see Table 3.5-1).

The accumulator tanks and associated check valves and piping are not credible sources of missiles for the containment structure. Components are prevented from becoming a source of damaging missiles by orienting the components so that anticipated missile trajectories are away from the containment structure. Some other high-pressure system components are located in the space between the crane wall and the steel containment structure. Protection is accomplished by orienting components so that the anticipated missile trajectories are away from the containment structure.

The emergency core cooling system (Section 6.3) includes four accumulator tanks which are located in separate rooms between the crane wall and the containment structure. The crane wall protects these tanks and their associated valves and piping from the postulated missiles generated within the lower compartment, and the Shield Building protects them from external missiles. The active components of the system (pumps, motors, and heat exchangers) are located in separate rooms in the Auxiliary Building. Therefore, these active components are protected from the postulated missiles generated within the lower compartment.

Isolation valves of the containment isolation system (Section 6.2.4) are located in three regions: 1) inside the containment structure, 2) between the containment structure and the Shield Building, and 3) outside the Shield Building. The isolation valves which are located inside the containment structure are protected from the postulated missiles generated in the lower compartment by the crane wall, and are protected from tornado-generated missiles by the Shield Building.

Even though the preceding methods have been used to protect the containment structure, emergency core cooling system, and the containment isolation system from potential internal missiles, the basic approach was to assure design adequacy against generation of missiles rather than to allow a missile to be generated and then try to contain the effects.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

The Watts Bar turbine-generators were originally manufactured by Westinghouse Electric Corporation (now Siemens Energy, Inc.).The Watts Bar turbine-generator unit consists of a double-flow high pressure turbine and three double-flow low pressure turbines with extraction nozzles arranged for seven stages of feedwater heating. The turbine utilizes a Westinghouse designed electrohydraulic control (EHC) system for control of both speed and load. The EHC system, composed of solid state electronic devices coupled through suitable electrohydraulic transducers to a high-pressure hydraulic fluid system, provides control of the main stop, governing, intercept, and reheat stop valves of the turbine. Emergency speed protection is provided by a mechanical overspeed trip mechanism, backed up by an electrical overspeed trip circuit.

Additional turbine protection is provided that will trip the turbine on evidence of low condenser vacuum, abnormal thrust bearing wear, or low bearing oil pressure (see Section 10.2.4 for a complete list). The turbine trip system is also equipped with solenoid-operated trip devices, which provide a means to initiate direct tripping of the turbine upon receipt of appropriate electrical signals, as shown In Figure 10.2-1. Also, when a turbine trip is initiated, the extraction system nonreturn valves are tripped to close by means of a pilot dump valve connected to the turbine trip system. For overpressure protection of the turbine exhaust hoods and the condenser, four rupture diaphragms which rupture at approximately 5 psig are provided on each turbine exhaust hood. Additional protective devices include exhaust hood high temperature alarm and manual trip.

Each stop, governing, reheating stop, and intercept valve is spring closed; therefore, it is necessary only to dump the high pressure fluid from under the servoactuators to close the valves.

For additional details on the turbine, see Section 10.2.

3.5.1.3.2 Potential Missile Sources and Missile Characteristics

The most significant source of turbine missile is a burst-type failure of one or more bladed shrunk-on disks of the low-pressure (LP) rotors.

Failures of the high-pressure (HP) and generator rotors would be contained by relatively massive and strong casings, even if failure occurred at maximum conceivable overspeed of the unit. Evaluations of existing and new retrofit HP rotors were performed considering various failure modes to assess the potential for generating a missile [23]. Ductile burst would require rotational speed beyond terminal speed as explained. Failure due to high cycle fatigue has not occurred in the past and the retrofit rotors have improved design safety factors. Failure due to low cycle fatigue (LCF) is unlikely since rotor LCF is significantly greater than 10,000 start cycles for original and retrofit rotors. Based on the successful operating history of nuclear HP rotors and the results of this evaluation, HP rotors of integral construction od no need to considered when assessing missile generation probability of nuclear turbines.

There is a remote possibility that some minor missiles could result from the failure of couplings or portions of rotors which extend outside the casings. These missiles would be much less hazardous than the LP disk missiles, due to low mass and energy and therefore, have not been considered.

The probability of a turbine missile generation (P1) is evaluated by conservatively considering two distinct types of LP shrunk-on disk failures, namely:

1. Failure at normal operating speed up to 120% of the rated speed and
2. Failure due to run-away overspeed greater than 120% of the rated speed for all disks.

Normal operating speed conditions are expected to occur one or more times per year of operation, but run-away overspeed conditions are expected to occur rarely. Missiles resulting from normal operating speed failures are the result of the brittle fracture of turbine blade wheels or portions of the turbine rotor itself. Failures of this type can occur during startup or normal operation. Missiles resulting from run-away overspeed failures would be generated if the overspeed protection system malfunctioned and if the turbine speed increased to a point at which the low-pressure wheels or rotor would undergo ductile failure.

The probability of normal operating speed reaching the value up to 120% of the rated speed is assumed to be 1.0, conservatively. It is also conservatively assumed that, given the overspeed protection system fails, the probability of a disk burst and that of casing penetration of the burst fragments is 1.0 for all disks. The overspeed probability is directly correlated to the inspection interval as well as the test frequency of the overspeed protection system.

Missiles from a turbine failure can be divided into the following two groups:

1. "low-trajectory" or "direct" missiles, which are ejected from the turbine casing directly toward an important to safety SSC, and
2. "high-trajectory" missiles, which are ejected upward through the turbine casing and may cause damage if the falling missile strikes an important to safety SSC.

Low Pressure Turbine Construction and Design

Each unit has three (3) LPs. On Unit 1, LPA and LPB are Westinghouse BB281 10-disc design rotors. LPC is a Siemens 13.9m^2 damped blade, advanced disc design (6-disc) rotor. All three LPs of Unit 2 have the advanced disc design rotor with 13.9m^2 free standing blades.

The double flow low pressure turbine incorporates high deficiency blading, diffuser type exhaust and liberal exhaust hood design. The low pressure turbine cylinder is fabricated from steal plate to provide uniform wall thickness thus reducing thermal distortion to a minimum. The entire outer casing is subjected to low temperature exhaust steam.

For LPA and LPB (Unit 1), the temperature drop of the steam from its inlet to the LP turbine to its exhaust from the last rotating blades is taken across three walls: an inner cylinder No. 1, a thermal shield, and an inner cylinder No. 2. For LPC (Unit 1) and the Unit 2 LPs, the temperature drop is across two walls: an inner casing and a thermal shield. This precludes a large temperature drop across any one wall, except the thermal shield which is not a structural element, thereby virtually eliminating thermal distortion. For LPA and LPB (Unit 1), the fabricated inner cylinder No. 2 (inner casing for LPC (Unit 1) and Unit 2 LPS) is supported by the outer casing at the horizontal centerline and is fixed transversely at the top and bottom and axially at the centerline of the steam inlets, thus allowing freedom of expansion independent of the outer casing. For LPA and LPB (Unit 1), inner cylinder No. 1 is, in turn, supported by inner cylinder No. 2 at the horizontal centerline and fixed transversely at the top and bottom and axially at the center line of the steam inlets, thus allowing freedom of expansion independent of inner cylinder No. 2. For LPA and LPB, inner cylinder No. 1 is surrounded by the thermal shield. For LPC (Unit 1) and the Unit 2 LPs, the inner casing is surrounded by the thermal shield. The steam leaving the last row of blades flow into the diffuser where the velocity energy is converted to pressure energy.

For LPA and LPB (Unit 1), the outer cylinder and the two inner cylinders are fabricated mainly of ASTM 515-GR65, or equivalent material. For LPC (Unit 1) and Unit 2 LPs, the outer cylinder and inner casing are fabricated mainly of ASTM 515-GR65, or equivalent material.

The low-pressure rotors are made of NiCrMoV alloy steel.

The shrunk-on discs are made of NiCrMoV alloy steel. There are ten (for LPA and LPB, Unit 1) and six (for LPC (Unit 1) and Unit 2 LPs) discs shrunk on the shaft with five per flow (for LPA and LPB, Unit 1) and three per flow for LPC (Unit 1) and Unit 2 LPs. These discs experience different degrees of stress when in operation. For LPA and LPB (Unit 1), Disc No. 2, starting from the transverse centerline, experiences the highest stress, while Disc No. 5 experiences the lowest. For LPC (Unit 1) and Unit 2 LPs, Disc No. 3 starting from the transverse centerline, experiences the highest stress, while Disc No. 1 experiences the lowest.

The minimum specified mechanical properties for the cylinders, rotor, and discs are shown in Section 10.2.3.1.

3.5.1.3.3 Primary Safety-Related Equipment Installations and Structures

The primary safety-related equipment installations and structures at the plant are those whose loss could lead to conditions in excess of the guidelines specified in 10 CFR 100. Items in this category are those in which a single strike by a potential turbine missile could result in a loss of the capability to function in the manner needed to meet these guidelines. At the Watts Bar Nuclear Plant, these are the a) Reactor Building, b) main control room, c) spent fuel pool, d) main steam valve rooms, and e) ERCW electrical conduits from manhole Nos. 1-3.

Nuclear power plants should protect their essential SSCs against both high-trajectory and low-trajectory turbine missiles resulting from the failure of main turbine-generator sets. Plants can protect essential SSCs against turbine missiles by any one of a number of approaches; however, the preferred option of Regulatory Guide (RG) 1.115 is as follows:

> For favorably oriented turbines, appropriately placing and orienting the turbine units such that essential SSCs are located outside the low-trajectory missile strike zone defined in RG 1.115 and limiting the calculated turbine missile generation frequency for high-trajectory missiles to a value less than 10^{-4} per year.

Each Unit’s turbine-generator is located south of the Reactor Building with its shaft oriented north-south and is considered to be favorably oriented. The turbine placement and orientation are shown in Figure 3.5-4. Favorably oriented turbine generators are located such that the containment and all, or almost all, safety-related structures, systems and components (SSCs) outside containment are excluded form the low-trajectory hazard zone described in RG 1.115. With the exception of the ERCW conduit, all safety-related structures, systems and components are located outside the low-trajectory missile strike zone. For the ERCW conduit target, the strike probability due to a low-trajectory turbine missile is considered to be zero since the rotation of the turbine will preclude a tangential missile from directly impacting the ERCW conduit as the turbine pedestal and Turbine Building structure provide barriers to the trajectory. Thus, postulated low-trajectory missiles cannot directly strike safety-related SSCs and are not considered a threat.

In addition, the safety-related SSCs are protected from high-trajectory turbine missiles by controlling the turbine missile generation frequency (P1) to be less than 10^{-4} per year (see Section 3.5.1.3.5).

3.5.1.3.4 Turbine Missile Protection Criterion

The turbine missile protection criterion utilized in the design of the Watts Bar Nuclear Plant was that the probability of unacceptable damage (P4) should not be significant. In this instance, an event having a probability of causing unacceptable damage on the order of about 10^{-7} per year per reactor unit at the plant is not considered significant. Therefore, for the two-unit Watts Bar Nuclear Plant, an event having a probability of occurrence less than 2×10^{-7} will fulfill this criterion.

3.5.1.3.5 Turbine Missile Hazard Evaluation

The probability of unacceptable damage resulting from turbine missiles, P4, is expressed as the product of the following three items:

1. the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing, P1,
2. the probability of ejected missiles perforating intervening barriers and striking essential SSCs, P2, and
3. the probability of essential SSCs that are struck failing to perform their safety functions, P3.

Stated in mathematical terms, P4 = P1 × P2 × P3. As noted in Section 3.5.1.3.4, P4 is limited to less than 10^{-7} per year, which the NRC staff considers to be an acceptable risk rate for the loss of an essential SSC from a single event.

In the past [11], analyses assumed the probability of turbine missile generation (P1) to be approximately 10^{-4} per turbine year for a favorably oriented unit, based on the historical failure rate. The strike probability (P2) was estimated on the basis of postulated missile sizes, shapes and energies and on available plant specific information such as turbine placement and orientation, number and type of intervening barriers, target geometry, and potential missile trajectories. The damage probability (P3) was generally assumed to be 1.0. The overall probability of unacceptable damage to safety-related systems (P4), which the sum over all targets of the product of these probabilities, was then evaluated for compliance with the NRC safety objective. This logic placed the regulatory emphasis on the strike probability, that is, it necessitated that P2 be made less than or equal to 10^{-3}, and disregarded all the plant specific factors that determine the actual P1 and its unique time dependency.

Although the calculation of strike probability was not difficult in principle, for the most part being not more than a straightforward ballistics analysis, it presented a problem in practice. The problem stemmed from the fact that numerous modeling approximations and simplifying assumptions were required to make tractable the incorporation into acceptable models of available data on the (1) properties of missiles, (2) interactions of missiles with barriers and obstacles, (3) trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and (4) identification and location of safety-related targets. The particular approximations and assumptions made tended to have a significant effect on the resulting value of P2. Similarly, a reasonably accurate specification of the damage probability (P3) was not a simple matter because of the difficulty in defining the missile impact energy required to render given safety-related systems unavailable to perform their safety functions and the difficulty in postulating sequences of events that would follow a missile-producing turbine failure.

Because of the uncertainties associated with calculating P2 and P3, the staff concludes that such analyses are "order of magnitude" calculations only. On the basis of simple estimates for a variety of plant layouts, the strike and damage probability product can be reasonably assumed to fall in a range that depends on the gross features of turbine generator orientation. For favorably oriented turbine generators, such as Watts Bar, the product of P2 and P3 tends to be in the range of 10^{-4} to 10^{-3} per year. For these reasons, in the evaluation of P4, the probability of unacceptable damage to safety-related systems from potential turbine missile, the staff is giving credit for the product of the strike and damage probabilities of 10^{-3} (P2xP3) for a favorably oriented turbine and is discouraging the elaborate calculation of these values. For these reasons, strike and damage calculations were not performed for Watts Bar instead the missile analysis [15] was updated for the new upgraded 13.9m^2 rotors as documented in the current missile reports [13].

Therefore, the NRC staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability (P2xP3) to the missile generation probability (P1) and, in the process, has attempted to integrate the various aspects of the issue into a single, coherent evaluation. The staff believes that maintaining an initial small value of P1 through turbine

testing and inspection is a reliable means of ensuring that the objectives precluding turbine missiles and unacceptable damage to safety-related structures, systems, and components can be met. It simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety is maintained.

Unit 1
A missile probability analysis was performed for the Unit 1 BB281-13.9m² damped blade rotor with advanced disc design shrunk-on discs. Based on conservative assumptions, the probability of turbine missile generation for unit speeds up to 120% of rated speed is 5.54×10^{-5}. It is observed that LPC, being a Siemens 13.9m² advanced disc rotor design, complies with the NRC limits up to 100,000 operating hours between successive disc inspections provided that no cracks are detected in those inspections. LPA and LPB rotors, being of Westinghouse design, remain unchanged and their recommended inspection interval of 60 months between LP disc inspections remains conservatively unchanged.

As documented in WCAP-16501-P and LTR-RAM-I-07-039 [21, 22], Unit 1 has elected to perform turbine valve test intervals at a frequency of every 6 months. This results in a probability of overspeed of 1.93×10^{-7} per year [13a].

Unit 2
A missile probability analysis was performed for the Watts Bar 2 BB281-13.9m² free standing rotors with advanced disc design shrunk-on discs. Based on conservative assumptions, the probability of an external missile for speeds up to 120% of rated speed is 2.13×10^{-6} for a disc inspection interval of 100,000 operating hours.

As documented in WCAP-16501-P and LTR-RAM-I-10-040 [21, 22], Unit 2 has elected to perform turbine valve test intervals at a frequency of every 6 months. This results in a probability of overspeed of 1.39×10^{-7} per year [13b].

As summarized in the missile reports [13], these probabilities (P1) for both Units 1 and 2 are well below the NRC limit of 10^{-4} per year for a favorably oriented unit. Therefore, it is concluded that the safety-related SSCs are protected from high-trajectory turbine missiles. The missile analysis methodology used was submitted to the NRC and approved in March 2004 [12].

Based on the calculated probabilities of turbine missile generation ($P1 < 10^{-4}$) and the NRC approved value of the strike and damage probability ($P2 \times P3 = 10^{-3}$) for favorably oriented turbines, the probability of unacceptable damage resulting from turbine missiles, P4, is less than 10^{-7} per year [15]. Therefore, for the two-unit plant, an event having a probability of occurrence less than 2×10^{-7} fulfills the criterion.

3.5.1.3.6 Turbine Missile Selection

Analyses described above indicate that the hazard from turbine missiles at the Watts Bar Nuclear Plant is not significant. In addition, the plant uses turbines designed, manufactured, installed, and operated in accordance with standards that minimize the possibility of an accident that may produce dangerous missiles. The unit has its essential, safety-related equipment installations and structures positioned to minimize the strike probability on these items.

Such findings indicate that the potential for turbine generated missiles at the Watts Bar Nuclear Plant is credible, but not significant. Therefore, turbine missile hazards need not be considered in the design of the Watts Bar Plant.

3.5.1.4 Missiles Generated By Natural Phenomena

Category I structures at Watts Bar Nuclear Plant are designed for tornado- generated missiles based on the following:

1. Spectrum A (see Table 3.5-7) was used in the design of:

    a. Manholes and protective slabs over manholes for Class 1E electric systems.

    b. Protection for Class 1E conduit duct runs.

    c. Hatch assemblies for personnel access openings to pipe tunnels A & B.

    d. Slabs supporting the ERCW piping at the Intake Pumping Station.

    e. Pipe encasement at the Diesel Generator Building.

    f. Refueling Water Storage Tank foundation.

    g. Roofs and walls of Category I structures except as noted for Spectrums B, C & D.

    h. Protection for diesel generator exhaust stacks.

    i. Protection for fuel oil storage tank vent lines in Diesel Generator Building.

2. Spectrum B (see Table 3.5-8) was used in the design of the equipment doors and bulkheads on the Diesel Generator Building.

3. Spectrum C (see Table 3.5-9) was used in the design of the structures not covered by Items 1, 2 or 4 (Intake pumping station structural steel roof, ERCW standpipe encasement, ERCW discharge overflow structure, and ERCW valve covers).

4. Spectrum D (See Table 3.5-17) was used in the design of the Additional Diesel Generator Building and any additional Category I Structures after July 1979.

3.5.1.5 Missiles Generated by Events Near the Site.

There are no postulated accidental explosions in the vicinity of the site (See Section 2.2.3). The only significant nearby industrial activity is the Watts Bar Steam Plant. Turbine missiles from this plant are treated in Section 3.5.1.3. No other missiles are considered significant. Therefore, the Watts Bar Nuclear Plant need not be designed for protection against missiles generated by explosions of trucks, trains, ships, barges, industrial facilities, hydrogen storage tanks, pipelines and military facilities.

3.5.1.6 Aircraft Hazards

There is one federal airway passing within two miles of the nuclear facility. The probability per year of an aircraft crashing into the plant (P_{FA})[17] is estimated in the following manner:

$$P_{FA} = C \times N \times A/w$$

where:

C = inflight crash rate per mile for aircraft using airway.

w = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles.

N = number of flights per year along the airway, and

A = effective area of plant in square miles.

For the Watts Bar site, $C = 4.0 \times 10^{-10}$, w = 8 miles, N = 2 x 2200/year, A = 0.01 mi^2. The value for N, per Section 2.2.2.5, was doubled to account for increases in aircraft traffic. Therefore, $P_{FA} = 2.2 \times 10^{-9}$/year. This conservative upper bound probability is less than the Standard Review Plan range of "1×10^{-7}" and indicates that further consideration of hazard from this federal airway is unnecessary.

There are no airports located within five miles of the site. There are no airports with projected operations greater than 500 d^2 movements per year located within 10 miles of the site or greater than 1,000 d^2 outside 10 miles, where d is the distance in miles from the site.

There are no military installations or any airspace use that might present a hazard to the site. Therefore, aircraft hazards need not be considered in the design of the Watts Bar Nuclear Plant.

3.5.2 Systems to Be Protected

Systems whose failure could lead to offsite radiological consequences or which are required for reactor shutdown and cooldown under normal and/or design basis conditions are listed in Table 3.2-1.

These include 1) the fluid handling equipment in ANS Safety Classes 1, 2a, 2b or 3 listed by system in Tables 3.2-2a and 3.2-2b, 2) Class 1E electrical systems and components listed in Table 3.2-3, and 3) the Heating, Ventilating, and Air Conditioning components in TVA Classes B, M, Q, and S as described in Table 3.2-2a. A more detailed discussion of required equipment and its safety-related implications may be found in the UFSAR sections covering individual system.

It is important to note that all of the equipment referenced above is required for some safety-related function but not all at once. The list of required equipment for a particular missile event depends on the nature of the missile (whether it is associated with or can cause a LOCA, whether or not the missile is being generated from a safety-related piece of equipment, etc). Hence, much of the missile damage consideration outside containment can be reduced to looking at those systems and components required for reactor shutdown, coolant system makeup, and for decay heat removal to the ultimate heat sink.

Table 3.5-14 lists outdoor features, including air intakes and exhausts, which may be required to perform a safety-related function coincident with or following the occurrence of a tornado.

For the Watts Bar Nuclear Plant the layout of structures and equipment is such that there are no systems or components which rely upon redundancy and separation alone for protection against externally generated missiles. There are however certain systems and components which, due to their location and/or separation of trains, are inherently protected against specific types of missiles. This protection by separation and location concept has been addressed where applicable in the missile analyses of Section 3.5.1.

3.5.3 Barrier Design Procedures

Historical Information - To arrive at a formula to use in computing penetration into concrete walls, a comparison was made of formulas listed in ORNL-NSIC-22, "Missile Generation and Protection in Lightwater-Cooled Power Reactor Plants." Four equations were studied in ORNL-NSIC-22 in connection with penetration in concrete. Two of these, the Army Corps of Engineers formula and the National Defense Research Committee formula, do not apply for impact velocities under 500 ft/sec and thus are not applicable here (velocity of 300 mph = 440 ft/sec). The remaining two equations are the modified Petry formula and the Ballistic Research Laboratory formula. These two formulas were compared for a 6-inch diameter missile of 100 pounds and a 16-inch diameter missile of 2,500 pounds with velocities in the range from 0 to 500 ft/sec. As seen in Figures 3.5-6 and 3.5-7, the Petry formula is the most conservative for velocities greater than 150 to 200 mph.

The following describes the barrier procedures utilized for concrete barriers. The depth to which a missile penetrated a concrete wall was estimated by use of the modified Petry formula. [6]

$$D' = 12\, KAV' \left[1 + e^{-4(a - 2)}\right]$$

where D' = depth of penetration

V = impact velocity

K = A material constant

A = weight of missile/ impact area of missile

$$V' = \log_{10}\left[1 + \frac{V^2}{215{,}000}\right]$$

$$a = \frac{T}{12KAV'}$$

T = wall thickness

The results are given in Figures 3.5-8 and 3.5-9. According to C. V. Moore,[7] spalling on the inside face of a wall does not occur for penetrations less than two-thirds the wall thickness.

Conservatism was assured by assuming nondeformable missiles in the penetration analysis using the modified Petry formula.

None of the postulated missiles described in Section 3.5.1, internal or external, will impair the capability of the engineered safety features to shut down the reactor or to maintain the reactor in a safe shutdown mode indefinitely. For portions of the engineered safety features located within the containment structure, protection against missiles generated inside containment is accomplished with the basic approach of assuring design adequacy against generation of credible missiles rather than to allow missile formation and try to contain the subsequent effects. Further, valves are oriented so that the trajectory of missiles will not likely pass through openings in the crane wall and the valve bonnets and stems will not penetrate the containment shell should they strike it. For these same engineered safety features, protection against tornado-generated missiles is provided by the Shield Building. If one of the pressurizer heaters in the bottom of the pressurizer should become loose and become a jet-propelled missile, it would move downward and could strike the pressurizer surge line beneath the pressurizer. The line will not be perforated and will not jeopardize the capability to bring the nuclear facility to a safe shutdown.

For those portions of the engineered safety features located outside the Shield Building and required for shutdown of the reactor and/or indefinite maintenance of the reactor in the safe shutdown mode, protection is provided against tornado-generated missiles. Protection is provided by locating these features within structures which have been designed to withstand damage by the spectrum of credible tornado-generated missiles.

The postulated missiles inside the containment as defined in Tables 3.5-1, 3.5-2, 3.5-3, and 3.5-4 have been investigated to determine their penetration characteristics. Penetration depths, or minimum thickness to just perforate, have been calculated based upon three commonly used equations. They are:

1. The Stanford Equation.

2. The Ballistics Research Laboratory Equation.

3. The Recht and Ipson Equation.

The minimum thicknesses to just perforate a plate having the characteristics of SA-516-GR70 carbon steel represent the largest values obtained from the three above-mentioned equations. The worst case involves a penetration depth that is 45% of the actual containment thickness. Based upon the analysis, it is reasoned that none of the postulated missiles pose a threat to the integrity of the containment.

Tornado missile impact loads, where required, were calculated based upon several applicable techniques. Impact loads for all missiles of Spectrum A (Table 3.5-7) except missile A3, the 4,000-pound automobile at 50 miles per hour, were determined by the relationships presented in Reference [9].

Missile A3 loads were based on actual test results and analysis technique. Time histories of decelerations were obtained from the National Highway Safety Bureau for automobile crash tests early in a crash safety program. Their time histories were converted to shock spectra by the usual methods and dynamic load factors were plotted against period of structures or elements of structures. An envelope was then constructed enveloping all spectra.

To determine the automobile impact load on a structure or element of a structure, the natural period of the item was determined and the appropriate loading obtained from the shock spectra. This technique yields the maximum load irrespective of time.

Values obtained from this technique have been corroborated with subsequent reports by the National Highway Safety Bureau. In Reference [5], time histories of forces are presented for several automobile crash tests which are closely confirmatory. The impact loads obtained by the previously described methods were then applied to the structures and the structures were analyzed for the effect of the loads by conventional analytical methods. Impact loads from the missiles of Spectrum C (Table 3.5-9) were calculated using the procedures of Reference [8]. See Section 3.5.1.4 for a discussion of structures designed for Spectrum C.

The structural steel grillage roof system for the Intake Pumping Station was designed for impact from the tornado missiles listed in FSAR Table 3.5-9. The EPRI testing results[18] were used to determine the possible maximum impact force for the steel grillage roof system. The force-time history of the end impact loading was further refined to account for the stiffness of the missile and the target. The impulse-momentum principle[19, 20] was used to analyze the midspan and end impacts. For the midspan impact, the response of the elastic-plastic single-degree system was considered along with a maximum ductility ration of $\mu = 20$ for bending deformation[8].

Tornado missile protection for all safety-related buried piping is provided by one of the four protective schemes described below.

1. 10 feet of compacted fine-grained soil.

2. 7 feet of compacted crushed stone.

3. 18 inches of conventional unreinforced concrete.

4. 18 inches of roller-compacted unreinforced concrete.

In each scheme, a 12-inch cushion of either compacted sand or fine-grained earthfill is required over the top of the pipe.

The acceptability of each scheme has been verified by a full-scale test program[16] in which missiles from the NRC spectrum were dropped from a helicopter into test pits of crushed stone or earthfill and onto concrete slabs. The missiles used in the testing were:

1. a 1,500-pound utility pole,

2. a 12-inch diameter schedule 40 steel pipe,

3. a 1-inch diameter steel rod,

4. a 3-inch diameter schedule 40 steel pipe, and

5. a 6-inch diameter schedule 40 steel pipe.

Of these missiles, the 12-inch pipe and utility pole caused the greatest penetration depths. Impact velocities of 200-215 ft/s were achieved for both the utility pole and 12-inch pipe which equals or exceeds the design velocities for those missiles as listed in Tables 3.5-7 and 3.5-9. The protective thicknesses listed above are based on the maximum thicknesses observed in the test program and are, therefore, conservatively chosen.

It is concluded that the missile protection criteria to which the plant has been analyzed and protected against comply with Criteria 2 and 4 of 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants.

3.5.3.1 Additional Diesel Generator Building (And Other Category I Structures Added After July 1979)

The openings in the walls and roof for access, ventilation, air intakes, and exhaust discharge, are designed to withstand the effects from the tornado missiles listed in Spectrum D of Table 3.5-17.

The 480V auxiliary board room (ABR) ventilation air intake vent is the only primary safety-related equipment located outside the Additional Diesel Generator Building[1] not protected against tornado missiles. The roof opening for the ABR vent is protected against tornado missile entry by a missile shield installed inside the roof structure. In-lieu of protecting the ABR vent from tornado missiles, operator actions are specified in the event of a tornado warning to restore ventilation cooling to the ABR. Overall structural response evaluation of concrete barriers to tornado missile impact was performed using the general requirements of Appendix C, ACI 349-76, "Code Requirements For Nuclear Safety-Related Concrete Structures." Minimum concrete thickness required to resist penetration, perforation or backface scabbing from these tornado missiles are given in Table 3.5-18.

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2. Deleted

3. Deleted

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7. Moore, C. V., 'The Design and Barricades for Hazardous Pressure Systems,' Nuclear Engineering and Design, Vol. 5, 1967, pp. 85-86. (Applicable to Unit 1 and Unit 2)

8. 'Design of Structures for Missile Impact', BC-TOP-9A, Revision 2 Topical Report, Bechtel Power Corporation, San Francisco, California, September 1974.

9. Westinghouse Systems Standard Design Criteria, "Protection Criteria Against Dynamic Effects Resulting from Pipe Rupture" (Identification No. 1.19).

10. Deleted

11. Letter from Mr. Herbert N. Berkow, NRC Director, to Mr. Stan Dembkoski, SWPC Director, dated March 30, 2004, Subject: Final Safety Evaluation Regarding Referencing the Siemens Technical Report No. CT-27332, Revision 2, "Missile Probability Analysis for the Siemens 13.9m2 Retrofit Design of Low-Pressure Turbines by Siemens AG", TAC No. MB7964.

12. Topical Report TP-04124-NP-A: "Missile Probability Analysis for the Siemens 13.9m2 Retrofit Design of Low-Pressure Turbine by Siemens AG", Submitted to the Nuclear Regulatory Commission, For Public Record, June 7, 2004, Siemens Westinghouse Power Corporation.

    Note: Topical Report TP-04108-P-A: "Missile Probability Analysis for the Siemens 13.9m² Retrofit Design of Low-Pressure Turbine by Siemens AG", Submitted to the Nuclear Regulatory Commission, Not For Public Record, June 7, 2004, Siemens Westinghouse Power Corporation. This is the proprietary version of the same report referenced above.

13a. CT-27547 Rev. 1, Missile Probability Analysis Report, TVA Watts Bar 1, BB281-13.9m² Damped Blades, October 14, 2016, Siemens Energy, Inc. (Applicable to Unit 1)

13b. CT-27467, Missile Report, TVA Watts Bar 2, BB281-13.9m2, October 6, 2009, Siemens Energy, Inc. (Applicable to Unit 2 only)

14. Deleted

15. Calculation TI-521, "Probability for Turbine Missile Strike Damage at Watts Bar Nuclear Plant."

16. CEB Report CEB-81-20, "Testing of Protective Cover for Essential Buried Structures."

17. U.S. Nuclear Regulatory Commission NUREG-0800, "Standard Review Plan," Section 3.5.1.6.III.2.

18. Stephenson, A. E., and G. E. Sliter, “Full-Scale Tornado-Missile Impact Tests,” Electric Power Research Institute Final Report, EPRI NP-440, July 1977.

19. Civil Design Guide DG-C1.5.7, Rev. 1, “Design of Slabs for Missile Impact.”

20. Topical Report TVA-TR78-4, "Design of Structures to Resist Missile Impact," TVA Office of Engineering Design and Construction (B41 851204 002).

21. WCAP-16501-P, Revision 0, "Extension of Turbine Valve Test Frequency Up to 6 Months for BB-296 Siemens Power Generation (Westinghouse) Turbines with Steam Chests", February 2006, Westinghouse Electric Company, LLC, Westinghouse Proprietary Class 2.

22a. LTR-RAM-I-07-039, "Review of Watts Bar Unit 1 for Applicability of WCAP-16501-P", September 2007 (Applicable to Unit 1 only, attached to WAT-D-11600)

22b. LTR-RAM-I-10-040, "Review of Watts Bar Unit 2 for Applicability of WCAP-16501-P", September 2010 (Applicable to Unit 2 only, attached to 25402-011-V1A-MG00-03478-001)

23. EC-02262, "Missile Generation Risk Assessment for Original and Retrofit Nuclear HP Rotors", December 17, 2002, Siemens Westinghouse Power Corporation.

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TABLE 3.5-1

SUMMARY OF POSTULATED CRDM MISSILE ANALYSIS

	POSTULATED MISSILES	WEIGHT[5] (LB)	THRUST AREA (IN^2)	IMPACT AREA (IN^2)	IMPACT[1] VELOCITY[4] (Ft/SEC)	KINETIC[1] ENERGY (Ft-Lbs)
1.	Mechanism Housing Plug	11	5.94[2]	7.07	450	34,600
2.	Drive Shaft Assembly	136	2.41	2.41	179	69,800
3.	Mechanism Housing[3] Plug And Drive Shaft Impacting On Same Missile Shield Spot	-	-	-	-	-
4.	Drive Shaft Latched To Mechanism	1500	2.41	11.04	34	26,400

NOTES:

(1) Velocities and kinetic energies for a distance of 4.389(ft) between top of CRDM Housing and Missile Shield.

(2) Flow discharge area (equal to thrust area-no expansion of jet assumed).

(3) Assume drive shaft further pushes housing plug into shield.

(4) For the calculational methods, see Appendix 3.5A.

(5) Dry weight

TABLE 3.5-2

TYPICAL POSTULATED VALVE MISSILE CHARACTERISTICS

Missile Description	Weight (lb)	Flow Discharge Area (in^2)	Thrust Area (in^2)	Impact Area (in^2)	Weight to Impact Area Ratio (psi)	Terminal[(1)] velocity (fps)
Safety Relief Valve Bonnet (3" x 6" or 6" x 6")	350	2.86	80	24	14.5	110
3 Inch Motor Operated Isolation Valve Bonnet (plus motor and stem) (3")	400	5.5	113	28	14.1	135
2 Inch Air Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
3 Inch Air Operated Spray Valve Bonnet (plus stem)	120	5.5	50	50	2.4	190
4 Inch Air Operated Spray Valve	200	9.3	50	50	4	190

NOTES:

1. For the calculational methods, see Appendix 3.5A

WBN
TABLE 3.5-3

POSTULATED PIPING TEMPERATURE ELEMENT ASSEMBLY MISSILE CHARACTERISTICS

1. For a tear around the weld between the boss and the pipe:

Characteristics	"without well"	"with well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	7.1 in^2	9.6 in^2
Missile Weight	11.0 lb	15.2 lb
Area of Impact	3.14 in^2	3.14 in^2
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	3.5 psi	4.84 psi
Velocity [(1)]	20 ft/sec	120 ft/sec

2. For a tear at the junction between the temperature element assembly and the boss for the "without well" element and at the junction between the boss and the well for the "with well" element.

Characteristics	"without well"	"with well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	3.14 in^2	3.14 in^2
Missile Weight	4.0 lb	6.1 lb
Area of Impact	3.14 in^2	3.14 in^2
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	1.27 psi	1.94 psi
Velocity	75 ft/sec	120 ft/sec

NOTES:
1. For the calculational methods, see Appendix 3.5A

TABLE 3.5-4

CHARACTERISTICS OF OTHER MISSILES
POSTULATED WITHIN REACTOR CONTAINMENT

	Reactor Coolant Pump Temperature Element	Instrument Well of Pressurizer	Pressurizer Heaters
Weight	0.25 lb	5.5 lb	15 lb
Discharge Area	0.50 in^2	0.442 in^2	0.80 in^2
Thrust Area	0.50 in^2	1.35 in^2	2.4 in^2
Impact Area	0.50 in^2	1.35 in^2	2.4 in^2
Missile Weight / Impact Area	0.5 psi	4.1 psi	6.25 psi
Velocity(1)	260 ft/sec	100 ft/sec	55 ft/sec

NOTES:

1. For the calculational procedures, see Appendix 3.5A

TABLE 3.5-5

DELETED

(See Section 10.2.3.1, historical information)

TABLE 3.5-6

DELETED

TABLE 3.5-7

TORNADO MISSILE SPECTRUM A FOR CATEGORY I STRUCTURES[1]

Missile[2]	Description	Design Velocity Exterior Wall ft/s (mi/h)	Design Velocity Roof System ft/s (mi/h)
A1	Wood plank, 2 in. x 4 in. x 12-ft long, weight 27 lbs	440 (300)	--
A2	Cross tie, 7 in. x 9 in. x 8.5 ft long, weight 186 lbs	440 (300)	--
A3	Automobile, weight 4000 lbs, up to 25 ft above grade at structure	73 (50)	--
A4	Steel pipe, 2-in. diameter, 7-ft long, weight 26 lbs	147 (100)	--
A5	Steel rod, 1-in. diameter x 3-ft long, weight 8 lbs	210 (143)	168 (115)
A6	Utility pole, 13.5 in. diameter x 35-ft long, weight 1490 lbs, up to 30 feet above grade	200 (136)	160 (109)

Notes:

(1) See Section 3.5.1.4.

(2) Missiles A1 through A4 were considered in original design. Missiles A5 and A6 were based on the structural adequacy of as-designed structures.

TABLE 3.5-8

TORNADO MISSILE SPECTRUM B
DIESEL GENERATOR BUILDING EQUIPMENT DOORS[1]

Missile	Description	Design Velocity ft/sec (mph)
B1	100-lb missile with 4-in. diameter for impact area	147 (100)
B2	10-ft length of 2 in. standard pipe impact endwise (weight = 36.5 lbs.)	147 (100)
B3	10-ft length of 1/2-in. standard pipe impacting endwise (weight 8.5 lbs.)	147 (100)
B4	Wood plank, 2 in. x 4 in. x 12 ft long, weight 27 lbs.	440 (300)
B5	Cross tie, 7 in. x 9 in. x 8.5 ft long, weight 186 lbs.	440 (300)
B6	Steel pipe, 2-in. diameter by 7 ft long, weight 26 lbs.	147 (100)

Missiles B1, B2, and B3 were considered in the design of the equipment doors. Additional protection is provided for missiles B4, B5, and B6.

Note:

1. See Section 3.5.1.4.

TABLE 3.5-9

TORNADO MISSILE SPECTRUM C
FOR CATEGORY I STRUCTURES[1]

		Design Velocity	
		Exterior Wall ft/s (mi/h)	Roof System ft/s (mi/h)
C1	Wood plank, 4 in. x 12 in. x 12 ft., weight 200 lbs.	368 (251)	294 (200)
C2	Steel pipe, 3-in. diameter, 10-ft long, weight 78 lbs	268 (183)	215 (147)
C3	Steel rod, 1-in. diameter, 3-ft long, weight 8 lbs	259 (177)	207 (141)
C4	Steel pipe, 6-in. diameter, 15-ft long, weight 285 lbs	230 (157)	184 (125)
C5	Steel pipe, 12-in. diameter, 15-ft long, weight 743 lbs	205 (140)	165 (112)
C6	Utility Pole, 13-1/2-in. diameter, 35-ft long, weight 1490 lbs.	241 (164)	205 (140)
C7	Automobile, frontal area 20 ft^2, 4000 lbs, up to 30 ft above grade	100 (70)	80 (56)

NOTE:

(1) See Section 3.5.1.4.

WBN-1

TABLE 3.5-10

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TABLE 3.5-11

DELETED

TABLE 3.5-12

DELETED

TABLE 3.5-13

DELETED

TABLE 3.5-14
(Sheet 1 of 2)

OUTDOOR SAFETY-RELATED FEATURES[1]
(Including Air Intakes and Exhausts)

	Feature	UFSAR Figure	Tornado Protection
1.	ERCW pumps, traveling screens, and ancillary features located on the deck of the intake pumping station.	3.8.4-50	Steel grillage roof system consisting of a series of wide flange beams (W21 x 49) spaced 9 inches on center and rotated 45 degrees. Walls of 15 inch reinforced concrete.
2.	ERCW Standpipes	2.5-225	Reinforced concrete enclosure (15-inch clear cover)
3.	ERCW Overflow Box	3.8.4-46a	Reinforced concrete cover (2-feet thickness)
4.	ERCW yard piping	9.2-40	Buried with one of the following minimum covers: 18 inches of concrete, 7 feet of crushed stone, or 11 feet of earthfill.
5.	RWST	3.8.4-35	3-1/2 feet high concrete wall with earth backfill surrounding tank to preserve necessary volume of water
6.	Pipe tunnels	3.8.4-35	Buried reinforced concrete cover (2-feet thickness)
7.	Diesel generator ventilation intakes and exhausts	3.8.4-26	Reinforced concrete canopy (1-foot thickness)
8.	Diesel generator engine combustion air exhausts	3.8.4-26	Reinforced concrete curb, 3 feet high and 18 inches thick around the exhaust
9.	Diesel generator electric board room air intakes (4 total)	3.8.4-26	Steel canopy with barrier protection (steel frame with 1 inch cover plate).
10.	480V transformer room, ventilation intakes and exhausts	1.2-1	Suspended steel grating (2-1/2 inch x 3/8 inch main bars and 1/2 in diameter connecting bars).
11.	Vital battery room exhausts	1.2-1	Same as for Item 10 except the fifth Vital Battery Room is same as Item 9.
12.	480V board room condensing unit air intake and exhaust (train A)	1.2-1	Reinforced concrete roof parapet (a minimum of 12 inches thick and 3 feet high)
13.	480V board room condensing unit air intake (train B)	1.2-1	Reinforced concrete canopy (18 inches thick)

NOTE:

(1) This tabulation consists of these outdoor features which are safety-related in the event of a tornado and as defined in Branch Technical Position ASB3-2 of the Standard Review Plan for Section 3.5.1-4 "Missiles Generated by Natural Phenomena."

TABLE 3.5-14
(Sheet 2 of 2)

OUTDOOR SAFETY-RELATED FEATURES
(Including Air Intakes and Exhausts)

Feature	UFSAR Figure	Tornado Protection
14. 480V board room condensing unit air exhaust (train B)	1.2-1	On roof of mechanical equipment rooms.
15. Class 1E duct banks and manholes	3.8.4-37	Buried with concrete protection (9 inch clear concrete cover for conduits; 12 inches total concrete thickness of manholes).
16. Fuel oil 7-day tank vent line (portion above roof)	3.8.4-26	Encased in reinforced concrete.
17. IPS El. 728' terrace roof openings which expose Cat. I electrical cable trays	3.8.4-50	Steel grating 4'-10" x 4' -10" Borden type C Style 18 size C A36, structural steel shroud (3/4" thickness), 18" thick precast concrete panels.
18. DGB fuel oil 1-day tank vent lines (portions protruding from exterior walls)	3.8.4-27	Steel plate shrouds (2" cover plate thickness)
19. Vacuum Relief Ducts in Aux. Bldg. Roof, Units 1&2	1.2-1	Steel plate canopy (1/2" thickness) over existing duct opening
20. Diesel Generator Bldg. roof access hatches (two)	3.8.4-26	Structural steel plate shrouds under DGB roof. Steel plate thickness varies from 1/4" to 1/2", as required
21. Emergency Pressurizer Fan opening on Cont. Bldg. roof, column lines C10-C11 & n-q	1.2-1	Steel plate canopy (1/2" min. thick.) over opening
22. Toilet and locker room exhaust fan opening on Cont. Bldg. roof, column lines C2-C3 & n-q	1.2-1	Steel plate canopy (1/2" min. thick.) over opening
23. Battery room exhaust vent in Cont. Bldg. roof, column lines C3-C4 & n-q	1.2-1	Steel plate canopy (1/2" min. thick.) over opening

TABLE 3.5-15
UNIT 1 ONLY

DELETED

TABLE 3.5-16
UNIT 1 ONLY

DELETED

TABLE 3.5-17

TORNADO MISSILE SPECTRUM D[1]

Missile Description	Weight (lb)	Cross Section	Length (ft)	Horizontal Velocity[2] (ft/sec)
Wooden Plank	115	4" x 12"	12	272
Steel Rod	9	1" dia	3	167
6" Schedule 40 Pipe	287	6" dia	15	171
12" Schedule 40 Pipe	743	12" dia	15	154
Utility Pole	1124	13-1/2" dia	35	180
Automobile	4000	6.5' x 4.3'	16.5	194

Notes:

(1) For Additional Diesel Generator Building and additional Category I structures after July 1979

(2) Vertical velocities of 70% of the postulated horizontal velocities are acceptable except for the 1 inch steel rod which shall have a vertical velocity equal to its horizontal velocity (167 ft/sec). These missiles are capable of striking in any horizontal or downward direction and at all elevations.

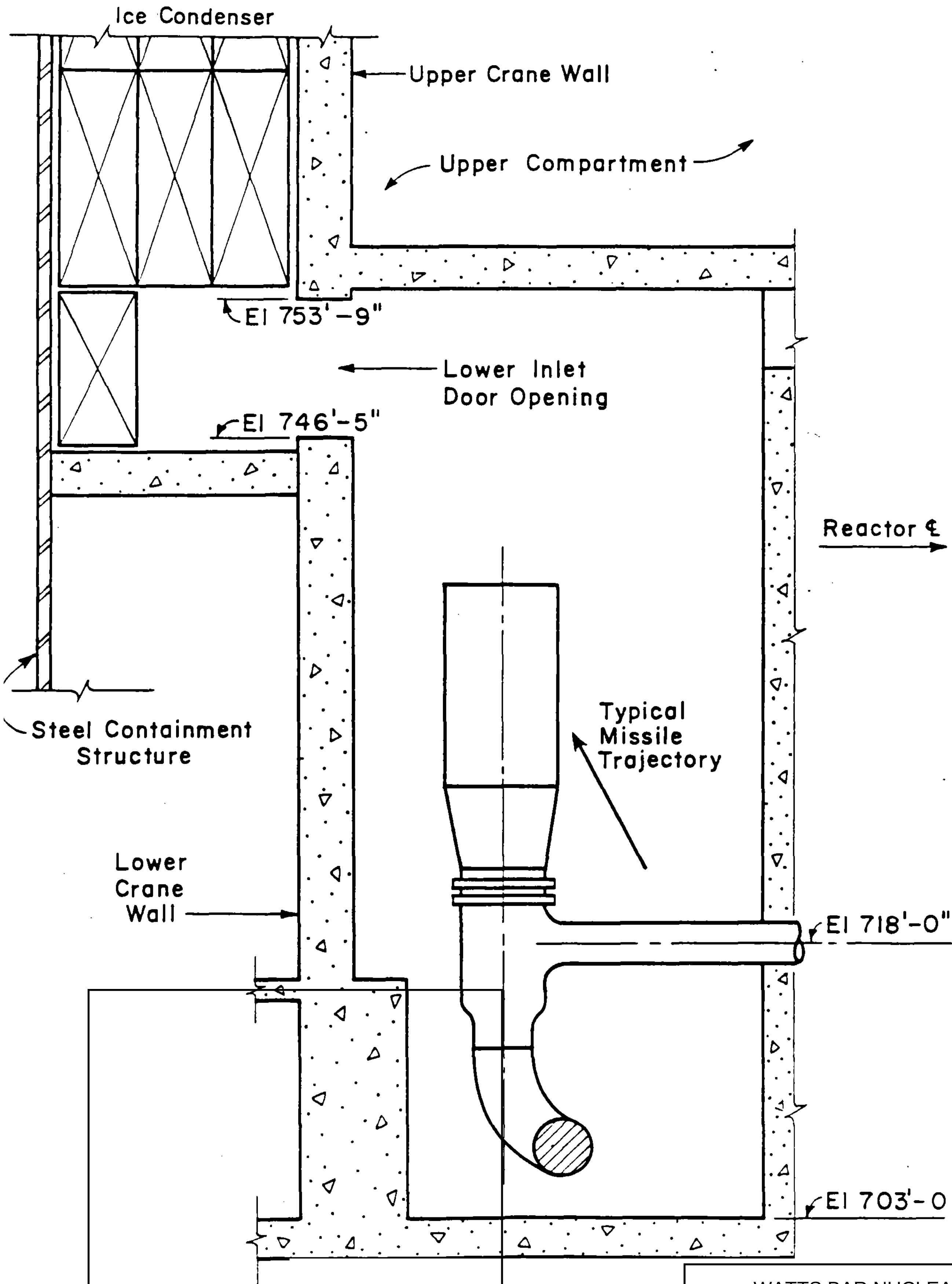
TABLE 3.5-18

Minimum Wall and Roof Thickness Requirements
To Resist the Effects of Tornado Missile Impact(1)

Tornado Intensity Region	28-Day Concrete Strength (PSI)	Wall Thickness (Inches)	Roof Thickness (Inches)
Region I	3000	23	18
	4000	20	16
	5000	18	14

NOTE:

(1) For the Additional Diesel Generator Building and additional Category I structures added after July 1979.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Ice Condenser Lower Inlet Door Opening, Typical Missile Trajectory Orientation

FIGURE 3.5-1

WBN-1

FIGURE 3.5-2

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WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Physical Dimensions of
Important Potential
Turbine Missiles

FIGURE 3.5-2

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.5-3

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Physical Dimensions of Important Potential Turbine Missiles FIGURE 3.5-3

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.5-4

FIGURE 3.5-5

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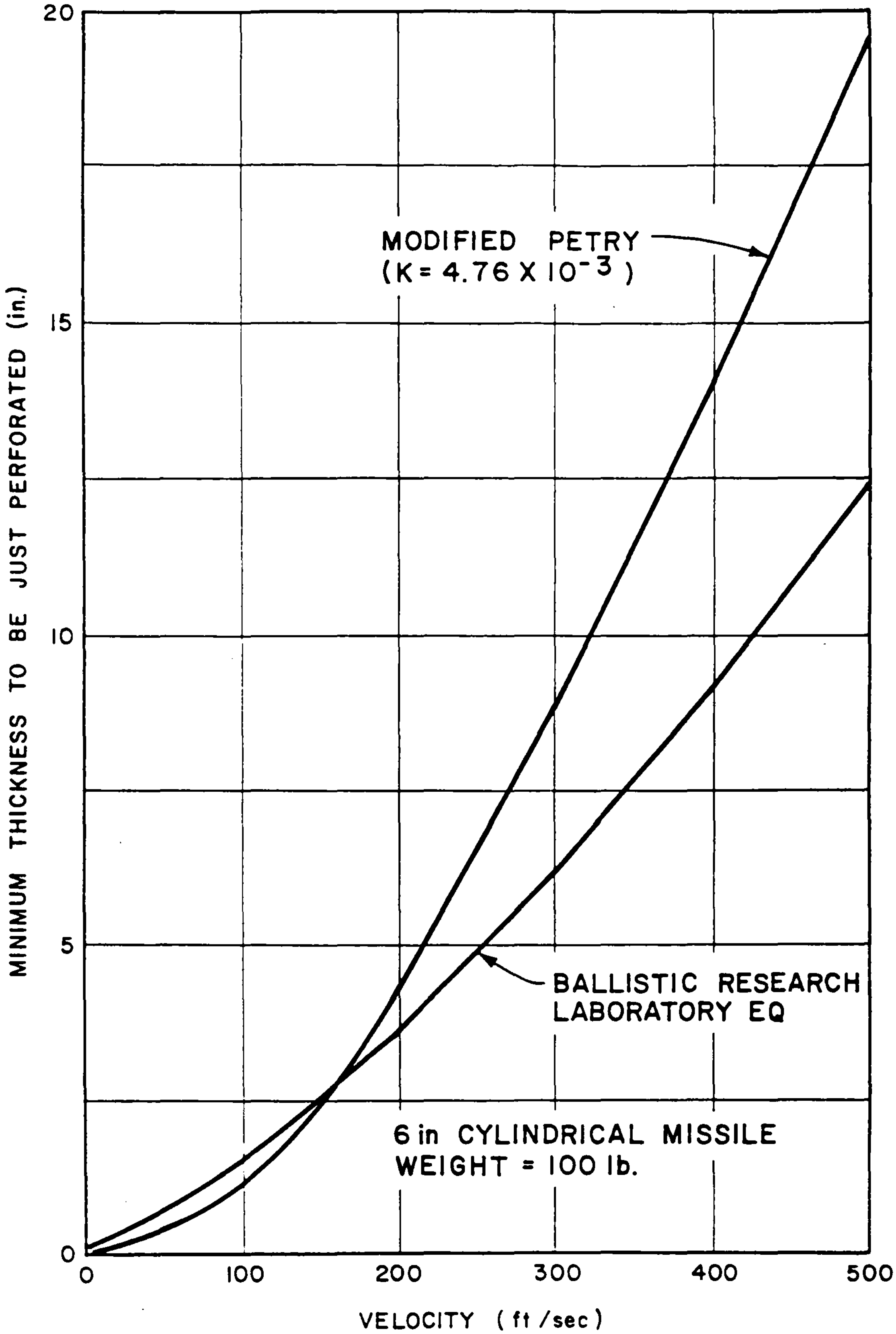
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Cross-Sectional Analysis of
Susceptibility of Critical
Components to Upward Turbine Missile
Trajectories

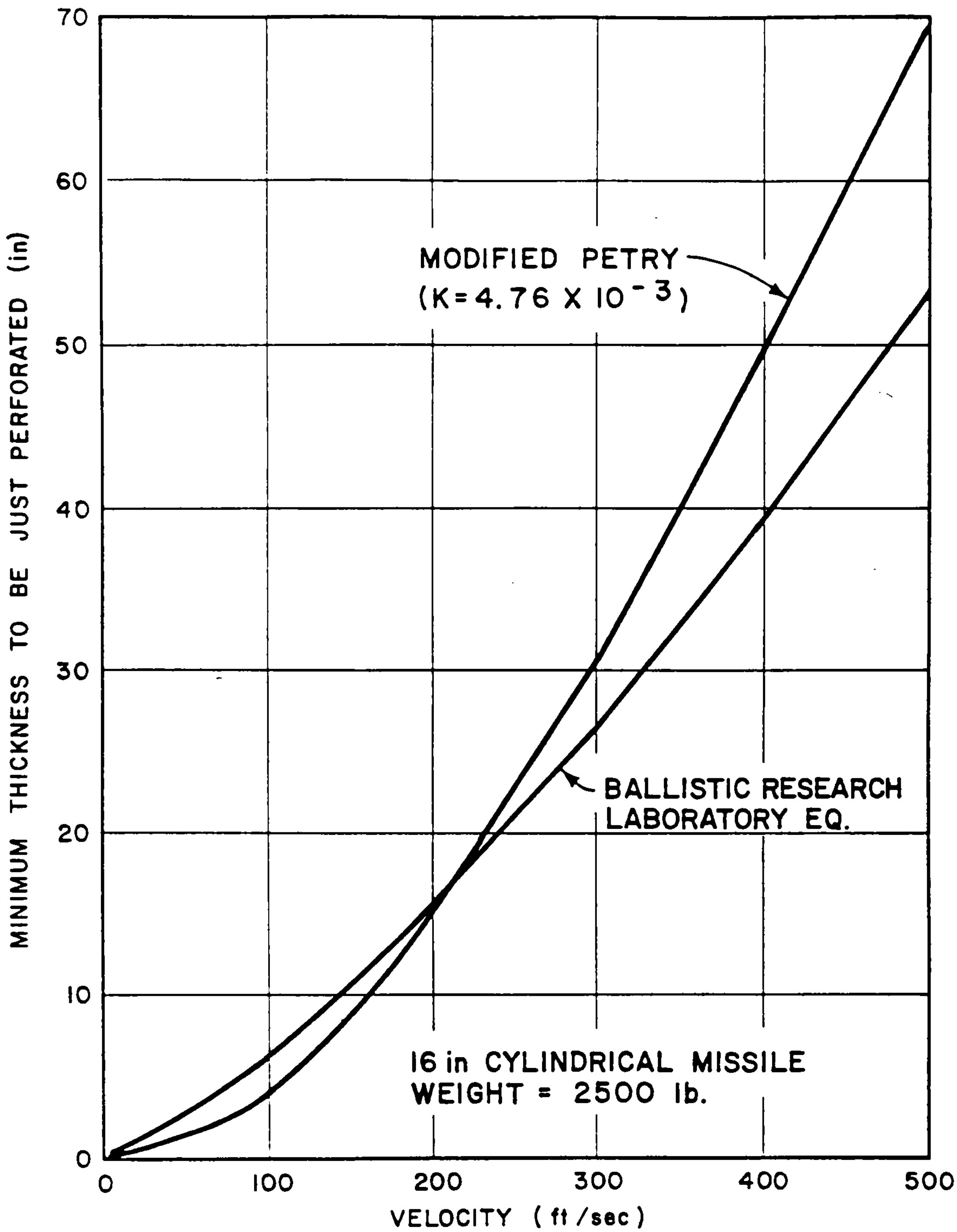
FIGURE 3.5-5



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Comparison of Missile Formulas

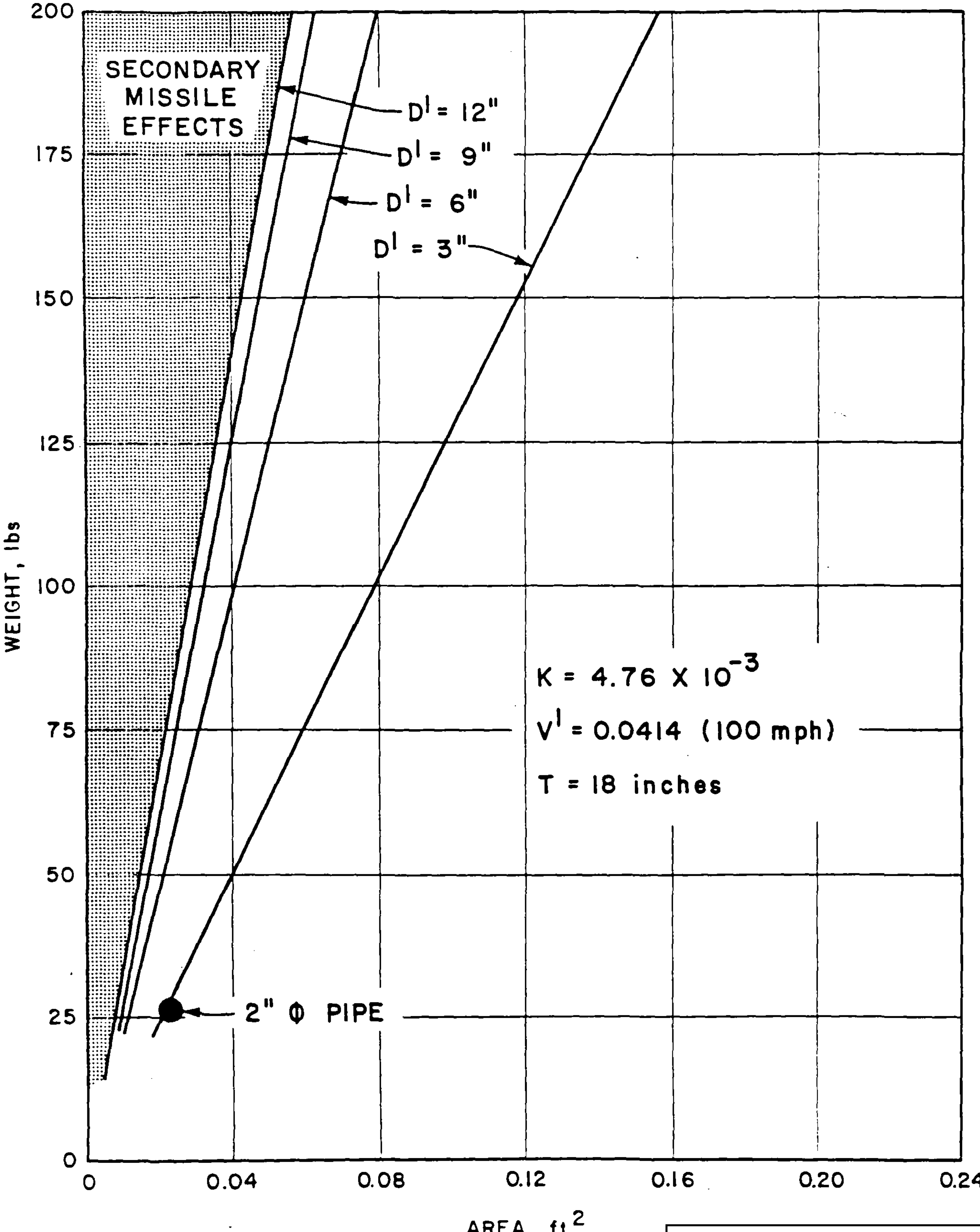
FIGURE 3.5-6



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Comparison of Missile Formulas

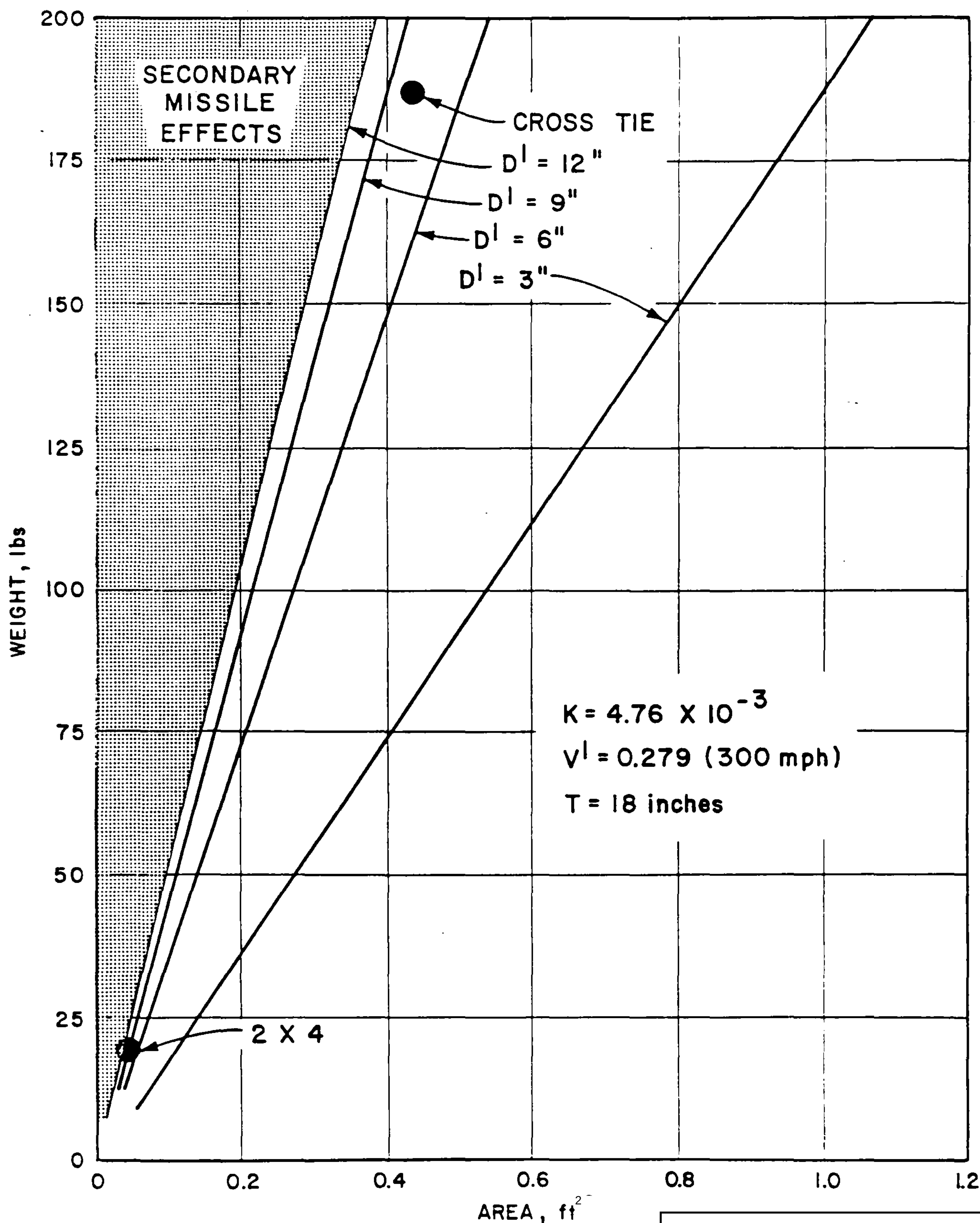
FIGURE 3.5-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Depth of Missile Penetration
for Tornado

FIGURE 3.5-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Depth of Missile Penetration
for Tornado

FIGURE 3.5-9

FIGURE 3.5-10

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WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LP Disc Missiles

FIGURE 3.5-10

FIGURE 3.5-11

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WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

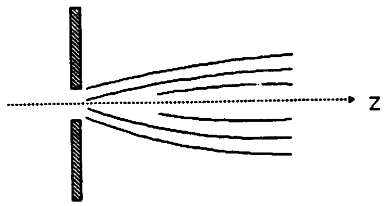
LP Cylinder & Blade Ring Fragments

FIGURE 3.5-11

APPENDIX 3.5A

ESTIMATES OF VELOCITIES OF JET PROPELLED MISSILES

A. Jet Stream Relations



For steady flow, assuming bulk properties across a cross section of a "free jet," the following conservation relations hold between the orifice and any downstream position.

Continuity in axial direction:

1) $W = G_o A_o = GA = \rho V_z A$

From conservation of axial momentum:

2) $$T_j = P_o A_o + \frac{\dot{W}}{g} V_o = PA + \frac{\dot{W}}{g} V_z = (P_o + \frac{G_o V_o}{g}) A_o$$

Where,

W = mass discharge rate

A = area of jet stream

V_z = axial fluid velocity

P = fluid pressure

G = mass flux rate

ρ = fluid density

T_j = total jet thrust

$$G_o = 0.61\sqrt{2\rho g(P_r - P_o)}$$

$$V_o = \sqrt{\frac{2g(P_r - P_o}{\rho}}$$

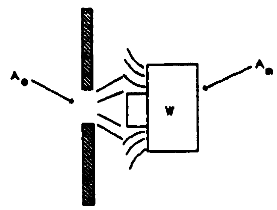
P_o = orifice pressure

P_r = reservoir pressure

Subscript o refers to orifice conditions.

These relations indicate that mass flow rate and axial jet force are constant at any downstream plane even though pressure, area, density, and fluid velocity change from one downstream position to another.

B. Missile Acceleration



Assuming that fluid impinging on missile imparts all of its axial momentum to the missile and splashes radially out of the stream, the force balance on the missile is:

3) $$\frac{W}{g}\frac{dV_m}{dt} = f(z)[P + \frac{G}{g}(V_z - V_m)]A2$$

Where,

f(z) = represents the fraction of the jet impinging

W = missile weight

V_m = missile velocity

For no expansion of jet, f(z) = 1, A = A_o ($A_m \geq A_o$ is assumed)

C. Effect on Jet Radial Expansion

For the general case, the missile would not receive the full thrust of the jet throughout its travel, and an estimate of f(z) with downstream distance is needed. For a first approximation, it is assumed that the jet expands with a constant half angle and that the thrust fraction, f(z), impinging on the missile is:

f(z) = 1 for $A_m \geq A(z)$

f(z) = $A_m/A(z)$ for $A_m \leq A(z)$

Where,

$A(z) = \pi(R_o + Z \tan \beta)^2$

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures, systems, and components important to plant safety be protected from the dynamic effects of a pipe rupture. This section of the UFSAR describes the design measures necessary to ensure compliance with this requirement. This section is subdivided into Part A and Part B. To be consistent with the standard format, all sections and subsection numbers are suffixed with either A or B.

Part A (3.6A) includes all piping systems inside and outside containment except the reactor coolant loop piping. The reactor coolant branch lines, however, are within the scope of this part. Also, jet impingement considerations of the reactor coolant loop on components other than those associated with the primary loop are within the scope of this report.

Part B (3.6B) includes the reactor coolant loop system except as stated in 3.6A.

Table 3.6-1
Unit 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
MAIN STEAM LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-2	1-MS-1	MS1-B0-1	32116	37800
		MS1-B0-2	29055	37800
		MS1-B0-3	16925	37800
		MS1-B0-4N	*	*
		MS1-B0-5N	*	*
		MS1-B0-6	30092	37800
3.6-3	1-MS-2	MS2-B0-1	45990	37800
		MS2-B0-2	29432	37800
		MS2-B0-3	26230	37800
		MS2-B0-4N	*	*
		MS2-B0-5N	*	*
		MS2-B0-6	25175	37800
3.6-4	1-MS-3	MS3-B0-1	29145	37800
		MS3-B0-2	26407	37800
		MS3-B0-3N	*	*
		MS3-B0-4N	*	*
		MS3-B0-5	27766	37800
		MS3-B0-6	26658	37800
3.6-5	1-MS-4	MS4-B0-1	31076	37800
		MS4-B0-2	30605	37800
		MS4-B0-3	14347	37800
		MS4-B0-4N	*	*
		MS4-B0-5N	*	*
		MS4-B0-6	25726	37800

Note: All breaks are circumferential ruptures.

* Branch connection to be supported in accordance with alternate analysis criteria. Stresses are not available.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL

WBN

Table 3.6-1
Unit 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
MAIN STEAM LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-2	1-MS-1	MS1-B0-1**	32116	37800
		MS1-B0-2**	29055	37800
		MS1-B0-3	16925	37800
		MS1-B0-4N	*	*
		MS1-B0-5N	*	*
		MS1-B0-6	30092	37800
3.6-3	1-MS-2	MS2-B0-1	45990	37800
		MS2-B0-2**	29432	37800
		MS2-B0-3**	26230	37800
		MS2-B0-4N	*	*
		MS2-B0-5N	*	*
		MS2-B0-6	25175	37800
3.6-4	1-MS-3	MS3-B0-1	29145	37800
		MS3-B0-2**	26407	37800
		MS3-B0-3N	*	*
		MS3-B0-4N	*	*
		MS3-B0-5	27766	37800
		MS3-B0-6**	26658	37800
3.6-5	1-MS-4	MS4-B0-1**	31076	37800
		MS4-B0-2**	30605	37800
		MS4-B0-3	14347	37800
		MS4-B0-4N	*	*
		MS4-B0-5N	*	*
		MS4-B0-6	25726	37800

Note: All breaks are circumferential ruptures.

* Branch connection stresses were not available.

** Breaks selected to satisfy the minimum of intermediate breaks are no longer required.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL

Table 3.6-2
UNIT 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
FEEDWATER LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-6	1-FW-1	FW1-B0-1	9651	32400
		FW1-B0-2	16919	32400
		FW1-B0-3N	*	*
		FW1-B0-4N	*	*
		FW1-B0-5N	*	*
		FW1-B0-6	24434	32400
		FW1-B0-7	23499	32400
		FW9-B0-1N	*	*
		FW9-B0-4N	*	*
3.6-7	1-FW-2	FW2-B0-1	24519	32400
		FW10-B0-1N	*	*
		FW2-B0-2	16306	32400
		FW2-B0-3N	*	*
		FW10-B0-4N	*	*
		FW2-B0-5N	*	*
		FW2-B0-6	24748	32400
		FW2-B0-7	23130	32400
3.6-8	1-FW-3	FW3-B0-1	22658	32400
		FW3-B0-2	15181	32400
		FW3-B0-3	13407	32400
		FW3-B0-4N	*	*
		FW11-B0-4N	*	*
		FW3-B0-6N	*	*
		FW11-B0-1N	*	*
		FW3-B0-7	19775	32400
3.9-9	1-FW-4	FW4-B0-1	9787	32400
		FW4-B0-2	12898	32400
		FW4-B0-3N	*	*
		FW4-B0-4N	*	*
		FW4-B0-5N	*	*
		FW4-B0-6	18915	32400
		FW4-B0-7	18203	32400
		FW12-B0-1N	*	*
		FW12-B0-4N	*	*

Note: All breaks are circumferential ruptures.

* Branch connection to be supported in accordance with alternate analysis criteria. Stresses are not available.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF VA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

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Table 3.6-2
UNIT 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
FEEDWATER LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-6	1-FW-1	FW1-B0-1	9651	32400
		FW1-B0-2**	16919	32400
		FW1-B0-3N	*	*
		FW1-B0-4N	*	*
		FW1-B0-5N	*	*
		FW1-B0-6**	24434	32400
		FW1-B0-7	23499	32400
		FW9-B0-1N	*	*
		FW9-B0-4N	*	*
3.6-7	1-FW-2	FW2-B0-1	24519	32400
		FW10-B0-1N	*	*
		FW2-B0-2**	16306	32400
		FW2-B0-3N	*	*
		FW10-B0-4N	*	*
		FW2-B0-5N	*	*
		FW2-B0-6	24748	32400
		FW2-B0-7	23130	32400
3.6-8	1-FW-3	FW3-B0-1	22658	32400
		FW3-B0-2**	15181	32400
		FW3-B0-3**	13407	32400
		FW3-B0-4N	*	*
		FW11-B0-4N	*	*
		FW3-B0-6N	*	*
		FW11-B0-1N	*	*
		FW3-B0-7	19775	32400
3.6-9	1-FW-4	FW4-B0-1	9787	32400
		FW4-B0-2**	12898	32400
		FW4-B0-3N	*	*
		FW4-B0-4N	*	*
		FW4-B0-5N	*	*
		FW4-B0-6	18915	32400
		FW4-B0-7	18203	32400
		FW12-B0-1N	*	*
		FW12-B0-4N	*	*

Note: All breaks are circumferential ruptures.

* Branch connection stresses were not available.

** Breaks selected to satisfy the minimum of intermediate breaks are no longer required.

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

WBN

TABLE 3.6-3
UNIT 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
AUXILIARY FEEDWATER SYSTEM STEAM SUPPLY LINE

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE 1)
3.6-10 (Unit 1)	1-AFD-8	AFD8-BO-1	18146	32400	C
	1-AFD-7	AFD7-BO-1	19515	32400	C
	1-AFD-7	AFD7-B1-2X	(Note 2)		C
	1-AFD-9	721	36751	32400	C,L
	1-AFD-9	719	35073	32400	C,L
	1-AFD-9	40	20305	32400	C
	1-AFD-9	1	6642	32400	C
	-	LMM	22438	32400	C
	-	L81 L82	23087	32400	C
	-		23000	32400	C

Notes: 1. C = Circumferential, L = Longitudinal Split
2. Not required to be postulated

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

TABLE 3.6-3
UNIT 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
AUXILIARY FEEDWATER SYSTEM STEAM SUPPLY LINE

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE 1)
3.6-10	1-AFD-8	AFD8-BO-1	18146	32400	C
(Unit 1)	1-AFD-7	AFD7-BO-1	19515	32400	C
	1-AFD-7	AFD7-B1-2X	(Note 2)		C
	1-AFD-9	721	36751	32400	C,L
	1-AFD-9	719	35073	32400	C,L
	1-AFD-9	40**	20305	32400	C
	1-AFD-9	1	6642	32400	C
	-	LMM	22438	32400	C
	-	L81**	23087	32400	C
	-	L82**	23000	32400	C

Notes: 1. C = Circumferential, L = Longitudinal Split
2. Not required to be postulated

** Breaks selected to satisfy the minimum number of intermediate breaks are no longer required.

Reflects Analysis Which Was Current At Time of Amendment 51 Submittal

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TABLE 3.6-3A

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
FOR AUXILIARY FEEDWATER SYSTEM STEAM SUPPLY LINE

Figure No.	Line No.	Break No.	Combined Stress (psi)	Rupture Stress 0.8 (1.2 S_h + S_a) (psi)
3.6-10	2-AFD-8	AFD8-B0-1	21426	32400
(Unit 2)	2-AFD-8	AFD8-B2-1**	-	32400
	2-AFD-7	AFD7-B0-1	17243	32400
	2-AFD-9	AFD7-B2-2**	16104	32400
	-	2	2199	32400
	-	91	25427	32400
	-	92**	18723	32400
		97**	17702	32400

Note: All breaks are circumferential.

** Breaks selected to satisfy the minimum number of intermediate breaks are no longer required.

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

Table 3.6-4
UNIT 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS - SI COLD LEG INJECTION

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE)
3.6-11 (Loop 1)	1-SI-5	SI5-B0-1N	30046	39448	C
		SI5-B0-2N	44038	39448	C,L
		SI5-B0-3	21902	38172	C
	1-SI-506	SI506-B0-1N	31669	37244	C
		SI506-B0-2N	35242	37244	C
		SI506-B0-3N	9354	38172	C
3.6-12 (Loop 4)	1-SI-4	SI4-B0-1N	23807	39448	C
		SI4-B0-2N	41325	39448	C,L
		SI4-B0-3N	27237	38172	C
	1-SI-511	SI511-B0-1N	14013	38172	C
		SI511-B0-2N	42275	37244	C
		SI511-B0-3N	32244	37244	C
3.6-13 (Loop 2)	1-SI-9	SI9-B0-1N	8827	39448	C
		SI9-B0-2	8950	39448	C,L,X
		SI9-B0-3	35707	38172	C
		SI9-B2-5	11998	39448	C,L
	1-SI-507	SI507-B0-1N	2878	37244	C
		SI507-B0-2N	25818	37244	C
		SI507-B0-3N	25815	37244	C
3.6-14 (Loop 3)	1-SI-10	SI10-B0-1N	13377	39448	C
		SI10-B0-3	12237	39448	C,L,X
		SI10-B0-4	15800	39448	C,L
		SI10-B0-5	43670	38172	C
		SI10-B1-6	16297	39448	C,L,X
		SI10-B1-7	11652	39448	C,L,X
	1-SI-510	SI510-B0-1N	45985 (61970)	37244	C
		SI510-B0-2N	43727 (57446)	37244	C
		SI510-B0-3N	41913 (56095)	37244	C

Note: C = Circumferential,
L = Longitudinal split
X = Break not required to be postulated

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

Table 3.6-4
UNIT 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS - SI COLD LEG INJECTION

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE)
3.6-11 (Loop 1)	1-SI-5	SI5-B0-1N	30046	39448	C
		SI5-B0-2N	44038	39448	C,L
		SI5-B0-3**	21902	38172	C
	1-SI-506	SI506-B0-1N**	31669	37244	C
		SI506-B0-2N**	35242	37244	C
		SI506-B0-3N**	9354	38172	C
3.6-12 (Loop 4)	1-SI-4	SI4-B0-1N	23807	39448	C
		SI4-B0-2N	41325	39448	C,L
		SI4-B0-3N**	27237	38172	C
	1-SI-511	SI511-B0-1N**	14013	38172	C
		SI511-B0-2N**	42275	37244	C
		SI511-B0-3N**	32244	37244	C
3.6-13 (Loop 2)	1-SI-9	SI9-B0-1N	8827	39448	C
		SI9-B0-2	8950	39448	C,L,X
		SI9-B0-3**	35707	38172	C
		SI9-B2-5	11998	39448	C,L
	1-SI-507	SI507-B0-1N**	2878	37244	C
		SI507-B0-2N**	25818	37244	C
		SI507-B0-3N**	25815	37244	C
3.6-14 (Loop 3)	1-SI-10	SI10-B0-1N	13377	39448	C
		SI10-B0-3	12237	39448	C,L,X
		SI10-B0-4	15800	39448	C,L
		SI10-B0-5**	43670	38172	C
		SI10-B1-6	16297	39448	C,L,X
		SI10-B1-7	11652	39448	C,L,X
	1-SI-510	SI510-B0-1N**	45985 (61970)	37244	C
		SI510-B0-2N**	43727 (57446)	37244	C
		SI510-B0-3N**	41913 (56095)	37244	C

Note: C = Circumferential, L = Longitudinal split X = Break not required to be postulated
Stress shown in parenthesis is Unit 2 stress at the same location

Breaks selected to satisfy the minimum number of intermediate breaks are no longer required. Note that longitudinal splits, although indicated above, were not required to be considered at intermediate locations where the criteria for a minimum number of break locations was applied.

** Class 1 stresses are not shown. Note that in this historic analysis the Class 1 piping was designed to the requirements of subsection C.

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal.

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TABLE 3.6-5
UNIT 1

SUMMARY OF STRESSES AT BREAK LOCATIONS
RHR/SI HOT LEG RECIRCULATION, LOOP 4

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR, U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-15	1-RHR-6	RHR6-B0-1N	59159	9242	24065	0.011	C	A
		RHR6-B0-2N	89977	10427	20308	0.872	C,L	C
		RHR6-B0-3N	93475	14639	10414	0.672	C,L	C
		RHR6-B0-4	91510	14148	14188	0.763	C,L	C
		RHR6-B0-5	77232	14128	20308	0.402	C,L	C
		RHR6-B0-6N	72556	4273	20554	0.302	C,L	C
		RHR6-B0-7N	78263	3018	26597	0.303	C,L	C
		RHR6-B0-8N	77616	15111	13909	0.173	C,L	C
	1-RHR-7	RHR7-B0-1	86735	10297	25388	0.555	C,L	C
		RHR7-B0-2N	93711	23140	24120	0.841	C,L	C
	1-SI-14	SI14-B0-1N	88393	21576	25922	0.432	C	A
		SI14-B0-2N	56917	8574	15679	0.0	C	B

Allowable stress intensity values: $3S_m$ = 58800 psi, $2.4S_m$ = 47040 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III.
2) U is based on NB-3653.5 if $S_n < 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n < 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or S > $2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per Equation 10.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

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TABLE 3.6-5
UNIT 2

SUMMARY OF STRESSES AT BREAK LOCATIONS
RHR/SI HOT LEG RECIRCULATION, LOOP 4

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR, U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-15	1-RHR-6	RHR6-B0-1N	59159	9242	24065	0.011	C	A
		RHR6-B0-2N	89977	10427	20308	0.872	C,L	B
		RHR6-B0-3N	93475	14639	10414	0.672	C,L	B
		RHR6-B0-4	91510	14148	14188	0.763	C,L	B
		RHR6-B0-5	77232	14128	20308	0.402	C,L	B
		RHR6-B0-6N	72556	4273	20554	0.302	C,L	B
		RHR6-B0-7N	78263	3018	26597	0.303	C,L	B
		RHR6-B0-8N	77616	15111	13909	0.173	C,L	B
	1-RHR-7	RHR7-B0-1	86735	10297	25388	0.555	C,L	B
		RHR7-B0-2N	93711	23140	24120	0.841	C,L	B
	1-SI-14	SI14-B0-1N	88393	21576	25922	0.432	C	A
		SI14-B0-2N	56917	8574	15679	0.0	C	C

Allowable stress intensity values: $3S_m$ = 48996-58950 psi, $2.4S_m$ = 39197-47160 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III.
2) U is based on NB-3653.5 if $S_n < 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n < 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or $S > 2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per Equation 10. (This is no longer required).

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

TABLE 3.6-6
UNIT 1

SUMMARY OF STRESSES AT BREAK LOCATIONS
SI HOT LEG RECIRCULATION < LOOPS 1, 2, AND 3

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-16	1-SI-15	SI15-B0-1N	45624	1788	20504	0.0	C	A
3.6-17	1-RHR-4	RHR4-B0-1	41998	14358	19733	0.0	C	A
	1-RHR-5	RHR5-B0-1	62824	33251	16372	0.006	C	A
		RHR5-B0-2N	58135	33855	15368	0.0	C	C
		RHR5-B0-3	59832	32948	15489	0.0	C	C

Allowable stress intensity values: $3S_m$ = 50592 psi, $2.4S_m$ = 40474 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III
2) U is based on NB-3653.5 if $S_n < 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n < 3S_m$, and $U > 0.1$; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or $S > 2.4S_m$, or $U > 0.1$
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per equation 10

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

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TABLE 3.6-6
UNIT 2

SUMMARY OF STRESSES AT BREAK LOCATIONS
SI HOT LEG RECIRCULATION < LOOPS 1, 2, AND 3

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-16	1-SI-15	SI15-B0-1N	45624	1788	20504	0.0	C	A
3.6-17	1-RHR-4	RHR4-B0-1	41998	14358	19733	0.0	C	A
	1-RHR-5	RHR5-B0-1	62824	33251	16372	0.006	C	A
		RHR5-B0-2N	58135	33855	15368	0.0	C	C
		RHR5-B0-3	59832	32948	15489	0.0	C	C

Allowable stress intensity values: $3S_m$ = 50592 psi, $2.4S_m$ = 40474 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III
2) U is based on NB-3653.5 if $S_n \leq 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n \leq 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or $S > 2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per equation 10, (this is no longer required).

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

TABLE 3.6-7

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TABLE 3.6-8

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TABLE 3.6-9 (Sheet 1 of 3)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4]

MAIN STEAM

PIPING SYSTEM Main Steam   PIPING NOMINAL DIA. 36 inch   PIPING SCHEDULE 1.307-inch wall

BREAK LOCATION	BREAK TYPE	THRUST[1] DIRECTION	WHIP FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
201 301 203 303	C	Downstream	Yes	Pipe whip into refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
				Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
205, 305, 207, 307	C	Downstream	Yes	Pipe whip into refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
				Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
		Upstream	Yes	Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
209, 309	C	Downstream	Yes	Pipe whip into refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
				Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
		Upstream	Yes	Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
211, 311	C	Downstream	Yes	Pipe whip damage to A1 wall of Auxiliary Building	Unacceptable, wall fails, environmental damage to essential components will result from steam entering Auxiliary Building	Restraints H31W, H21W
		Upstream	Yes	Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
217, 317	C	Downstream	Yes	Pipe whip damage to A1 wall of Auxiliary Building	Unacceptable, wall fails, environmental damage to essential components will result from steam entering Auxiliary Building	Restraints H31W, H21W
		Upstream	Yes	Pipe impact on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank

See Sheet 3 for Notes

TABLE 3.6-9 (Sheet 2 of 3)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4] (cont'd)

MAIN STEAM

PIPING SYSTEM Main Steam PIPING NOMINAL DIA. 36 inch PIPING SCHEDULE 1.307-inch wall

BREAK LOCATION	BREAK TYPE[1]	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
217, 317	L	Up	Yes	Jet impingement on ceiling of counting room and radio-chemical laboratory (unit 1 only)	Unacceptable, ceiling fails, environmental damage to essential components due to steam entering elevation 713 of the Auxiliary Building	Sleeves S217 S317
	L	Down	Yes	Pipe impact on ceiling of counting room and radio-chemical laboratory (unit 1 only)	Unacceptable, ceiling fails, environmental damage to essential components due to steam entering elevation 729 of the Auxiliary Building	Sleeves S217 S317
		Left	Yes	Pipe whip into Auxiliary Building HVAC intake	Unacceptable, environmental damage to essential components due to steam entering elevation 737 of the Auxiliary Building	Sleeves S217, S317
		Right	Yes	Jet impingement on Auxiliary Building HVAC intake	Unacceptable, environmental damage to essential components due to steam entering elevation 737 of the Auxiliary Building	Sleeves S217, S317
218, 318	C	Upstream	Yes	Pipe whip damage to south wall of south steam valve room	Unacceptable, damage to main steam and feedwater isolation valves located in valve room	Restraints G22W, G32W
418, 118, 218, 318	L	Right	Yes	Jet impingement on spreading room exhaust duct in C11-wall (unit 2 only)	Unacceptable, loss of Control Building habitability due to steam environment	HVAC to have 3 psi backdraft damper installed to prevent steam from entering control building
419, 119, 219, 319	C	Upstream	Yes	Pipe impact on elevation 755, which supports control room HVAC equipment	Unacceptable, floor fails and results in environmental damage to control room	Restraints L42D, L12D, L22D, L32D
420, 120, 220, 320	C	Upstream	Yes	Pipe impact on elevation 755, which supports control room HVAC equipment	Unacceptable, floor fails and results in environmental damage to control room	Restraints L42D, L12D, L22D, L32D
423 124	L	Up	Yes	Pipe impact on elevation 729 floor of Turbine Building adjacent to doors to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125
423, 124	L	Down	Yes	Jet Impingement on elevation 729 floor of Turbine Building adjacent to doors to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125

See Sheet 3 for Notes

TABLE 3.6-9 (Sheet 3 of 3)
SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4] (cont'd)

MAIN STEAM

PIPING SYSTEM Main Steam PIPING NOMINAL DIA. 36 inch PIPING SCHEDULE 1.307-inch wall

BREAK LOCATION	BREAK TYPE	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
424, 125, 225, 325	C	Upstream	Yes	Pipe impact on the N-wall of Control Building	Unacceptable, wall fails, environmental damage to essential components within Control Building	Restraints M42S, M32S, M22S, M12S
424, 125	L	Up	Yes	Pipe impact on elevation 729 floor of Turbine Building adjacent to door to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125
424, 125	L	Down	Yes	Jet impingement on elevation 729 floor of Turbine Building adjacent to door to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125
425, 126, 226, 326	C	Upstream	Yes	Pipe impact on N-wall of Control Building	Unacceptable, wall fails, environmental damage to essential components in Control Building	Restraints M42S, M12S, M22S, M32S
	Through-Wall Leakage Cracks					
	Through-wall leakage crack break below Control Building HVAC exhaust ducting at elevation 755 on Q-wall (unit 2 only)			Through-wall leakage crack break would fill control room HVAC with steam	Unacceptable, loss of habitability of control room	HVAC to have 3 psi backdraft damper installed to prevent steam from entering control room
	Through-wall leakage crack break below Auxiliary Building HVAC intake canopy at elevation 743 on A1-wall			Through-wall leakage crack break would fill Auxiliary Building with steam	Unacceptable, environmental damage to essential components in Auxiliary Building	HVAC to have temperature sensors installed which control intake fans, preventing steam from entering Auxiliary Building

In all other cases effects of through-wall leakage crack breaks are acceptable.

Notes:

1) Direction of thrust on pipe. Jet load is opposite.

For circumferential (C) breaks consider upstream thrust on the upstream pipe and downstream thrust on the downstream pipe.
For longitudinal (L) breaks consider up, down, lateral left, and lateral right thrust (facing downstream).

2) Whip trajectory is governed by hinge mechanism and direction of vector thrust of break force. Maximum 180° rotation about any plastic hinge. Sweep of jet is governed by pipe motion.

3) Type of effect (jet, whip, environment, etc.) and components affected.

4) This applies to unit 1. Unit 2 is opposite hand unless otherwise noted.

TABLE 3.6-10 (Sheet 1 of 2)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4]

FEEDWATER

PIPING SYSTEM Feedwater   PIPING NOMINAL Dia. 18 inch   PIPING SCHEDULE 80

BREAK LOCATION	BREAK[1] TYPE	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
418, 118, 218, 318	L	Left	No	Jet impingement on spreading room exhaust duct in C11-wall (unit 2 only)	Unacceptable loss of Control Building ventilation	Provide 3 psi backdraft damper
420, 120, 220, 320	L	Down	Yes	Pipe impact on elevation 708 floor of Control Building	Unacceptable, failure of floor allows pipe whip into electrical board room air handling units below. Environmental damage to essential components and loss of control room habitability may result.	Restraints J42U, J12U, J22U, J32U
421, 121, 221, 321	C	Downstream	Yes	Pipe impact on elevation 708 floor of Control Building	Unacceptable, failure of floor allows pipe whip into electrical board room air handling units below. Environmental damage to essential components and loss of control room habitability may result.	Restraints J42U, J12U, J22U, J32U
422, 122, 222, 322	C	Downstream	Yes	Pipe impact on elevation 708 floor of Control Building	Unacceptable, failure of floor allows pipe whip into electrical board room air handling units below. Environmental damage to essential components & loss of control room habitability may result.	Restraints J42U, J12U, J22U, J32U
423, 123, 223, 323	C	Upstream	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W, K12W, K22W, K32W
424, 124, 224, 324	C	Upstream	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W, K12W, K22W, K32W
425, 125, 225, 325	C	Downstream	Yes	Pipe impact on elevation 755 floor	Unacceptable, floor fails and results in environmental damage to control room.	Restraints K42W, K12W, K22W, K32W
	L	Down	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W, K12W, K22W, K32W

See Notes on Sheet 2

TABLE 3.6-10 (Sheet 2 of 2)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4]

FEEDWATER

PIPING SYSTEM Feedwater PIPING NOMINAL Dia. 18 inch PIPING SCHEDULE 80

BREAK LOCATION	BREAK[1] TYPE	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
426, 126, 226, 326	C	Downstream	Yes	Pipe impact on elevation 755 floor	Unacceptable, floor fails and results in environmental damage to control room.	Restraints K42W, K12W, K22W, K32W
	L	Down	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W. K12W, K22W, K32W
428, 128, 228, 328	C	Downstream	Yes	Pipe impact on elevation 755 floor	Unacceptable, floor fails and results in environmental damage to Control Building.	Restraints K42W, K12W, K22W, K32W
	Through-Wall Leakage					
	Through-wall leakage crack break below Control Building HVAC exhaust ducting at elevation 755 on Q-wall (unit 2 only)			Through-wall leakage crack break would fill control room HVAC with steam	Unacceptable, loss of habitability of control room	HVAC to have 3 psi backdraft damper installed to prevent steam entering control room
	Through-wall leakage crack break below Auxiliary Building HVAC intake canopy at elevation 743 on A1-wall			Through-wall leakage crack break would fill Auxiliary Building HVAC with steam	Unacceptable, environmental damage to essential components in Auxiliary Building	HVAC to have temperature sensors installed, which control intake fans, preventing steam from entering Auxiliary Bldg.

In all other cases, effects of through-wall leakage crack breaks are acceptable.

Notes:

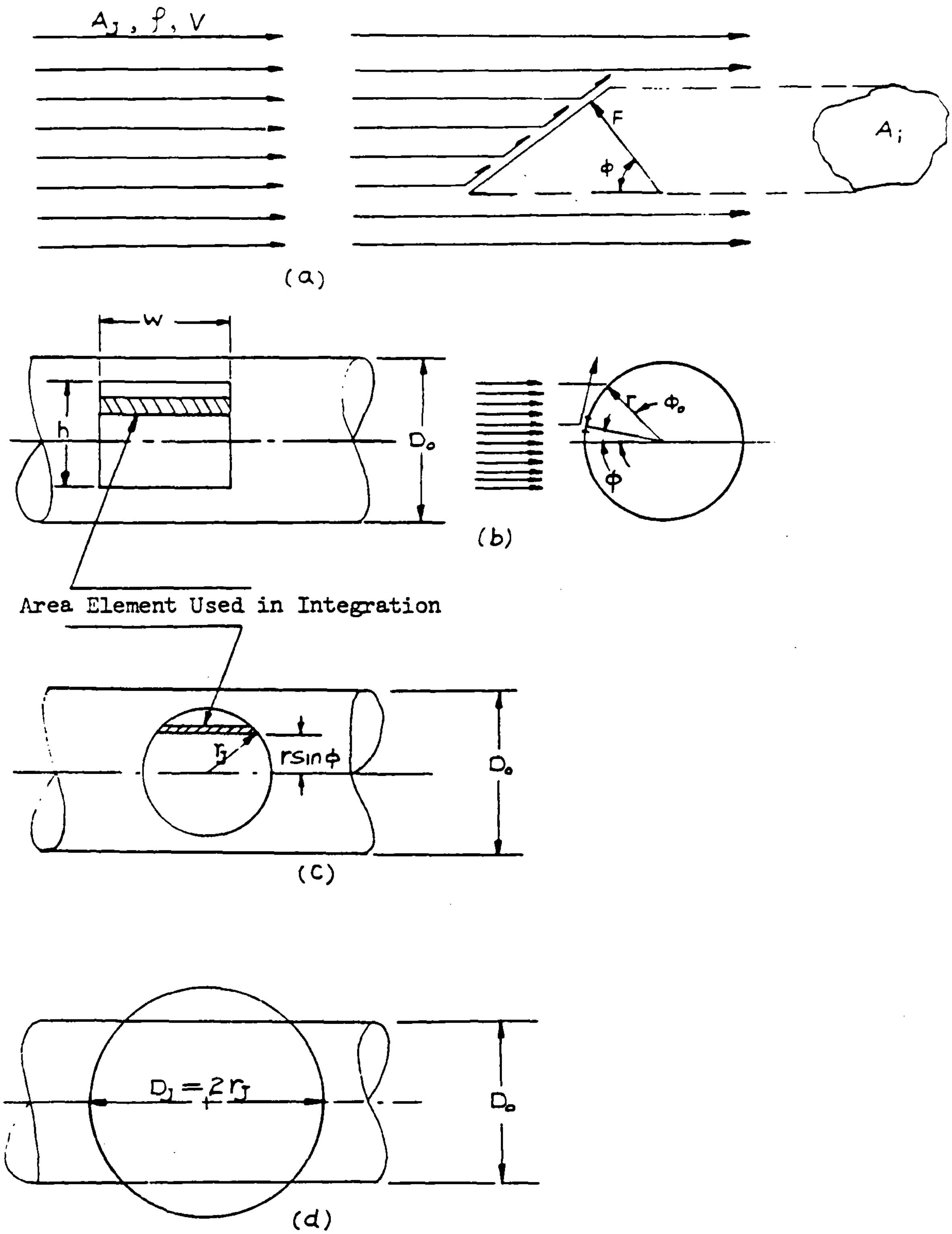
1) Direction of thrust on pipe. Jet load is opposite.
For circumferential (C) breaks consider upstream thrust on the upstream pipe and downstream thrust on the downstream pipe.
For longitudinal (L) breaks consider up, down, lateral left, and lateral right thrust (facing downstream).

2 Whip trajectory is governed by hinge mechanism and direction of vector thrust of break force. Maximum 180° rotation about any plastic hinge. Sweep of jet is governed by pipe motion.

3) Type of effect (jet, whip, environment, etc.) and components affected.

4) This applies to Unit 1. Unit 2 is opposite hand unless otherwise noted.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL

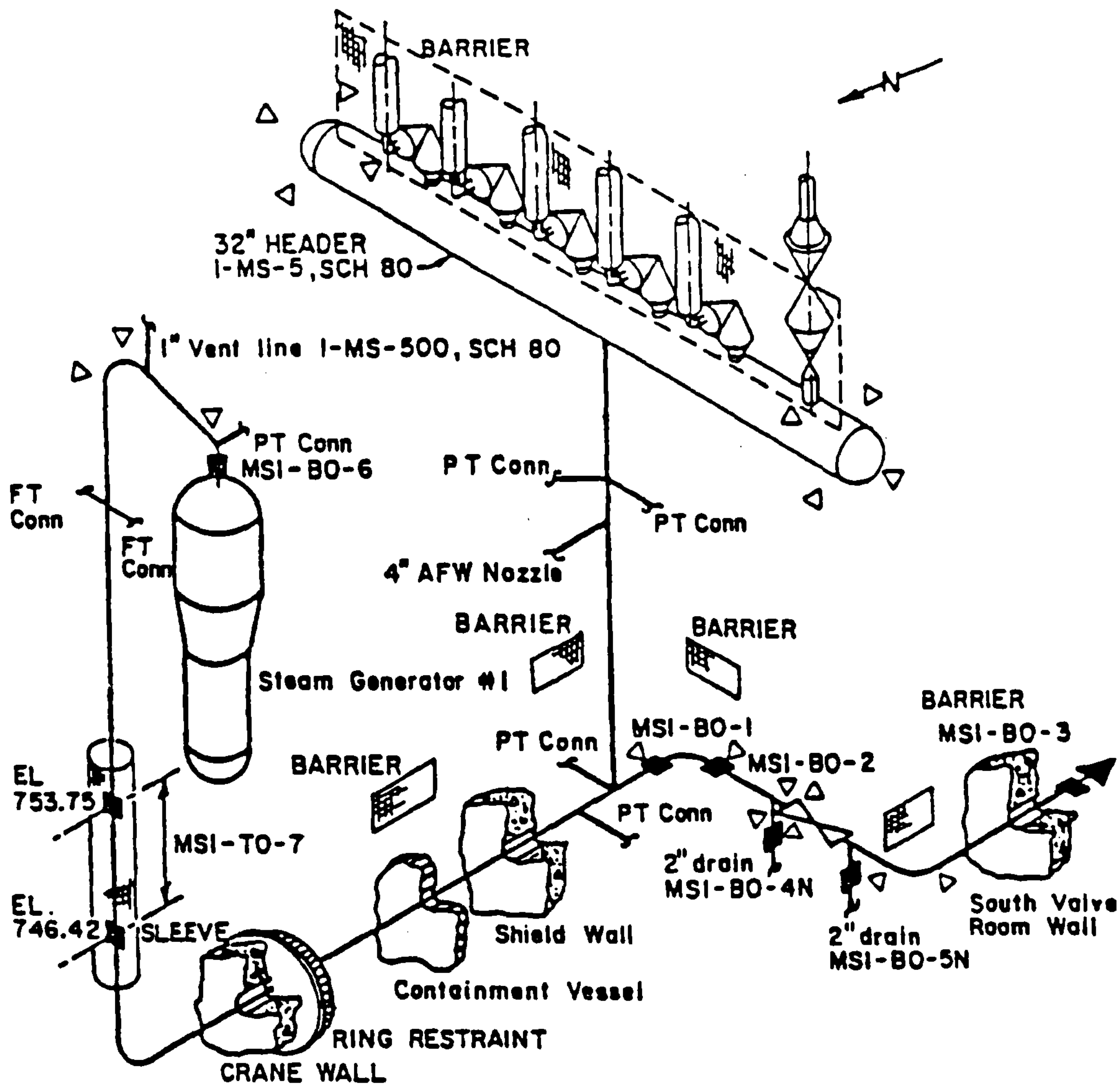


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Shape Factors

FIGURE 3.6-1

MAIN STEAM LINE FROM
STEAM GENERATOR #1

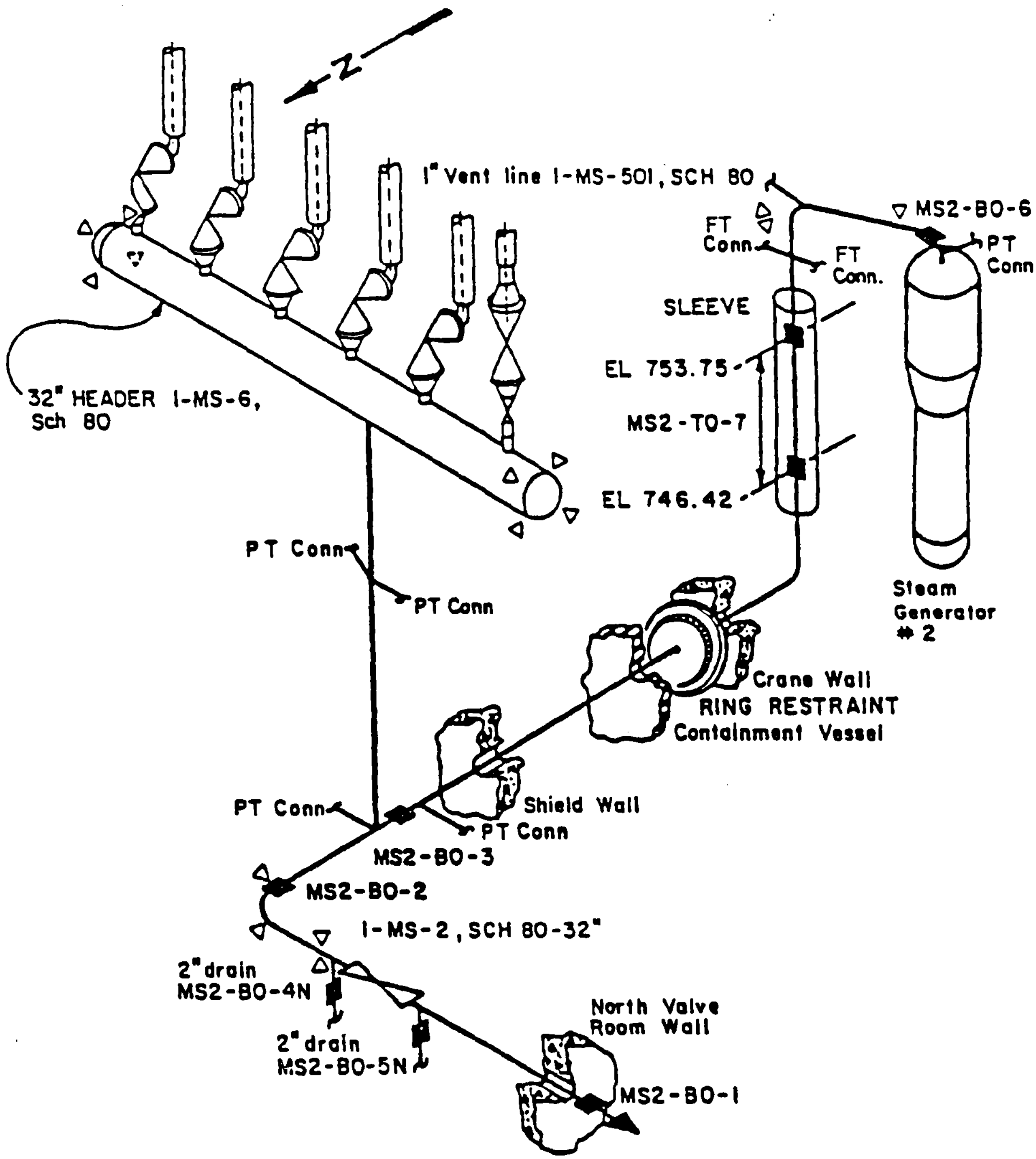


▷ DENOTES DIRECTION & LOCATION OF PROTECTIVE DEVICE RUPTURE RESTRAINTS, TYPICAL ALL FIGURES

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-2

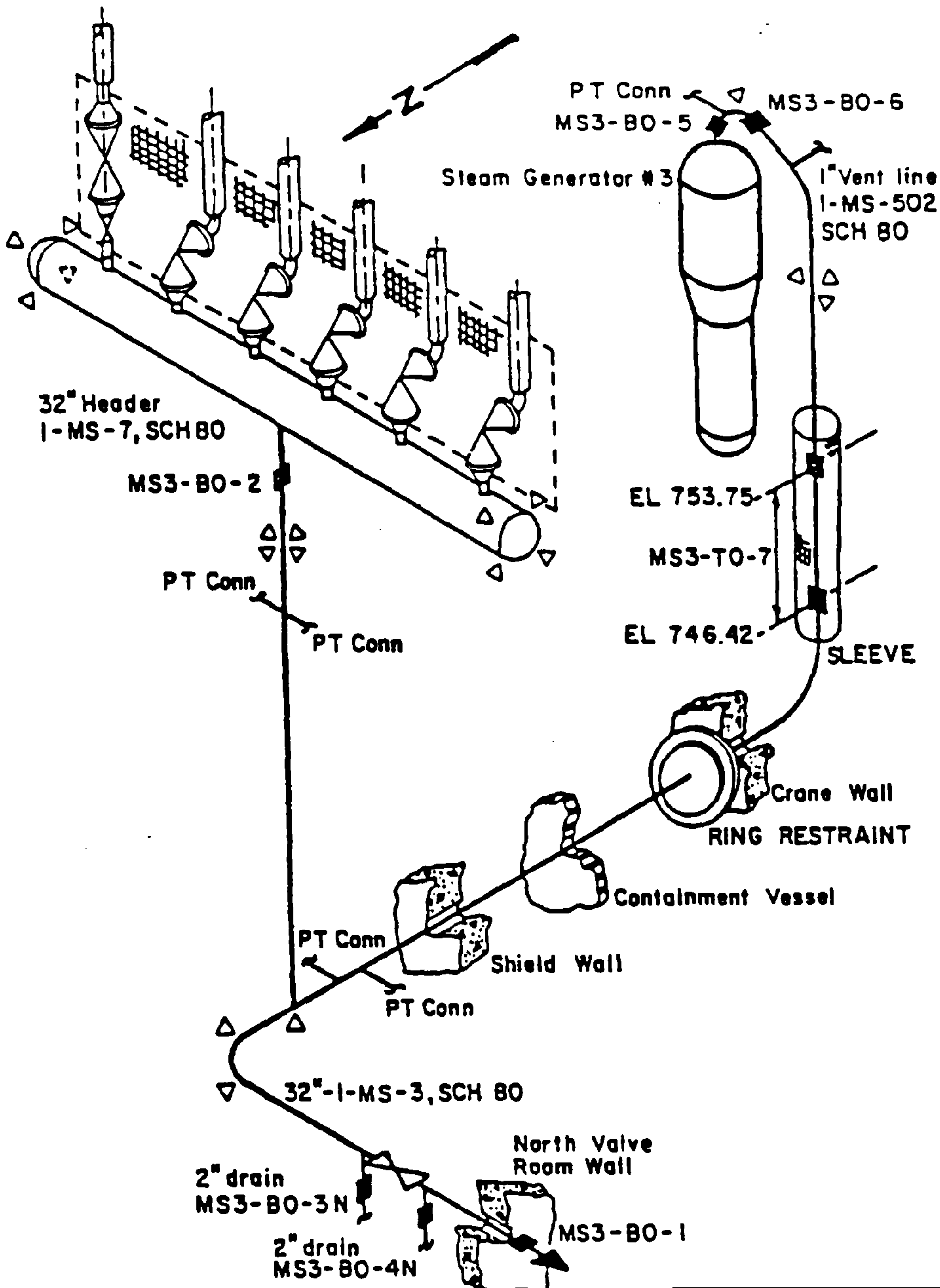
MAIN STEAM LINE FROM STEAM GENERATOR #2



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-3

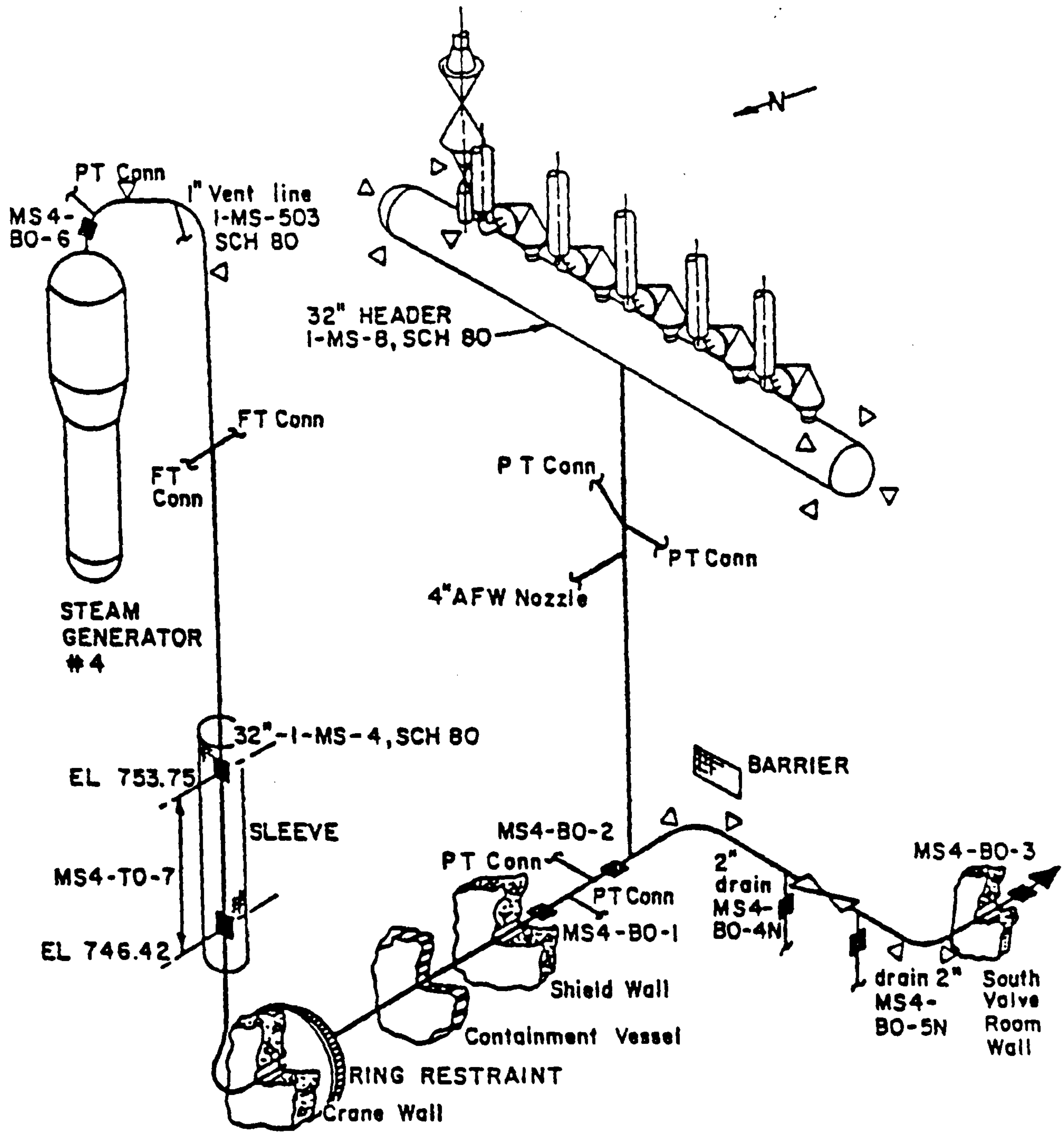
MAIN STEAM LINE FROM STEAM GENERATOR #3



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-4

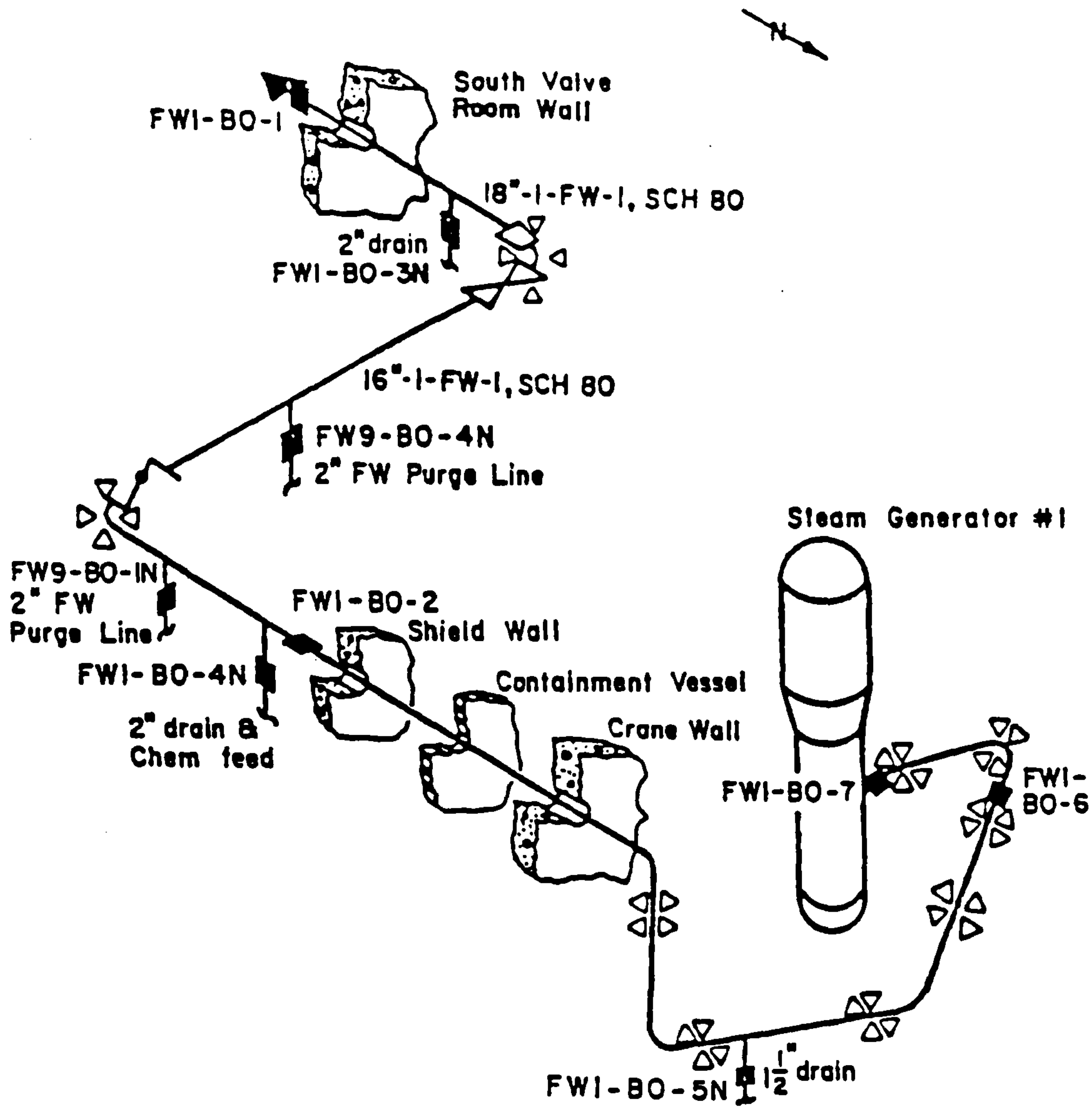
MAIN STEAM LINE FROM
STEAM GENERATOR #4



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-5

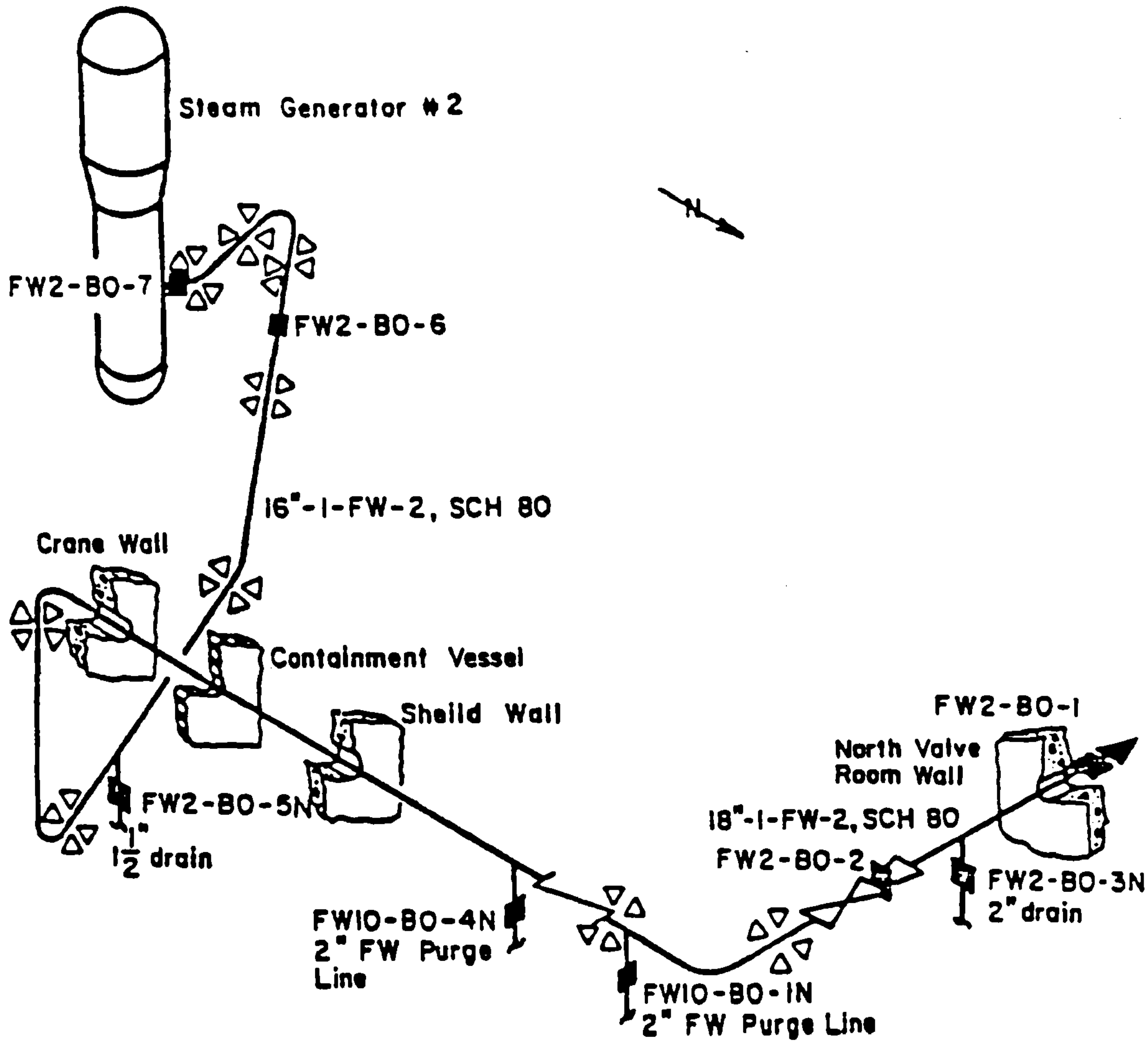
FEEDWATER LINE TO
STEAM GENERATOR #1



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-6

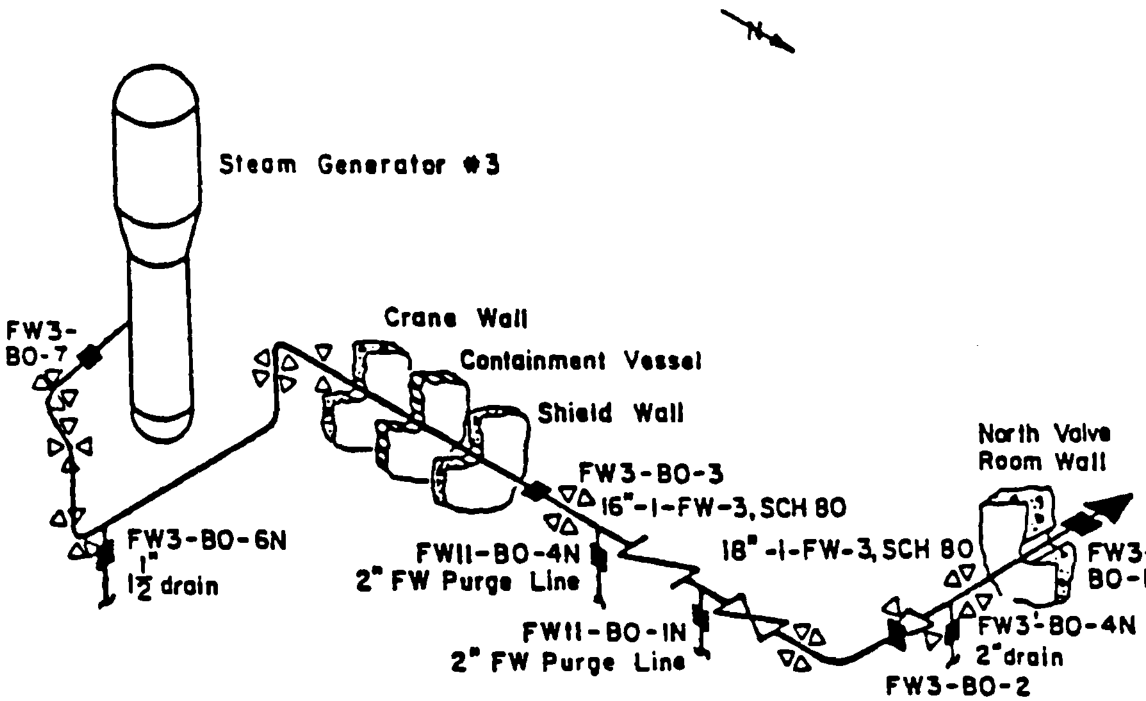
FEEDWATER LINE TO
STEAM GENERATOR #2



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

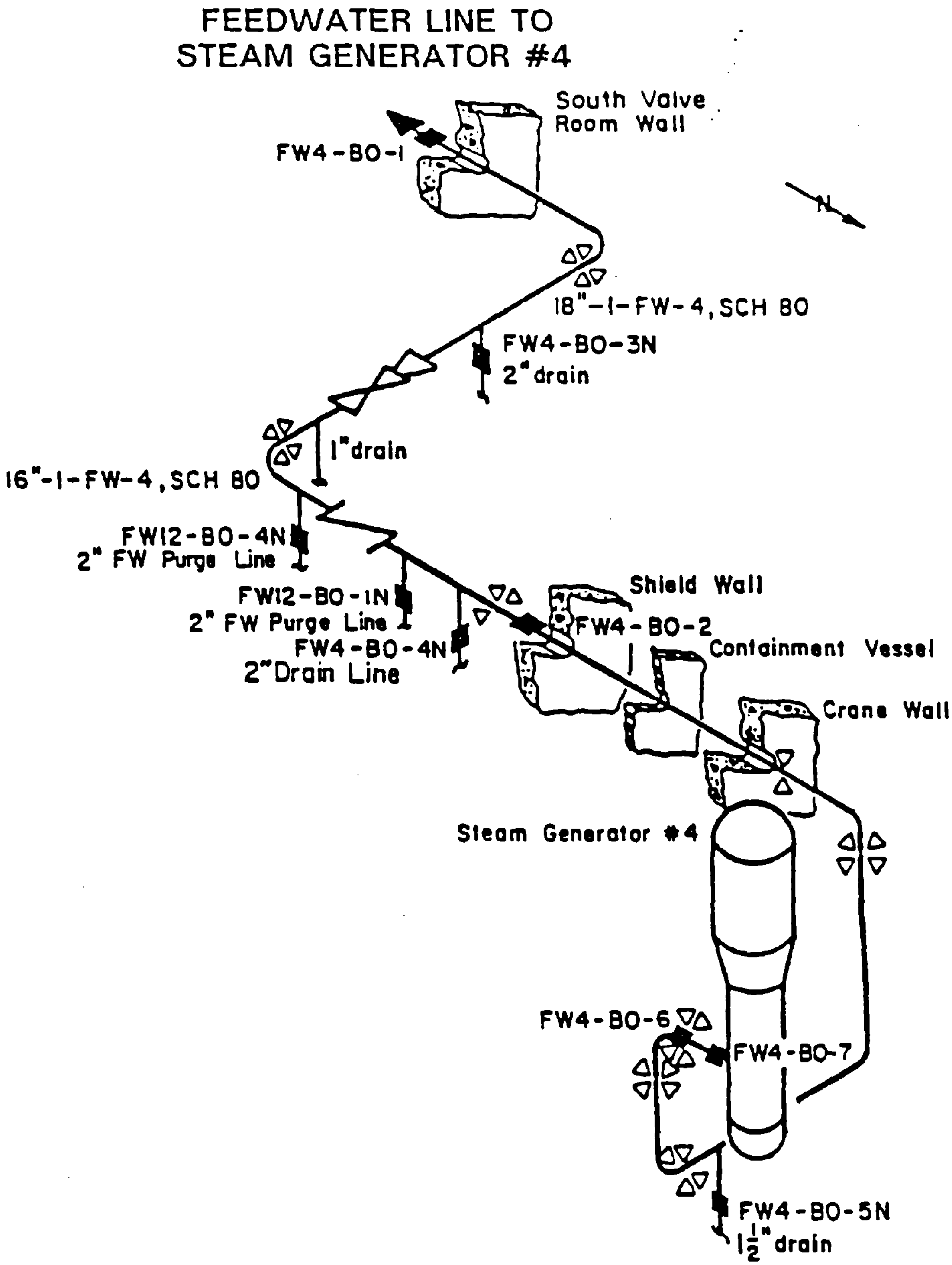
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-7

FEEDWATER LINE TO
STEAM GENERATOR #3



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-8

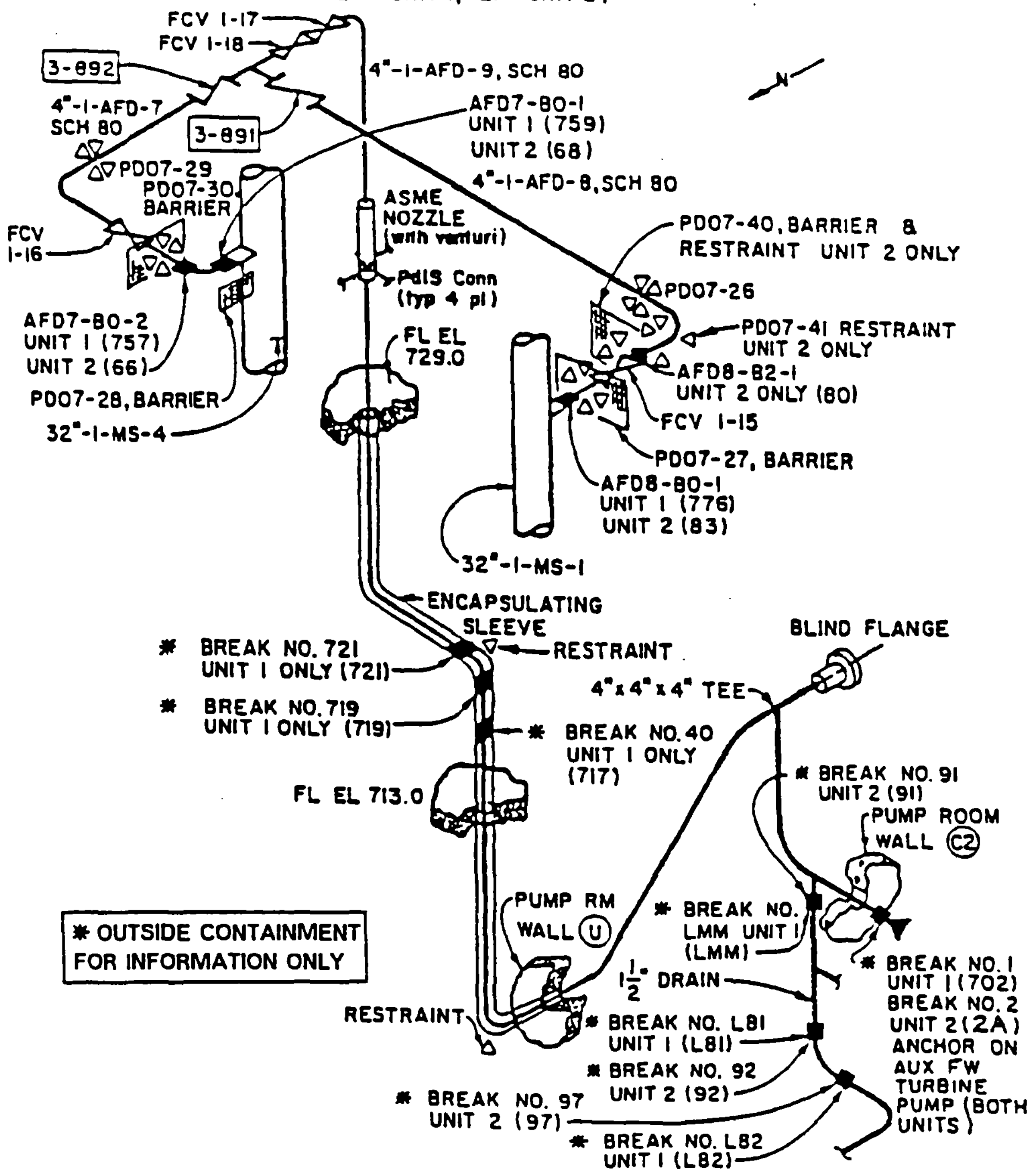


REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FIGURE 3.6-9

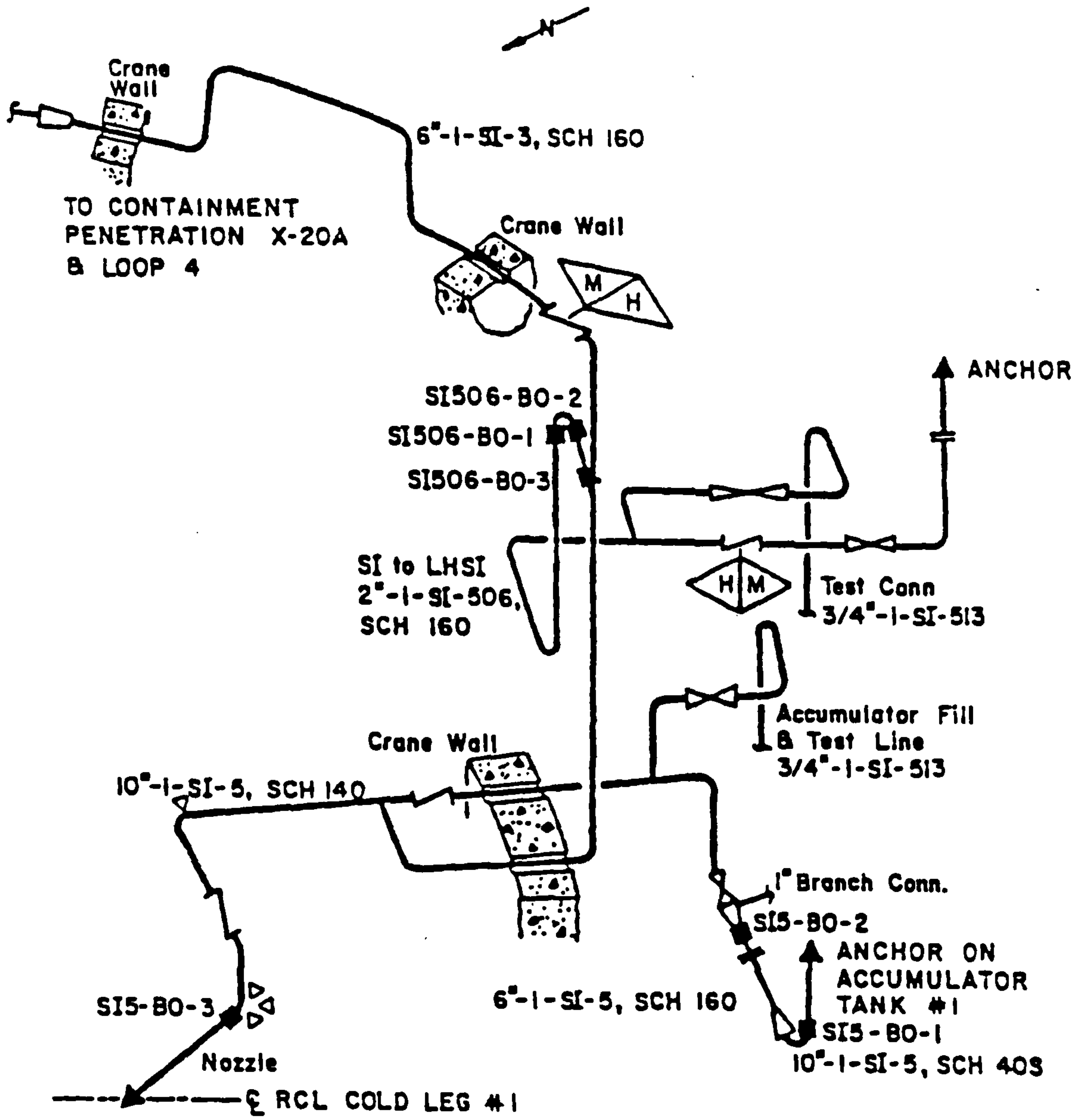
AUXILIARY FEEDWATER STEAM SUPPLY LINES



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 62 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-10

SI COLD LEG INJECTION
LOOP 1



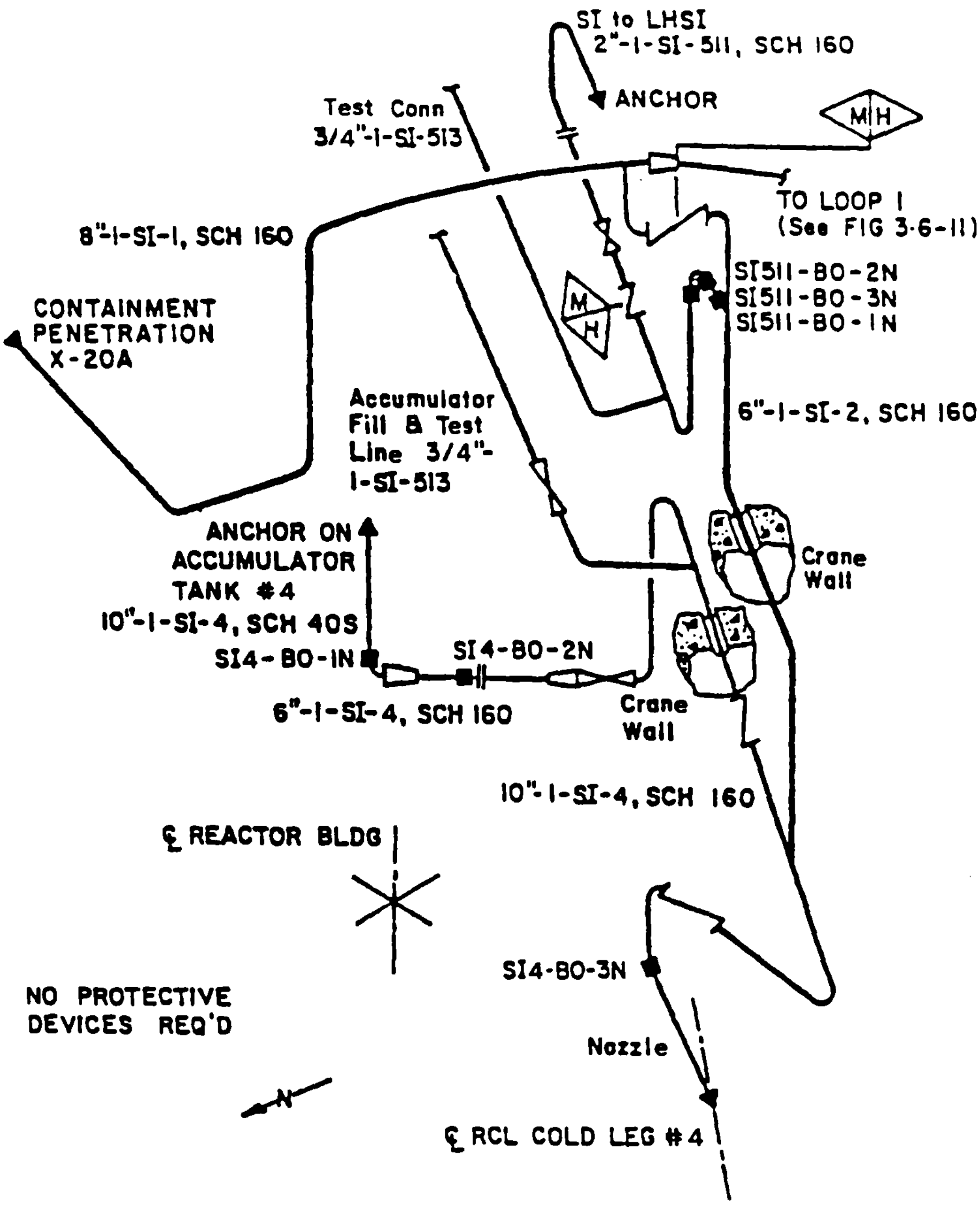
REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FIGURE 3.6-11

SI COLD LEG INJECTION

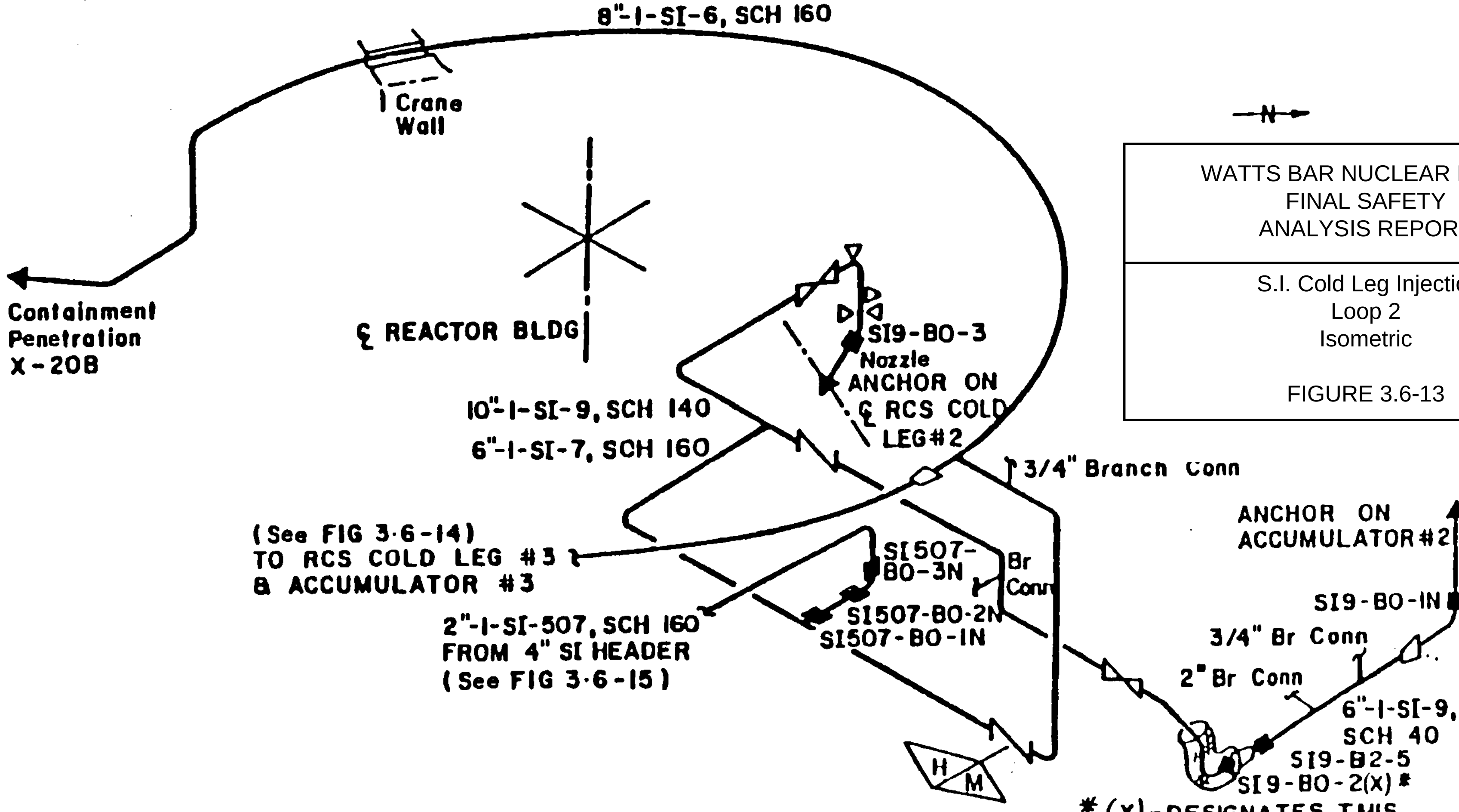
LOOP 4



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-12

SI COLD LEG INJECTION
LOOP 2



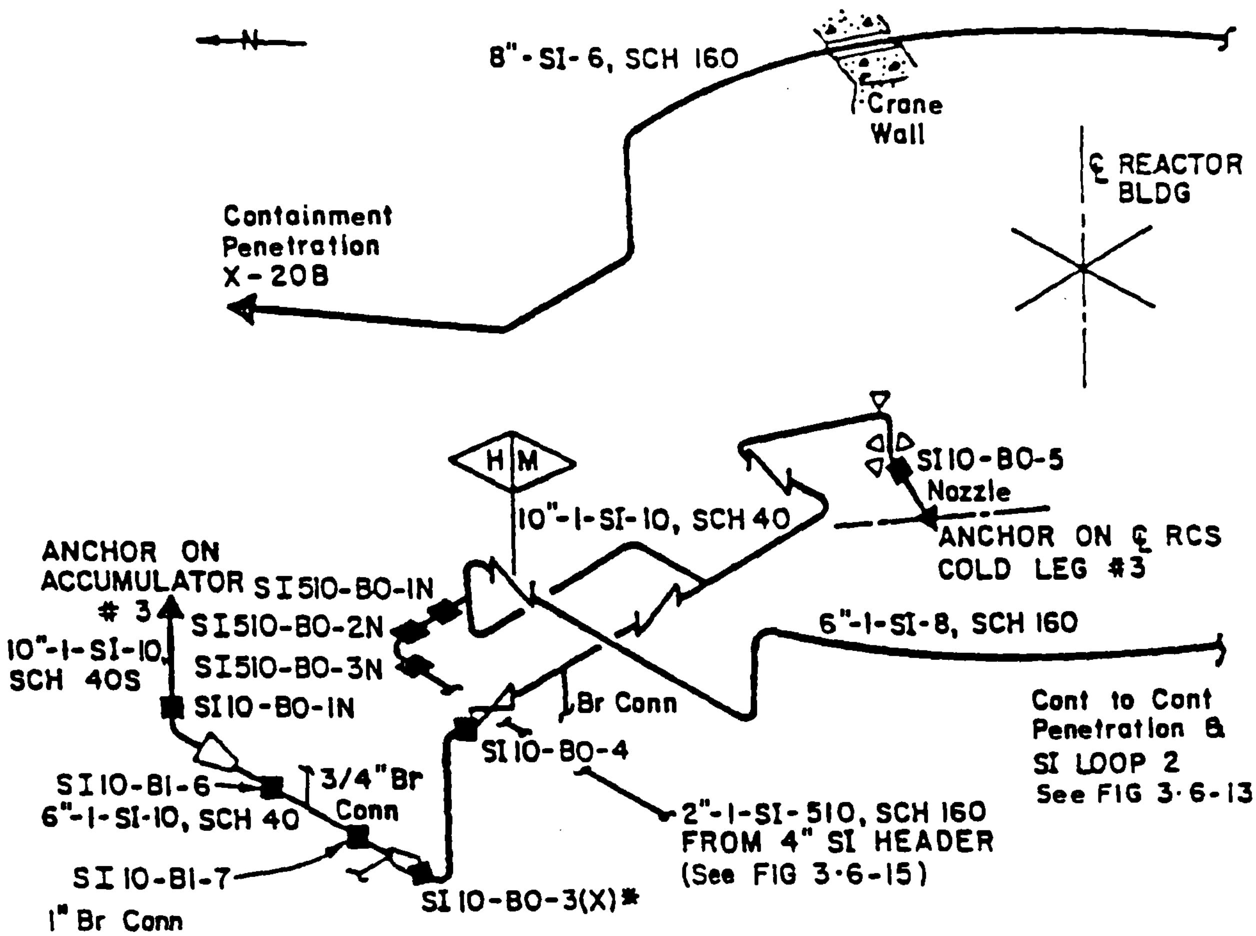
* (X)-DESIGNATES THIS BREAK NOT REQUIRED FOR LICENSING

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 61 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
S.I. Cold Leg Injection Loop 2 Isometric FIGURE 3.6-13

SI COLD LEG INJECTION

LOOP 3

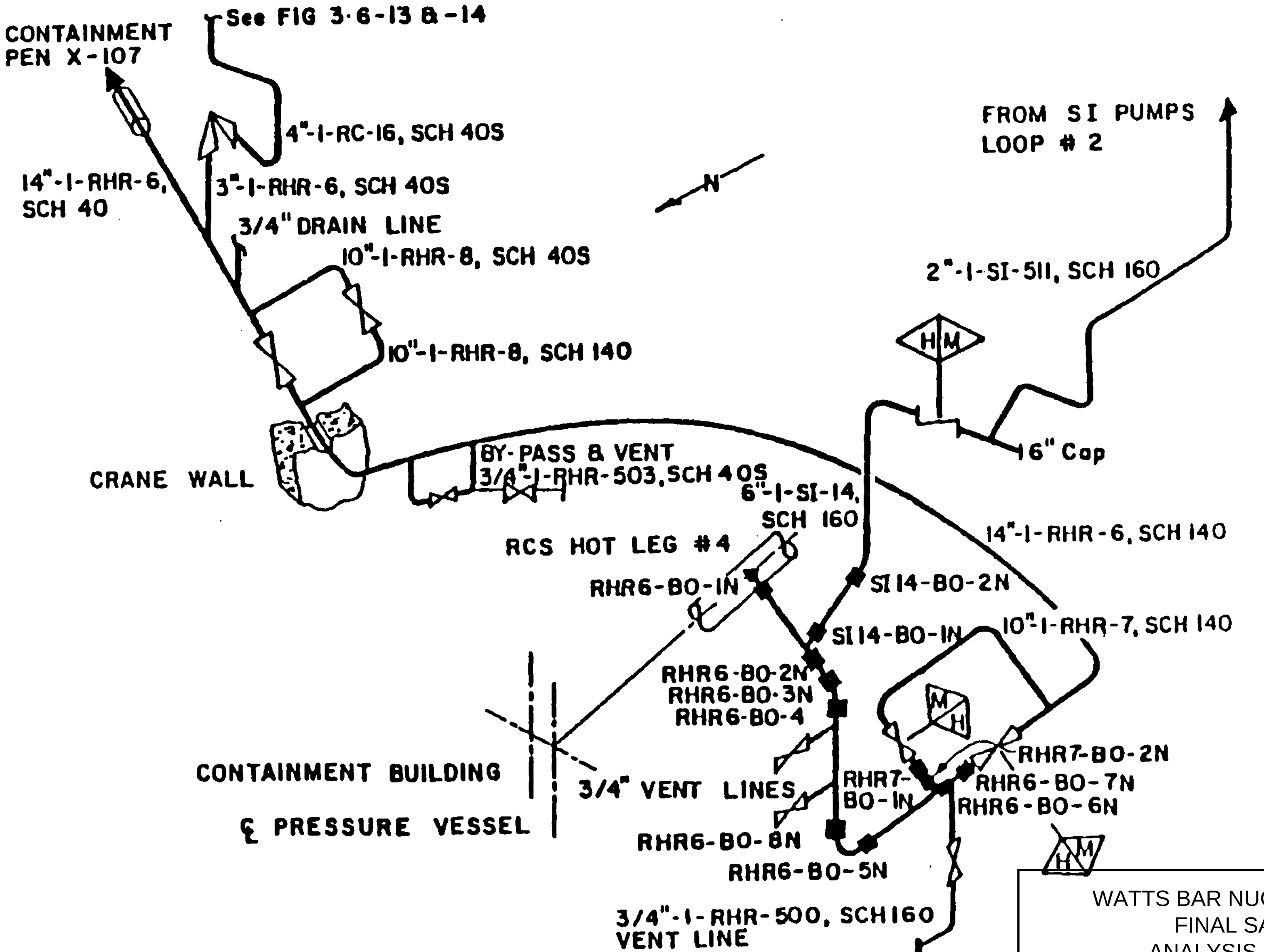


*(X)- DESIGNATES THIS BREAK NOT REQUIRED FOR LICENSING

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-14

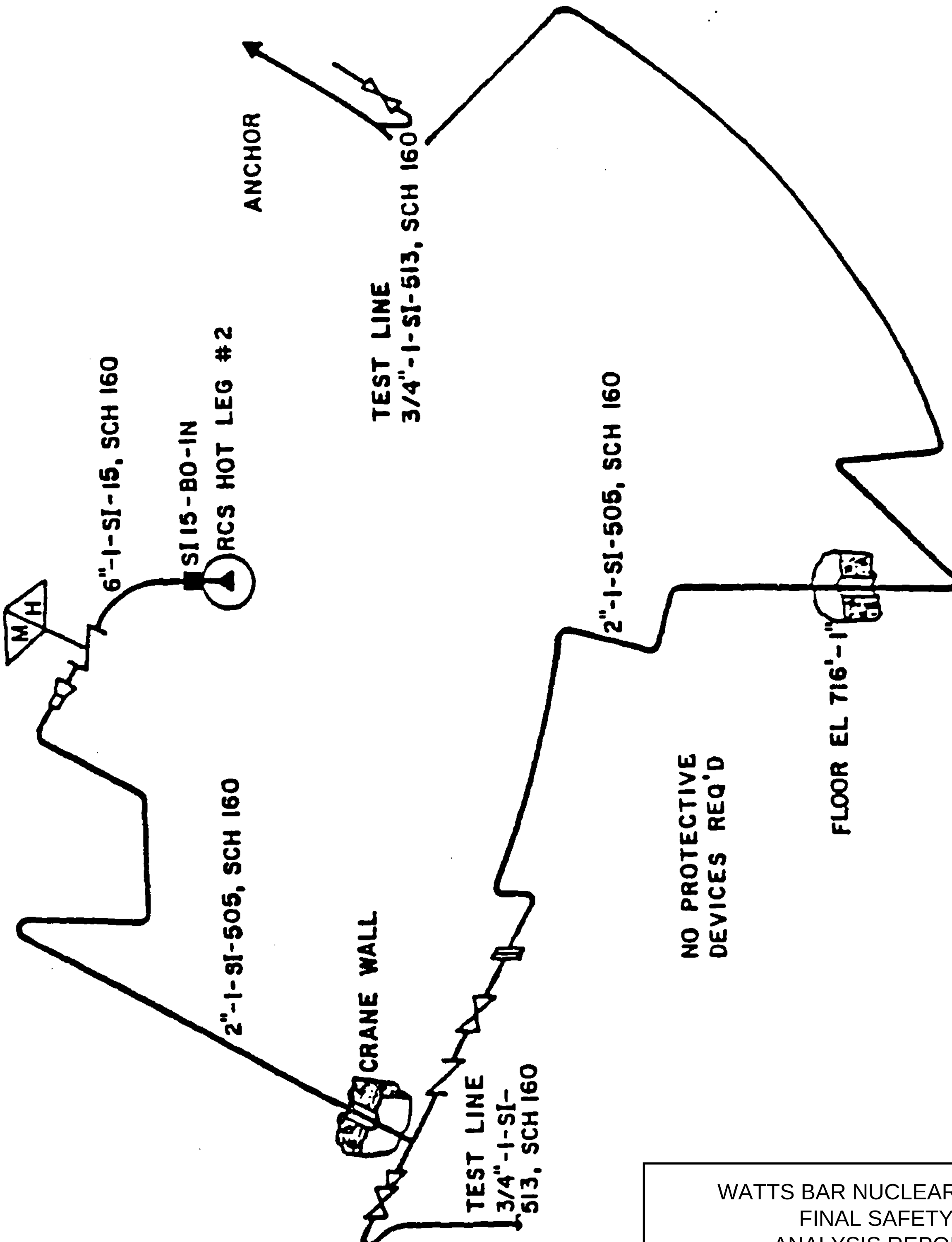
REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
61 SUBMITTAL



RHR/SI HOT LEG
RECIRCULATION, LOOP 4

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
RHR/S.I. Hot Leg Recirculation Loop 4 Isometric FIGURE 3.6-15

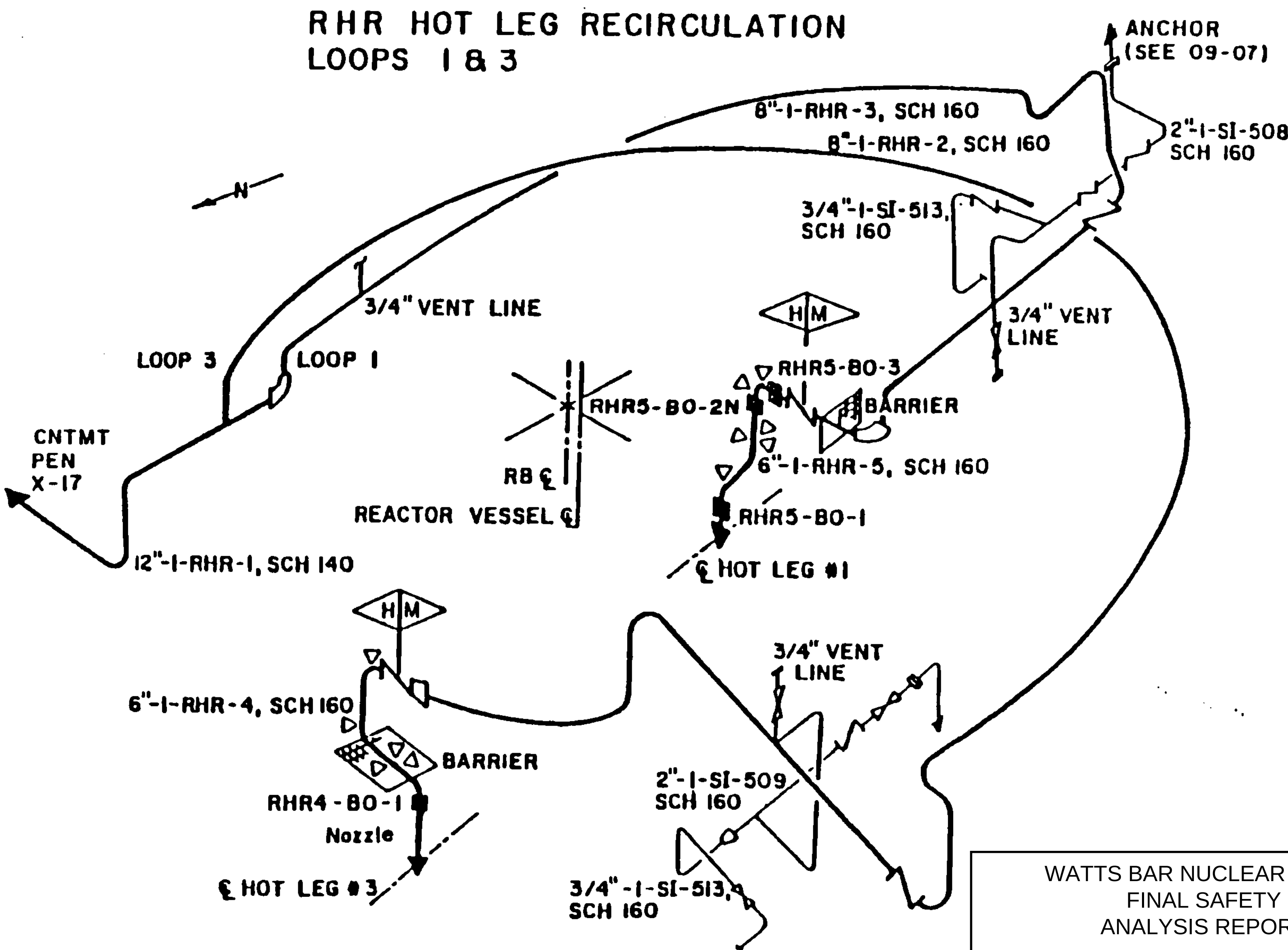
SI HOT LEG RECIRCULATION
LOOP 2



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
S.I. Hot Leg Recirculation Loop 2 Isometric FIGURE 3.6-16

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 61 SUBMITTAL.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RHR Hot Recirculation
Loops 1 and 3
Isometric
FIGURE 3.6-17

FIGURE 3.6-18

THRU

FIGUR 3.6-20

DELETED

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-21

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-22

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-23

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-24

APPENDIX 3.6A

PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (EXCLUDING REACTOR COOLANT SYSTEM PIPING)

Criteria presented herein regarding break size, shape, orientation, and location are in accordance with the guidelines transmitted to TVA by the NRC in letter, dated December 1972, and subsequent amendments for outside containment, and NRC Regulatory Guide 1.46 for inside containment. These criteria also include considerations which are further clarified in the NRC Branch Technical Positions ASB 3-1 and MEB 3-1 where appropriate. Arbitrary intermediate breaks (AIBs) postulated in accordance with the documents noted above are eliminated by NRC Generic Letter 87-11[4].

The final routing of field routed systems was not completed until late in the plant construction schedule. Field-routed piping generally possesses very little potential, insofar as their functions are concerned, toward affecting plant shutdown. Their failure can, however, cause damage to other components and equipment, especially electrical, which may be required for shutdown of the plant. Field-routed and field-located items such as electrical conduit, cable trays, instrument and control lines, and junction and terminal boxes, etc., are protected as required for plant shutdown. Where field routing was required, guidance was provided to minimize the number of unacceptable interactions. A followup field review and evaluation, for identifying unacceptable interactions and ensuring implementation of corrections was performed.

The following definitions and assumptions are applicable to this section:

DEFINITIONS

1. Acceptable Interaction

   A pipe rupture interaction for which, from a systems standpoint, the net required safety functions for a particular rupture are not impaired when assuming a single active component

2. Active Component

   Any component which must perform a mechanical motion or change of state during the course of accomplishing a primary safety function.

3. Double-Ended Rupture

   A circumferential pipe rupture where flow is sustained from both ends of the break.

4. Environmental Effects

   The wetting, pressure, temperature, flammable, radiation, etc., conditions within the 'zone of influence' (Definition 28) of a pipe rupture.

5. <u>Essential Systems and Components</u>

Systems and components required to shutdown the reactor and/or mitigate the consequences of a postulated pipe failure without offsite power. The seismic classification of essential components and systems is in accordance with Regulatory Guide 1.29.

6. <u>High Energy Fluid Systems</u>

Fluid systems that, during normal plant conditions, satisfy the following:

a. Maximum operating temperature exceeds 200 °F, and

b. Maximum operating pressure exceeds 275 psig.

Systems may be classified as moderate energy (see Definition 13) if the total time that the above conditions are exceeded is less than either of the following:

a. One percent of the normal operating life span of the plant.

b. Two percent of the time period required for the system to accomplish its design function.

7. <u>Inside Containment</u>

Inside containment is defined for pipe rupture evaluation purposes to include all piping inside the Shield Building and the main steam valve rooms. The actual containment boundary for integrity purposes is normally taken at the second isolation valve.

8. <u>Jet Impingement Force</u>

The jet force on an object resulting from a ruptured pipe. The magnitude of this force depends on such parameters as the thermodynamic conditions of the fluid in the pipe, distance of the pipe rupture from the target and the shape of the target.

9. <u>Jet Thrust</u>

That reactive dynamic force on a ruptured pipe due to a fluid being accelerated out of a break.

10. <u>Line-Mounted Valves</u>

Valves located in a line and supported by the line.

11. <u>Loss-of-Coolant Accident (LOCA)</u>

LOCA is defined as a net loss of reactor coolant inventory when makeup is provided only by the normal makeup system and an orderly shutdown of the plant is prevented. Normal makeup is sized to maintain a constant reactor coolant system (RCS) inventory with a rupture equivalent to a 3/8-inch diameter hole. Therefore, a rupture is considered a LOCA when the flow rate is greater than the equivalent flow from a 3/8-inch diameter hole.

12. <u>LOCA Boundary</u>

For piping extended from the RCS, the boundary of postulated pipe rupture which cannot be isolated when assuming a single active failure shall be defined as follows:

a. First locked closed or administratively closed isolation valve (pressurizer safety valves are examples). The valves forming the Class 1 boundary in all drain lines are considered as administratively closed.

b. Second of two normally open, remotely operable, independent isolation valves capable of automatic closure and verification that they will close.

c. First normally closed check valve capable of verification that it is closed and capable of providing isolation from a reactor coolant source.

d. Second of two normally open check valves capable of verification that they will close and capable of providing isolation from a reactor coolant source. (Verification that a check valve will close should be interpreted as meaning 'capable of periodic test that will verify its capability of closure, such as during a refueling outage.')

e. First normally open and remotely operable automatic isolation valve following a normally open check valve (capable of providing isolation from a reactor coolant source) if both are capable of verification that they will close.

If a pipe failure beyond the above defined boundary of possible isolation could result in a normally open boundary valve failing to close, then a LOCA may exist beyond that boundary.

13. <u>Moderate Energy Fluid Systems</u>

Fluid systems that, during normal plant conditions, satisfy either of the following:

a. Maximum operating temperature is 200°F or less or

b. Maximum operating pressure is 275 psig or less.

Other systems which may be classified as moderate energy are discussed in Definition 6.

14. Normal Plant Conditions

Plant operating conditions during reactor startup, refueling, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

15. Outside Containment

Outside containment includes all of those regions not included in the definition of 'Inside Containment' (Definition 7)

16. Pipe Whip

The movement of a pipe caused by jet thrust resulting from a pipe failure. Pipe whip is assumed to occur in the plane defined by piping geometry and configuration unless limited by structural members, pipe restraints, or pipe stiffness.

17. Primary Safety Function

The passive or active function of a structure, system, or component which must remain functional to assure directly: (1) the integrity of the RCPB, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the guideline exposure of 10 CFR 100.

18. Postulated Piping Failures

Longitudinal splits, circumferental ruptures, or through-wall leakage cracks.

19. Protective Structures or Compartments

Structural units provided to separate or enclose redundant trains of safety-related systems or enclose high and moderate energy lines. (These structures are designed as Seismic Category I)

20. Reactor Coolant Pressure Boundary (RCPB)

Those pressure containing components such as pressure vessels, piping, pumps, and valves, which are:

a. Part of the reactor coolant system or

b. Connected to the reactor coolant system, up to and including any of the following:

1. The outermost containment isolation valve in system piping which penetrates the containment.

2. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the containment.

3. The reactor coolant system safety and relief valves.

21. Safety Related

Those plant features which are important to safety because they perform either a primary safety function or a secondary safety function.

22. Secondary Safety Function

The function of a portion of a structure, systems or component which must retain limited structural integrity because its failure could jeopardize the achievement of a primary safety function or because it forms an interface between Seismic Category I and Seismic Category I(L) or non-seismic plant features.

23. Seismic Category I

Those structures, systems, or components which perform primary safety functions are designated as Seismic Category I and are designed and constructed so as to assure achievement of their primary safety functions at all times including a concurrent safe shutdown earthquake (SSE).

24. Seismic Category I(L)

Those portions of structures, systems, or components which perform secondary safety functions and are designed and constructed so as to assure achievement of their secondary safety functions at all times including a concurrent safe shutdown earthquake (SSE).

25. Shutdown Logic Diagram

A logic diagram identifies safety related systems and safety functions and actions required for shutdown to safe conditions.

26. Single Active Component Failure

A single active failure is the failure of an active component to complete its intended function upon demand. The failure of an active component of a fluid system is considered to be a failure of the component to perform its function not the loss of structural integrity. The direct consequences of a single active failure are evaluated. (A single active failure is postulated to occur simultaneously with the pipe failure; passive failures are not postulated.)

27. Terminal Ends

Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, etc), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run may be considered as a terminal end of the branch run unless each of the following conditions are met:

a. That branch is modeled with the main piping run.

b. A rigorous ASME, Class 1, 2, or 3 analysis is conducted.

c. The nominal size of the branch line, in the vicinity of the branch connection, is greater than or equal to one-half the nominal size of the run.

28. Zone of Influence

The maximum physical range of the direct effects of pipe whip, jet impingement, and/or the environmental effects resulting from a pipe failure.

ASSUMPTIONS

In analyzing the effects of postulated piping failures, the following assumptions shall be made relative to plant and system operation before and after a pipe failure.

1. Operating Mode

All normal plant operating modes (see Definition 14) shall be investigated when evaluating the effects of a postulated pipe failure.

2. Single Active Component Failure

A single active failure is assumed in systems used to mitigate consequences of the postulated piping failure and to shutdown the reactor. The single active failure is assumed to occur in addition to and concurrent with the postulated piping failure and any consequences of the piping failure.

3. Available Systems

All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account shall be taken of the postulated failure and its consequences such as unit trip and loss of offsite power and of the assumed single active component failure and its consequences. The feasibility of carrying out operator actions shall be judged on the basis of ample time and adequate access to equipment being available for the proposed actions. No operator action is assumed to be initiated for at least 10 minutes after pipe failure.

4. Offsite Power

In general, if it is the worst case, offsite power shall be assumed to be unavailable during a portion of or throughout the sequence of events that follow a pipe failure. This loss of offsite power shall be assumed to act concurrently with the postulated pipe failure and the single active failure. If it can be shown that the loss of offsite power is not a consequence of the pipe failure, then a loss of offsite power is not assumed.

5. Unintended Operation of Equipment

The performance of an unintended active function by equipment not within the zone of influence of a pipe failure shall not be postulated. Unintended operation of equipment within the zone of influence of the pipe failure may occur if caused by the pipe failure, provided the unintended operation is a credible postulation. Unintended operation will not be considered to place equipment in any operating mode other than those modes for which it is normally required to function.

6. Operator Response

It shall be assumed that a proper sequence of events is initiated by the operator to bring the plant to a safe condition, with the capability of going to a cold shutdown if required. However, it shall be assumed that no operator action is initiated for at least 10 minutes after pipe failure. Additional time will be allocated for actions outside the main control room.

3.6A.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6A.1.1 Design Bases

3.6A.1.1.1 List of Potential Targets

Safety-related systems or components that are located proximate to and are susceptible to the consequences of failures of piping systems are discussed in Section 3.11.

3.6A.1.1.2 Interaction Criteria

The following criteria define how interactions are evaluated:

1. Pipe Whip Interaction

A whipping pipe is not considered to inflict unacceptable damage to other pipes and associated supports of equal or greater size and wall thickness. A whipping pipe is considered capable of only developing through-wall leakage cracks in other pipes of equal or greater size with smaller wall thickness.

Any active component (electrical, mechanical, and instrumentation and control) shall be assumed incapable of performing its active function following impact by any whipping pipe unless an analysis or test is conducted to show otherwise. Active components in pipe lines which are allowed to whip are assumed to be incapable of performing their active functions unless the line is sufficiently restrained to control the motion of the components to limits for which they have been qualified.

Structural components shall be assumed to fail upon experiencing pipe impact loads that exceed the allowable limits. Plastic action of steel, yield line methods etc., may be used to determine the allowable limits where applicable.

2. Jet Impingement Interactions

Jet impingement force from a pipe is not considered to inflict unacceptable damage to other pipes and associated supports of equal or greater size and wall thickness. The jet impingement force is considered capable of only developing through-wall leakage cracks in other pipes of equal or greater size with smaller wall thickness.

Active components (electrical, mechanical, and instrumentation and control) shall be assumed incapable of performing their function when subjected to a jet unless the active component is enclosed in a qualified spray-proof enclosure (such as one qualified to the NEMA IV, Hosedown Test Standard), the component is known to be insensitive to such an environment, or unless justified that the active function will not be impaired.

When the jet consists of steam or subcooled liquid that flashes at the break, unprotected components located at a distance greater than 10 diameters (ID) from the break or equivalent diameter of the crack shall be assumed undamaged by the jet without further analysis. The basis for this criterion is contained in Reference [5].

Concrete erosion that may result from jet impingement shall be assumed to be of insufficient magnitude to jeopardize structural integrity.

3. Environmental Interaction

An active component (electrical, mechanical, and instrumentation and control) shall be assumed incapable of performing its active function upon experiencing environmental conditions exceeding any of its environmental ratings. However, credit for the component may be taken if sufficient time is available for accomplishing its function before environmental ratings are exceeded.

3.6A.1.1.3 Acceptability Criteria

1. Systems

   The capability to eventually achieve a cold shutdown condition shall not be jeopardized even if the pipe failure is followed by a single active failure. The system requirements and available redundancy shall be that shown on a shutdown logic diagram, as supplemented by current system descriptions and equipment lists, for mitigating the effects of the postulated failure.

   Repair of failures may be considered to assure achievement of the cold shutdown condition where such repairs can be shown to be practicable and timely, and provided the unit can be held in a safe state during the time required for the repair.

2. Protective Structures

   The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, should not preclude habitability of the control room or access to surrounding areas required for safe control of reactor operations that are needed to cope with the consequences of the piping failure.

   For piping systems that are enclosed in suitably designed structures or compartments to protect other structures, systems, and components important to safety, pipe breaks shall be postulated according to section 3.6A.2 and the resulting jet thrust loading effects determined. "Worst case" breaks may be postulated in a piping component within the protective structure or compartment at locations which result in the maximum loading from the impact of the postulated ruptured pipe and jet discharge force on each wall, floor, and roof of the structure or compartment, including internal pressurization.

3.6A.1.1.4 Protective Measures

Where physical separation of source and target and relocation or rerouting are not feasible, the following protective devices will be provided to mitigate the unacceptable consequences of the postulated ruptures.

1. Pipe Whip Restraints: An engineered structure which permits limited pipe motion and rotation but limits or prevents unrestricted pipe whip. Crushable material may be used with certain restraints to absorb the kinetic energy of the ruptured pipe, and to limit the loads on the restraint structure.

2. Jet Deflector: A barrier which shields a target from the forces and environmental conditions within a jet.

3. Impact Barrier: An engineered structure located to limit pipe motion and designed to withstand the impact of a whipping pipe.

4. Pipe Sleeve: A metal sleeve that encloses a portion of a process pipe and is designed to restrict and redirect jet forces.

Welding for protective structures designed to the requirements of AISC (see Section 3.8.1.2, Item 2) was in accordance with the American Welding Society, "Structural Welding Code," AWS D1.1 (see Section 3.8.1.2, Item 4). Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 (see Section 3.8.1.2, Item 12) may be used after June 26, 1985, to evaluate weldments that were designed and fabricated to the requirements of AISC/AWS.

3.6A.1.2 Description of Piping System Arrangement

Separation was the primary consideration in the piping system layout and arrangement. Where physical separation is not feasible, protective devices shall be provided as required. Protection shall be provided such that the environmental design limits of mechanical and electrical equipment required for safe shutdown are not exceeded. Habitability is discussed in Section 6.4.

3.6A.1.3 Safety Evaluation

Safety functions shall be identified for initiating event by means of shutdown logic diagrams (SLD). The SLD shall identify at least one success path from each postulated event to each protective function required to prevent the event's potentially unacceptable results. Each SLD shall include the set of all safety systems necessary to provide the protective function specified at the end of the success path. Shutdown logic diagrams may be supplemented by current system descriptions and equipment lists.

For each postulated pipe rupture, credible unacceptable interaction shall be evaluated.

Possible interactions shall be evaluated to determine their credibility, damage potential, and acceptability from the standpoint of a safe shutdown capability.

In establishing system requirements for each postulated break, it is assumed that a single active component failure occurs concurrently with the postulated rupture.

3.6A.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6A.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6A.2.1.1 Pipe Failure Type, Size, and Orientation

1. Circumferential Rupture

   The break area is equal to the effective cross-sectional flow area of the pipe at the break location. The plane of the break is normal to the pipe flow axis. Flow may be out of each of the broken ends (double ended rupture) of the pipe, depending upon reverse flow capability.

This break is applicable to high energy piping and branch runs whose diameter is greater than 1-inch nominal pipe size. Circumferential ruptures are assumed to result in a lateral offset of one pipe diameter unless mitigating devices, structure members, or the inherent pipe stiffness can be specifically shown to limit this offset.

2. Longitudinal Split

The break area is assumed to be equal to the effective pipe cross-sectional flow area at the break location. If the break occurs at a transition from a smaller pipe to a larger pipe, the flow area is defined as one-half the sum of the upstream and downstream cross-sectional flow areas. The length of the break is two pipe inside diameters and is parallel with the pipe flow axis. As an alternate analysis procedure, fluid flow may be assumed to be from a circular opening equal to the effective cross-sectional flow area of the pipe. In the absence of a detailed analysis, the break is assumed at any location around the circumference of the pipe. Alternatively, a single split may be assumed at the point on the circumference of highest tensile stress as determined by a detailed stress analysis. This break is applicable to high energy pipe that has a nominal pipe size of 4 inches or larger.

3. Through-Wall Leakage Crack

The crack area may be based on a circular opening with an area equal to an equivalent rectangular opening of one-half the piping inside diameter in length and one-half the wall thickness in width and can be oriented in any direction.

3.6A.2.1.2 Break Location

1. High Energy Fluid System

A. ASME Section III Class 1 Piping Runs

Circumferential ruptures and longitudinal splits, in accordance with Sections 3.6A.2.1.1 (Item 1) and 3.6A.2.1.1 (Item 2), are postulated to occur at the following locations in ASME Section III Class 1 piping:

1. The terminal ends of piping or branch runs (circumferential ruptures only).

2. At intermediate locations per either one of the following [method a or method b]:

a. At each location of potential high stress and fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, and flanges, or

b. At all locations where either one of the following are met.

1) $S_n < 2.4\ S_m$* (Equation 10) and U > 0.l (U calculated according to NB-3653.5); or

2) $S_n > 2.4\ S_m$* (Equation 10) and $S_e > 2.4\ S_m$ (Equation 12), or

$S > 2.4\ S_m$ (Equation 13),

or U > 0.l (U calculated according to NB-3653.6)

*For stress qualification to Summer 1973 Code, use 3.0 S_m
For stress qualification to Winter 1982 Code, use 2.4 S_m

Where:

S_n = primary-plus-secondary stress-intensity range, as calculated from Equation 10 in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III for normal and upset plant condition loads with the upset plant condition loads defined as: sustained loads + all system operating transients associated with upset condition + OBE.

S_m = allowable design stress-intensity value, as defined in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

U = the cumulative usage factor, as calculated in accordance with Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_e = Nominal value of expansion stress as defined in equation (12) of NB-3653.6 of ASME Code, Section III.

S = The range of primary plus secondary membrane plus bending stress intensity as defined in equation (13) of NB-3653.6 of ASME Code, Section III.

Longitudinal splits need not be postulated in Class 1 piping at terminal ends or branch connections.

Through-wall leakage cracks are postulated in all high energy pipe outside containment whose diameter is greater than one inch nominal pipe size. Through-wall leakage cracks are not postulated in high energy piping inside containment whose diameter is greater than 1 inch nominal pipe size. However, through-wall leakage cracks are postulated in the main steam and feedwater lines inside containment where impingement could occur on the ice condenser doors. Also, through-wall leakage cracks, may be postulated in other high-energy lines in particularly susceptible areas.

B. Break Locations in ASME Section III Class 2 and 3 Piping Runs

Circumferential ruptures and longitudinal splits, in accordance with Sections 3.6A.2.1.1 (Item 1) and 3.6A.2.1.1 (Item 2), are postulated to occur at the following locations in ASME Section III Class 2 and 3 piping:

1. The terminal ends of piping or branch runs (circumferential ruptures only).

2. At intermediate locations selected by either one of the following method a) or method b):

   a. At each location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves and flanges, or

   b. At all locations where the stress, S, exceeds 0.8 (1.2 S_h + S_a)

where:

S = stresses under the combination of loadings associated with the normal and upset plant condition plus OBE loadings, as calculated from the sum of Equations (9) and (10) in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_h and S_a = Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, for Class 2 and 3 piping as defined in Subarticle NC-3600 of ASME Code, Section III.

Through-wall leakage cracks are postulated as indicated in Section 3.6A.2.1.2 (Item 1A).

C. Exceptions for Longitudinal Splits and Circumferential Ruptures

The following exceptions are applicable to high-energy Class 1, 2 and 3 piping and to high-energy non-safety class piping for which a Class 2 or 3 analysis is conducted.

1. Longitudinal splits need not be postulated at terminal ends or branch connections.

2. When values defined in 3.6A.2.1.2 are exceeded for Class 1 piping or the stresses exceed 0.8 (1.2 S_h + S_a), for Class 2 and 3 piping longitudinal splits need not be postulated if the stress in the axial direction is greater than or equal to 1.5 times the stress in the circumferential direction; and circumferential ruptures need not be postulated if the stress in the circumferential direction is greater than or equal to 1.5 times the stress in the axial direction.

D. High Energy Non-Safety-Class and Field Routed Fluid Systems

Circumferential ruptures and longitudinal splits in high energy non-safety-class and high energy field routed piping components are postulated to occur at terminal ends and at intermediate pipe fittings, flanges, and valves. Through-wall leakage cracks are postulated as indicated in Section 3.6A.2.1.2, Item 1A.

Break locations in high energy non-safety-class systems, which are analyzed to the same requirements as Class 2 or 3 piping, (these cases will be fully coordinated and documented) may be postulated according to the requirements of Section 3.6A.2.1.2, Item 1B.

2. Moderate Energy Fluid Systems

Circumferential ruptures and longitudinal splits are not postulated in any moderate energy lines. Through-wall leakage cracks are postulated in moderate energy piping which exceed a nominal pipe size of 1 inch, but may be excluded where either of the following rules apply.

A. Piping systems are located in areas containing systems and/or components important to safety enveloped by previously postulated high energy breaks in the same region.

B. Where the maximum stress, S, as defined in Section 3.6.A.2.1.2 (Item 1B) is less than or equal to 0.4 (1.2 S_h + S_a) for Class 2 and 3 piping or where S_n by equation 10 is less than or equal to 1.2 S_m for Class 1 piping.

The cracks should be postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding. It shall be at any location on the pipe circumference or along the surface of the pipe.

3. High/Moderate Energy Interfaces

Line supported valves sometimes form the interface between high energy lines and moderate energy lines. In this case, the fixity as implied in the word, 'terminal,' does not exist at the line supported valve. This condition is treated as if there were no terminal.

3.6A.2.1.3 Failure Consequences

The failure interactions that must be evaluated to determine the consequences of failure are dependent upon the energy level of the pipe considered. They are as follows:

1. High Energy Piping

    Circumferential ruptures and longitudinal splits

    a. Pipe whip.
    b. Jet impingement.
    c. Environmental effects.

    Through-wall leakage cracks

    a. Jet impingement
    b. Environmental effects

2. Moderate Energy Piping

    Through-wall leakage cracks

    a. Environmental effects.

In particularly susceptible areas, the jet impingement load associated with a through-wall leakage crack in moderate energy piping with the pressure exceeding 275 psig shall also be considered.

3.6A.2.1.4 Flooding

Flooding consequences are also considered in addition to the local effects listed above in Section 3.6A.2.1.3 from piping failures. Additional environmental concerns are addressed in Section 3.11.2.

1. High Energy Line Breaks (HELBs)

    For the purposes of flooding evaluations, fluid systems that, during normal plant conditions are either in operation or maintained pressurized under conditions where maximum operating temperature exceeds 200°F are conservatively classified as high energy. This is bounding since for a given line, the flow from a high energy break emanates from a larger break area than flow from a moderate energy crack. The circumferential rupture is the bounding break for HELB flooding analyses.

    Systems classified as high energy are re-classified as moderate energy if the total time that the above conditions are exceeded is less than either of the following:

    a. 1% of the normal operating life span of the plant, or
    b. 2% of the time required for the system to accomplish its system design function.

    The systems evaluated for high energy break flooding include the reactor coolant, main steam, feedwater, auxiliary boiler, auxiliary feedwater steam supply, and chemical and volume control system.

2. Moderate Energy Line Breaks (MELBs)

For the purposes of flooding evaluations, fluid systems are classified as moderate energy that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where: (1) The maximum operating temperature is 200°F or less or (2) the 1 or 2% exclusion rules described above are applicable. The through-wall leakage crack is the postulated break for the MELB flooding analysis. Flood levels are calculated for the plant on an area basis. Both submergence and structural loading are addressed in the flooding studies.

HELB and MELB flooding effects are evaluated on all essential equipment on a case by case basis. If it is determined that an essential component is not qualified or cannot be demonstrated to operate under the adverse flood conditions, then the essential component is protected. Protection is accomplished by relocating the component or by installing a barrier or curb. Safe shutdown is ensured for design basis HELB/MELB flooding events through these actions.

3.6A.2.1.5 Leak-Before-Break Application

The application of leak-before-break as applied to the primary loop piping is discussed in Section 3.6B.1.

In addition, leak-before-break technology has been applied to the pressurizer surge line to eliminate the dynamic effects of a pressurizer surge line rupture as a design basis for Watts Bar Nuclear Plant. This is in accordance with the final rule change to General Design Criteria 4.[12] Authorization for their elimination is discussed in Reference [9] and is based on fracture mechanics results presented in References [10] and [11].

3.6A.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6A.2.2.1 Assumptions

1. The thrust load acting on the pipe due to a blowdown jet is equal and opposite to the jet.
2. The discharge coefficient is equal to 1.0.
3. The break opens to its defined size in 1 millisecond.
4. For the purpose of estimating jet forces, the blowdown is to an infinite volume at standard conditions.
5. The initial fluid condition within the pipe prior to rupture is that for normal plant operating condition.
6. The jet profile expansion half angle is 20 degrees.

3.6A.2.2.2 Blowdown Thrust Loads

The thrust force at any time, T (t) is given by

$$T(t) = \left(\frac{\rho_E V_E^2}{g_c} + [P_E - P_A] \right) A_{jE}$$

where:

ρ_E = fluid density at break at time t

V_E = fluid velocity at break at time t
A_{jE} = pipe break exit area
P_E = control volume pressure at break at time t
P_A = ambient pressure
g_C = gravitation constant

A simplified analysis may be conducted by assuming that the fluid is blowing down in a steady-state condition with frictionless flow from a reservoir at fixed absolute pressure Po. (Po is the initial line pressure.) When the fluid is subcooled, nonflashing liquid, the flow will not be critical at the break area so that:

$$P_E = P_A$$

and

$$V_E = \sqrt{2g_c\left(P_O - P_A\right)/\rho_E}$$

If $P_A \ll P_o$ the thrust force may be conservatively approximated by:

$$T = 2P_0A_{jE}$$

When the fluid is saturated, flashing or super-heated vapor, the fluid can be assumed to be a perfect gas. The velocity for critical flow at the break area is given by:

$$V_E = \sqrt{Kg_c\,P_E/\rho_E}$$

And

$$P_E = P_O\left[\frac{2}{K+1}\right]^{\frac{k}{K-1}}$$

where

$K = C_p/C_v$ is a ratio of specific heats
C_p = specific heat at constant pressure
C_v = specific heat at constant volume

A value of K = 1.3 is justified for steam as being conservative. If $P_E \gg P_A$, the thrust force may be conservatively approximated by:

$$T = 1.26 \ P_o \ A_{jE}$$

3.6A.2.2.3 Jet Impingement Loads

The loads on an object exposed to the jet from a pipe break can be determined from the blowdown thrust and the profile of the impinged object.
where

$$Y_j = T \cdot \frac{A_i}{A_j} \cdot S_F \cdot D_{LF} \cos\phi$$

Y_j =Normal load applied to a target by the jet

A_i =Cross-sectional area of jet intercepted by target structure

A_j =Total cross-sectional area of jet at the target structure

S_F =Shape factor

D_{LF} =Dynamic load factor

T =Total blowdown thrust at break as calculated in Section 3.6A.2.2.2

ϕ =Angle between jet axis and a line perpendicular to the target.

The ratio A_i/A_j represents the proportion of the total mass flow from the jet which is intercepted by target structure. A dynamic load factor of 2.0 shall be used in the absence of an analysis justifying a lower value. The following shape factors are recommended.

Jet impinging on a slab [Figure 3.6-1 sector (a)]

$$S_F = 1$$

Rectangular jet impinging on a pipe larger than jet [Figure 3.6-1 sector (b)]

$$S_F = 1 - \frac{h}{2D_o}$$

Rectangular jet impinging on a pipe with h greater than D_o [Figure 3.6-1 sector (b)]

$$S_F = ½$$

Circular jet impinging on pipe with jet diameter ($D_j = 2r_j$) less than pipe diameter [Figure 3.6-1 sector (c)]

$$S_F = 1 - 0.424 \frac{D_j}{D_o}$$

Circular jet impinging on pipe with jet diameter greater than pipe diameter [Figure 3.6-1 sector (d)]

$$S_F = 0.576$$

These are the most common cases that will occur in the pipe rupture evaluation. Other shape factors may be obtained by idealizing the surface as infinitesimal planes and performing an integration over the area impinged upon by the jet.

3.6A.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6A.2.3.1 General Criteria for Pipe Whip Evaluation

1. The dynamic nature of the piping thrust load shall be considered. In the absence of analytical justification to the contrary, a dynamic load factor of 2.0 may be applied in determining piping system response.

2. Nonlinear (elastic-plastic strain hardening) pipe and restraint material properties may be considered as applicable.

3. Pipe whip shall be considered to result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the vector thrust of the break force. A maximum of 180° rotation may take place about any hinge.

4. The effect of rapid strain rate of material properties may be considered. A 10% increase in yield strength may be used to account for strain rate effects.

3.6A.2.3.2 Main Reactor Coolant Loop Piping System

The dynamic analyses applicable to the reactor coolant loop piping are discussed in Section 3.6B.

3.6A.2.3.3 Other Piping Systems

The pressure time history, jet impingement load on targets, and the thrust resulting from the blowdown of postulated ruptures in piping systems shall be determined by thermal and hydraulic analyses or a conservative simplified analyses.

In general, the loading that may result from a break in piping will be determined using either a dynamic blowdown or a conservative static blowdown analysis. The method for analyzing the interaction effects of a whipping pipe with a restraint will be one of the following:

1. Equivalent static method

2. Lumped parameter method

3. Energy balance method.

In the cases where time history or energy balance method is not used, a conservative static analyses model will be assumed. The loading factors to be used for the static model are discussed in Section 3.6A.2.3.5.

The lumped parameter method is carried out by utilizing a lumped mass model. Lumped mass points are interconnected by springs to take into account inertia and stiffness properties of the system. A dynamic forcing function or equivalent static loads may be applied at each hypothesized break point with unacceptable pipe whip interactions. Clearances and inelastic effects will be considered in the analyses.

The energy balance method is based on the principle of conservation of energy. The kinetic energy of the pipe generated during the first quarter cycle of movement will be assumed to be converted into equivalent strain energy, which will be distributed to the pipe or the support. The strain in the restraint shall be limited to 50% of the ultimate uniform strain.

3.6A.2.3.4 Simplified Pipe Whip Analysis

A conservative method may be used to determine for a given rupture whether pipe whip takes place. This method is based on calculation of the minimum internal forces necessary to form a plastic hinge in the pipe, and the number of hinges required for a pipe whip mechanism.

Occurrence of a pipe whip is dependent on formation of a sufficient number of hinges to develop a mechanism. Two commonly encountered examples are:

A. Cantilever pipe with end load

$$T_{whip} = \frac{M_{ult}}{L}$$

B. Continuous pipe supported at both ends with lateral load

$$\mathrm{T_{whip}} = 2\mathrm{M_{ult}}\left(\frac{1}{\mathrm{L_1}} + \frac{1}{\mathrm{L_2}}\right)$$

Where

L, L_1, L_2 = Distance from support to load.

M_{ult} = The ultimate moment

T_{whip} = The thrust load at which pipe whip will occur.

The applied thrust load shall consider a dynamic amplification factor of 2.0 unless an analysis is performed to justify a lesser value.

3.6A.2.3.5 Pipe Whip Restraint Design

The design limits which shall be used in the design of pipe whip restraints are shown in the following table:

Type of Design	Plastic	Elastic
Loading Combination	$D+L+T_a+P_a+Y_r+Y_j+Y_m$	$D+L+T_a+P_a+Y_r+Y_j+Y_m$
Stress/strain limits	50% uniform Ultimate strain	1.5 S_m or 1.2 S_y, but not to exceed 0.7 S_u

Note: Earthquake and pipe rupture are not assumed to exist concurrently when evaluating the pipe whip restraints.

Where:

D	=	Dead load
L	=	Live
T_a	=	Thermal load resulting from postulated break
P_a	=	Pressure load resulting from postulated break
Y_r	=	Pipe restraint reactions resulting from postulated break
Y_j	=	Jet impingement load generated by postulated break
Y_m	=	Pipe whip impact load resulting from postulated break
Sm	=	Design stress - intensity
Sy	=	Yield stress
Su	=	Ultimate tensile stress

Dynamic response amplification was accounted for by multiplication of loads by appropriate dynamic factors or through use of dynamic analysis. The following dynamic load factors were used for the local structure components design.

1. For piping system with no gaps at the restraint, a dynamic load factor of 2.0 was applied regardless of pipe size.

2. For piping system with gaps not exceeding 1 inch at the restraint, a dynamic load factor of 3.0 may be applied.

3. A linear interpolation for gaps between zero and 1 inch may be made. The above dynamic factors in items 1 and 2 are applicable to small line (6-inch nominal diameter or less) without subsequent analyses. Items 2 and 3 may also be applied to large lines (larger than 6-inch nominal diameter) providing sufficient analyses are performed to show that the dynamic factor has not been exceeded.

4. For gaps in excess of 1 inch, dynamic load factors shall be justified by analyses.

3.6A.2.3.6 Energy Absorbing Materials

An energy absorbing material (crushable honeycomb) is sometimes used to absorb the kinetic energy of the ruptured pipe and to limit the loads on the restraint structure. For systems where the energy balance method of analysis is used, the kinetic energy of the pipe generated during the first quarter cycle of movement will be assumed to be converted into equivalent strain energy, which will be distributed on the pipe or the support. The actual crush shall not exceed 90% of the available crush depth.

3.6A.2.4 Guard Pipe Assembly Design Criteria

Guard pipes for penetrations are classified as TVA Class K. The chemical and mechanical tests and nondestructive examinations shall be in accordance with the ASME Material Specification. Markings and certified mill tests shall be in accordance with the requirements for process pipe. All welding shall be made in accordance with ASME Code, Section III, NC-4000. All girth butt welds shall be magnetic particle or liquid penetrant inspected in accordance with Appendix IX of ASME Code, Section III. Acceptance standards shall be in accordance with NE-5000.

The guard pipe shall be designed for the same temperature and pressure as the process pipe. However, the allowable stresses shall be 90% of yield strength (0.2% offset) at design temperature.

The guard pipe shall be designed to have its lowest natural frequency greater than 33 Hz where possible to allow the zero period acceleration to be used. Where 33 Hz is not practical, the actual frequencies expanded by 10%, shall be used in conjunction with the appropriate floor response spectra, to determine the design acceleration. The seismic loading shall be that which results from input accelerations of 1.5 g horizontal and 1 g vertical for the operating basis earthquake and twice these values for the safe shutdown earthquake.

Inservice inspections and accessibility requirements are discussed in Section 5.2.8 for ASME Class 1 systems, Section 6.6 for ASME Class 2 and 3 systems, and Section 3.8.2.7.9 for ASME Class MC and metallic liners of Class CC components. Penetration assemblies to be used for piping penetrations of containment areas are discussed in Section 3.8.2.

If circumferential ruptures or longitudinal splits are postulated in the process pipe (in accordance with Section 3.6A.2.1.2) at locations enclosed by the guard pile, the guard pipe shall be capable of mitigating the consequences of the break. If no circumferential ruptures or longitudinal splits are postulated in the process pipe, arbitrary through-wall leakage cracks shall be assumed and the guard pipe shall be capable of mitigating the consequences of the cracks.

3.6A.2.5 Summary of Dynamic Analysis Results

A letter from J. E. Gilleland to Mr. Giambusso dated May 16, 1974, submitted CEB Report No. 72-22, "Evaluation of the Effects of Postulated Pipe Failures Outside of Containment for the Sequoyah Nuclear Plant Units 1 and 2." In this letter it was stated that this report is also applicable to the Watts Bar Nuclear Plant and upon completion of the Watts Bar piping outside the containment, any differences between the Sequoyah and Watts Bar designs would be addressed in the Watts Bar UFSAR. The major differences between the Watts Bar and Sequoyah designs outside containment are the main steam and feedwater routing in the open bay area of the Control Building.

3.6A.2.5.1 Stress Summary and Isometrics - Inside Containment

The stress summary for each of the postulated break locations for the following systems larger than 4 inches in nominal size are presented in Tables 3.6-1 through 3.6-6;

Table No.	System Description
3.6-1	Main Steam Lines
3.6-2	Main Feedwater Lines
3.6-3	Auxiliary Feedwater Steam Supply Lines - Unit 1
3.6-3A	Auxiliary Feedwater Steam Supply Lines - Unit 2
3.6-4	SI Cold Leg Injection
3.6-5	RHR/SI Hot Leg Recirculation, Loop 4
3.6-6	SI Hot Leg Recirculation, Loops 1,2, and 3

Isometrics showing break type locations, protective device locations and constrained directions of the above systems are presented in Figures 3.6-2 through 3.6-17.

Inside containment the isometrics for Unit 2 are generally opposite hand to Unit 1. The stress summaries and isometrics are based on the analysis current at the time of the amendment submittal indicated on the figures and tables. This information is considered representative and presents typical historical results for Units 1 and 2.

3.6A.2.5.2 Summary of Protection Requirements and Isometrics Outside Containment

A summary of protection requirements including break types and locations for main steam and main feedwater lines are presented in Tables 3.6-9 and 3.6-10, respectively. Isometrics showing break types, locations. protective device locations and constrained directions for these lines are shown in Figures 3.6-21 through 3.6-24.

Outside containment, the break types, break locations, isometrics, protective device locations and constrained directions for Unit 2 are generally opposite hand to Unit 1. The isometrics and protection requirements reflect the analysis current at the time of amendment submittal indicated on the figures and tables. This information is considered representative and presents typical historical results for Units 1 and 2.

APPENDIX 3.6B

PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6B.1 Break Locations And Dynamic Effects Associated With Postulated Primary Loop Pipe Rupture

The dynamic effects of postulated double-ended pipe ruptures in the reactor coolant loop piping have been eliminated from the design basis of the Watts Bar Nuclear Plant by the application of leak before break technology in accordance with the final rule change to General Design Criterion 4 (Reference 12). Authorization for their elimination is provided in Reference [6] and is based on fracture mechanics analysis results presented in References [7] and [8].

The plant design bases was revised in several areas to take advantage of the elimination of reactor coolant loop (RCL) pipe breaks. The protective measures taken to mitigate the dynamic effects of these breaks remain in place. However, these protective devices no longer perform a pipe whip restraint function. See UFSAR Section 5.5 and Figures 5.5-11, 5.5-12, and 5.5-13.

In other areas, design basis analyses have been conducted based on the original postulated double-ended breaks. Even with the elimination of these dynamic effects, these analyses continue to demonstrate the adequacy and acceptability of the plant design. These analyses shall remain the analyses of record unless indicated otherwise in this safety analysis report.

Leak-before-break has also been applied to the pressurizer surge line as discussed in Section 3.6A.2.1.5.

As stipulated in the final rule change to GDC-4, a non-mechanistic double-ended rupture of the largest pipe in the reactor coolant system is still postulated for the purposes of containment design, ECCS design, and environmental qualification of electrical and mechanical equipment.

Previously postulated breaks in branch lines (except the pressurizer surge line) attached to the reactor coolant loops remain unaffected.

3.6B.2 Analytical Methods to Define Forcing Function and Response Models

The reactor coolant loop breaks used in determining the forcing functions (discussed below) and in calculating the resulting hydraulic transients and loadings have been eliminated as noted in Section 3.6B.1. However, these analyses envelope the effects of any remaining breaks, e.g., in branch lines at the loop attachment points, and as such continue to demonstrate the adequacy of the design for these loadings.

Following is a summary of the methods used to determine the dynamic response of the reactor coolant loop associated with postulated pipe breaks in the loop piping. Detailed descriptions of the methods are given in Reference [1].

In order to determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated loss of coolant accident (LOCA), it is necessary to have a detailed description of the hydraulic transient.

Hydraulic forcing functions are calculated for the ruptured and intact reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the reactor coolant system. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates and calculates the time history of forces at appropriate locations in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire reactor coolant system. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and density. These are supplied to the thrust calculation, together with appropriate plant layout information to determine the time dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid through out the reactor system. The MULTIFLEX 3.0[17] computer code was developed with this capability, which is an enhancement and extension of MULTIFLEX 1.0[2], NRC reviewed and approved computer code developed for the same space-time dependent analysis of nuclear power plants. The MULTIFLEX 3.0 features which differ from MULTIFLEX 1.0 are primarily related to vessel forces. The loop forcing functions do not differ significantly from those generated using the NRC approved MULTIFLEX 1.0 model. MULTIFLEX 3.0 has been accepted by NRC for several applications[13],[14],[15],[16] and has been extensively used for the LOCA analyses of various 2, 3 and 4 loop nuclear plants.

MULTIFLEX is a digital computer program for calculation of pressure, velocity, and force transients in reactor primary coolant systems during the subcooled, transition, and the early saturation portion of blowdown caused by LOCA. During this phase of the accident, large amplitude rarefaction waves are propagated through the system with the velocity of sound causing large differences in local pressures. As local pressures drop below saturation, causing formation of steam, the amplitudes and velocities of these waves drastically decrease. Therefore, the largest forces across the loop piping due to wave propagation occur during the subcooled portions of the blowdown transient. MULTIFLEX includes mechanical structure models and their interaction with the thermal-hydraulic system, although these features are only involved in the vessel and steam generator modeling.

The THRUST computer program was developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA.

The blowdown hydraulic loads on primary loop components are computed from the equation.

$$F = 144A\left[(P-14.7) + \frac{\dot{m}^2}{\rho_g A_m^2 \ 144}\right]$$

Which includes both the static and dynamic effects. The symbols and units are:

- F = Force, lb_f
- A = Actual calculated break flow area, ft^2
- P = System pressure, psia
- $\dot{m}$ = Mass flow rate, lb_m/sec
- ρ = Density, lb_m/ft^3
- g = Gravitational constant = 32.174 ft/sec^2
- A_m = Mass flow area, ft^2

In the model to compute forcing functions, the reactor coolant loop system is represented by a similar model as employed in the blowdown analysis. The entire loop layout is described in a coordinate system. Each node is fully described by: 1) blowdown hydraulic information, and 2) the orientation of the streamlines of the force nodes in the system, which includes flow areas, and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The THRUST Code is described in Reference [3].

3.6B.3 Dynamic Analysis of the Reactor Coolant Loop Piping Equipment Supports and Pipe Whip Restraints

The dynamic analysis of the reactor coolant loop piping for the LOCA loadings is described in Section 5.2.1.10.

Section 5.2 defines the loading combinations, associated with the reactor piping systems, considered to assure the integrity of vital components and engineered safety features.

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3. 'Documentation of Selected Westinghouse Structural Analysis Computer Codes,' WCAP-8252, Revision 1, May 1977.

4. 'Relaxation in Arbitrary Intermediate Pipe Rupture Requirements', NRC Generic Letter 87-11, June 19, 1987.

5. 'Two Phase Jet Loads' NUREG/CR-2913, January 1983.

6. NRC Letter to TVA, dated May 17, 1990, “Safety Evaluation of Primary Loop Piping.”

7. TVA Letter to NRC dated April 17, 1989, "Elimination of Primary Loop Pipe Breaks."

8. "Technical Justification For Eliminating Large Primary Loop Pipe Rupture As The Structural Design Basis For Watts Bar Units 1 & 2", WCAP- 11984 (Non-Proprietary) and WCAP-11985 (Proprietary), November 1988.

9. NRC letter to TVA, dated April 28, 1993, "Leak-Before-Break Evaluation of the Pressurizer Surge Line".

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15. WCAP-11004-P/WCAP-11005 (NP), “Comparison of DATA for Beaver Valley Power Station Unit 2 with WCAP-9735 Data, Prepared for NRC Review in Conjunction with Review of WCAP-9735, Docket No. 50-412,” D. R. Bhandari, K. Takeuchi, M. E. Wills, November 1985.

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17. K. Takeuchi, et al., “MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model,” WCAP-9735, Revision 2, Proprietary, and WCAP-9736, Revision 1, Non-Proprietary, February 1998.

3.7 SEISMIC DESIGN

The original analyses of Category I structures were performed using methodologies that were prevalent prior to issuance of the Standard Review Plan (SRP) (NUREG-0800, Rev. 1). Throughout this section, the bases for these analyses are called the "Original Seismic Analysis Criteria" and analysis results (Amplified Response Spectra (ARS), forces, displacements, etc.) using these criteria are termed Set A. The plant's design basis is Set A criteria.

As a result of various seismic analysis issues identified during 1987-1989, reanalysis of some structures was necessary. The intent of the reanalysis was to demonstrate, by addressing these issues, the seismic design adequacy of structures, systems and components. Evaluations of the adequacy of existing hardware are based on SRP compatible criteria and current practices. This criteria, called the "Evaluation Seismic Analysis Criteria," includes the Site Specific Response Spectra (SSRS) developed for WBN, three-dimensional seismic models, and SRP compatible damping values. Evaluation criteria analysis results are termed Set B criteria.

In order to develop seismic input for future designs and modifications of existing designs, the Category I structures analyzed for Set B criteria were also reanalyzed using the original criteria with current modeling techniques, including soil-structure interaction. These analyses results are termed Set C.

The SRP 1981, Revision 1 formed the basis for Set B and Set C analyses, updated to the provisions of SRP, 1989, Revision 2. Specific evaluations were performed for the following:

a. The requirement of varying the soil shear modulus by +100%, -50% from the best-estimate (mean), and the best estimate soil shear modulus.

b. The limitation of hysteretic soil damping ratio to the maximum of 15%.

The seismic responses (ARS, accelerations, displacements, forces, and moments) defined by the envelope of Set B and Set C (Set B+C) are for use in new designs and modifications. New designs and modifications initiated after October 1, 1989, are based on Set B+C responses.

Underground electrical conduit banks were evaluated using Set B criteria. Conduit banks were reevaluated because the original seismic analysis was not retrievable, and the design criteria had been revised to incorporate the design requirement to consider axial loads in the analysis of conduit banks. Set B and Set C analysis were not performed for the Waste Packaging Area, (WPA), and Condensate Demineralizer Waste Evaporator Structure, (CDWE), since these two structures do not house any safety-related systems and components. Furthermore, Set B and Set C analyses were not performed for the essential raw cooling water system (ERCW) retaining walls, miscellaneous yard structures and Class 1E electrical system manholes and handholes because the seismic design input for these features is the ground motion; thus, the generation of ARS are not necessary, and there are no outstanding issues which necessitate a reevaluation. If a reevaluation of such features to resolve CAQ's etc., is required, Set B ground motion will be used in the reevaluation.

3.7.1 Seismic Input

3.7.1.1 Ground Response Spectra

Vibratory ground motions are defined by two sets of site seismic design response spectra: The Modified Newmark Ground Response Spectra or Original Site Design Response Spectra for Set A and Set C analyses and the Site Specific Ground Response Spectra for Set B (Evaluation) analyses.

3.7.1.1.1 Original Site Ground Response Spectra (Set A and Set C)

The original site seismic design response spectra which define the vibratory ground motion of the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) for rock-supported structures are shown in Figures 2.5-236a and 2.5-236b. The maximum rock acceleration for the SSE is 0.18g for horizontal motion and 0.12g for vertical motion. The OBE is equal to one-half the SSE, as outlined in Section 2.5.2.7 (historical information), with maximum horizontal and vertical rock accelerations of 0.09g and 0.06g, respectively.

3.7.1.1.2 Site Specific Ground Response Spectra (Set B)

Seismic input motions for the evaluation of existing structures, systems, and components are defined by the top-of-rock SSRS shown in Figures 3.7-4a through 3.7-4r. Peak SSE and OBE top-of-rock accelerations are 0.215g (horizontal SSE), 0.15g (vertical SSE), 0.09g (horizontal OBE), and 0.06g (vertical OBE).

3.7.1.2 Design Time Histories

3.7.1.2.1 Time Histories for Original Site Ground Response Spectra (Set A and Set C)

For time history analyses, four artificial acceleration time histories were developed so that the response spectra produced by the arithmetic average of the response spectra of each individual record envelope the site seismic design response spectra. Figures 3.7-1 through 3.7-4 show the comparison, for the various damping ratios, of these averaged response spectra and the site seismic design response spectra for the SSE. Table 3.7-1 lists the system period intervals at which the response spectra are calculated.

3.7.1.2.2 Time Histories for Site Specific Ground Response Spectra (Set B)

Set B analyses utilize three statistically independent acceleration time histories. The response spectra for these three statistically independent time histories are shown in Figures 3.7-4a through 3.7-4r. These time histories satisfy the SRP design spectra enveloping requirements.

The power spectral density function (PSDF) enveloping criteria of NUREG/CR-5347 were used to ensure adequate energy content of the artificial time histories. The PSDF enveloping criteria are that the PSDFs of artificial time histories whose response spectra envelope the 84th-percentile target response spectra should generally envelope the "minimum required" target PSDF for the corresponding non-exceedance probability level to ensure adequate motion energy contents of artificial time histories. The minimum required target PSDF is defined as the 80% of the target PSDF. The minimum required horizontal and vertical 84th-percentile target PSDFs for the Watts Bar site-specific ground motions were calculated and compared with the corresponding PSDFs of the artificial time histories as shown in Figures 3.7-4s, 3.7-4t, and 3.7-4u.

As can be seen from Figures 3.7-4s and 3.7-4t, the PSDFs of the horizontal artificial time histories envelope the corresponding minimum required 84th-percentile target PSDFs in the frequency range of 0.7 cps to 25 cps. The PSDF of the artificial time history H2 dip slightly below the horizontal minimum required 84th-percentile target PSDF in the small frequency range of 0.5 cps to 0.7 cps. This slight dip is considered inconsequential because the response spectral values of H2 time history envelope the site-specific response spectra in this frequency range and no structural frequencies of Category I structures exist in this low frequency range. Thus, the horizontal SSRS-compatible artificial time histories have adequate motion energy contents and their PSDFs satisfy the PSDF enveloping criteria proposed in NUREG/CR-5347 in the frequency range of 0.7 cps to 25 cps.

Similarly, as can be seen from Figure 3.7-4u, the PSDF of the vertical artificial time history envelope the corresponding minimum required, 84th-percentile target PSDF in the frequency range from 1.6 to 25 cps. The PSDF of the artificial time history has very slight dips below the vertical minimum required 84th-percentile target PSDF in the small frequency ranges of 0.40 to 0.42 cps, and 1.2 to 1.6 cps. These slight dips are considered inconsequential because the response spectral values of the vertical time history envelope the site-specific response spectra in these frequency ranges and no structural frequencies of Category I structure exist in this low frequency range. Thus, the vertical SSRS-compatible artificial time history has adequate motion energy contents and its PSDF satisfy the PSDF enveloping criteria proposed in NUREG/CR-5347 in the frequency range of 1.6 to 25 cps.

3.7.1.3 Critical Damping Values

The specific percentages of critical damping values used for Category I structures, systems, and components are provided in Table 3.7-2 for Sets A, B, and C.

3.7.1.4 Supporting Media for Seismic Category I Structures

A complete description of the supporting media for each Seismic Category I structure is provided in Section 2.5.4 (historical information). Pertinent data concerning the supporting media for Set A, B, and C analyses of each Seismic Category I structure is also given in Table 3.7-3.

3.7.2 Seismic System Analysis

This section describes the seismic analysis performed for Category I structures.

3.7.2.1 Seismic Analysis Methods

The seismic methods of analysis used for the Category I structures listed in Section 3.2.1 are described in the following sections.

3.7.2.1.1 Category I Rock-Supported Structures - Original Analyses (Set A)

The seismic analyses of Category I structures were based upon dynamic analyses using the lumped mass normal mode method with idealized mathematical models. The inertial properties of the models were characterized by the mass, eccentricity, and mass moment of inertia of each mass point. Mass points were located at floor slabs, changes in geometry, and at intermediate points to adequately model the structure. The stiffness properties were characterized by the moment of inertia, area, shear shape factor, torsion constant, Young's modulus, and shear modulus. Significant modes of vibration were considered in determining the total response. For structures with built-in asymmetry and open structures which have low torsional resistance, coupled translation and torsion were included in the dynamic analyses. Torsional effects for the closed structures with small eccentricities have insignificant effect on the responses. To demonstrate this, a dynamic analysis study of the steel containment vessel, including an accidental eccentricity of 5% of the diameter, showed that the induced torsion had a negligible effect on the acceleration response spectra. Structural response was calculated in both the east-west and north-south directions except where symmetry justifies analyses in one direction. The effect of the vertical component of earthquake motions on the structural response was included.

For structures surrounded by soil, the effect of the soil stiffness on the structural response was determined by replacing the soil with springs of equivalent stiffness. Due to seismic motion, the soil pressure against structures was increased above the static soil pressure. The magnitude of this increase was determined by using the shaking table experiments performed for the design of TVA's Kentucky Hydro Project.[1] For a ground acceleration of 0.18g the static soil pressure was increased by 46% for a dry fill and 22% for a saturated fill. This incremental increase was combined with the static pressure as a triangle of pressure whose apex is at the rock surface and maximum ordinate is at the ground surface. In addition to the soil pressure increase as described above for a saturated fill, the hydrostatic pressure of water within the fill was increased 22%. This incremental increase was combined with the static water pressure as a triangle of pressure whose apex is at the water surface and maximum ordinate is at the rock surface or bottom of structure. Calculations using the shaking table experiment results have been confirmed using information in Reference [2]. A more detailed description of the seismic analyses of Category I rock-supported structures is discussed below.

The in situ measured shear wave velocity of the bedrock upon which the structures are founded has an average value of 5,900 feet per second (Section 2.5.4.8, historical information). Therefore, the effect of structure-foundation interaction was investigated for the major structures. The results of the investigation are discussed below as one of the parameters associated with the analysis of those structures.

The structural response was computed using the response spectrum modal analysis method. The techniques used to account for the three components of motion and the method of combining modal responses when computing the structural response of a structure are explained in Sections 3.7.2.6 and 3.7.2.7, respectively.

Response spectra were produced by the time history-modal analysis method using the four artificial accelerograms discussed in Section 3.7.1.2 and the techniques of Sections 3.7.2.5 and 3.7.2.9.

When torsion is considered, accelerations and deflections were calculated at the farthest points on the structure from the shear center, on the axis perpendicular to the direction of motion. The moment and shear due to earthquake motion were used in combination with other appropriate loads to determine overturning moments.

The response was calculated for both the OBE and the SSE, except when the same percentages of critical structural damping were specified for both earthquake levels, in which case the response was calculated for the OBE only (the SSE results are twice the OBE results). For applicable stress criteria, see Section 3.8.

The damping ratios used in the dynamic analyses of the structures are given in Table 3.7-2.

To ensure that the results of the seismic analysis of the structures were used in the design, the analyses became part of the nuclear plant design criteria and were submitted to the design sections responsible for design and to the principal engineer. For more detailed procedures and criteria of design control measures, see Section 3.8.

Shield Building

Two separate, distinct analyses were performed on the reinforced concrete structure to determine the response of the structure to horizontal motion when modeled as a cantilever beam and the response of the dome to vertical motion when modeled as a shell.

The Watts Bar Shield Building is identical to the Shield Building at TVA's Sequoyah Nuclear Plant. The building has been assumed to have identical structural properties in both the east-west and north-south directions. A sketch of the lumped mass model is shown in Figure 3.7-5 and the structural properties are listed in Table 3.7-4. The dome was considered a rigid body and its weight added at mass point 25. The dynamic analyses in both the horizontal and vertical directions was done by the normal mode response spectrum method. Although no structural eccentricities exist in the building, an accidental eccentricity of 5% of the diameter was assumed in the design. Periods for the normal modes of vibration are listed in Table 3.7-5.

Since torsion is considered, the maximum structural accelerations and deflections will not occur at the center of mass but rather at the point on the structure farthest from the shear center. For the Shield Building the shear center is located at the geometric center. Accordingly, all structural accelerations and deflections as well as the floor response spectra have been computed at a point located on the shell wall. Structural responses were calculated for both the OBE and SSE using structural damping of 2 and 5 percent, respectively.

Foundation-structure interaction studies were performed to determine the response characteristics of the Shield Building steel containment-interior concrete system to rocking-type motion. These analyses were performed considering lumped-mass models of the structure coupled with foundation springs. These springs were calculated as detailed by Whitman.[3] The results of these investigations indicated that the Shield Building accelerations would increase by less than 15% compared to the accelerations of a rigid base, single structure system. As a result, all spectra used to compute structural response and all accelerograms used to compute floor response spectra were multiplied by a factor of 1.15. The site response spectra for structures without rocking have previously been shown in Figures 2.5-236a and 2.5-236b. The effects of the soil which partially surround the building were investigated for the Sequoyah Nuclear Plant[4] and the effects are negligible.

Floor response spectra were computed for four individual artificial earthquakes (increased in amplitude by 15%) and the result found by taking the arithmetic mean of the four analyses. Spectra were computed for damping ratios of 0.005, 0.01, and 0.02 for the OBE and 0.005, 0.01, 0.02, 0.030, 0.040, and 0.050 for the SSE.

Vertical modes of vibration were calculated for comparison with the results for the dome as a shell. The rigid-body simulation of the dome as performed in the analysis of the cantilever beam model does not provide an accurate representation of the response of the dome to vertical earthquake excitation. Thus, an analogy was developed using shell theory to determine the earthquake moments and forces in the dome.

Figure 3.7-6 illustrates the logic performed in the analysis. The shell model is shown in Figure 3.7-7.

A flexibility matrix was developed using the shell model and the analysis performed using the response spectrum modal analysis techniques. The modes involving primarily deformation of the cylinder as computed in this analysis (modes 1 and 5) compare favorably with modes 1 and 2 of the vertical lumped mass cantilever beam analysis as shown in Table 3.7-5 (periods agree within 3%). Modes 2, 3, and 4 are primarily modes of vibration involving the dome. Also, the total meridional force at the base of the building as calculated by this method compares closely with the total force at the base in the cantilever beam analysis. This indicated the appropriateness of the analogy.

The structural response for the shell model was calculated for both the OBE and the SSE using structural damping of 2% and 5%, respectively.

Interior Concrete Structure

The idealized lumped mass model of the reinforced concrete structure used in the dynamic earthquake analysis is shown in Figure 3.7-8. Element properties are given in Table 3.7-6 and mass point properties in Table 3.7-7. The foundation structure interaction analysis of the Shield Building interior concrete-steel containment system discussed above for the Shield Building analysis revealed no significant change in the response of the interior concrete structure as compared to the response assuming a fixed base. Therefore, the dynamic analysis was performed using a fixed base model.

The dynamic earthquake analysis was performed by the response spectrum modal analysis technique. The results were computed for both the OBE and SSE conditions with structural damping of 2% and 5% respectively. The effects of torsion and longitudinal motion were considered. Periods for the normal modes of vibrations are listed in Table 3.7-8.

Response spectra were produced for damping values of 0.005, 0.01, and 0.02 for the OBE at mass points 1, 2, 3, 4, 6, 8, 10, 12, 13, and 14 for motion in both the east-west and north-south directions. Response spectra for vertical motion were obtained at ground and at mass point 14, and linear interpolation (Section 3.7.2.5) was used to produce vertical spectra at intermediate mass points. Response spectra were produced for damping values of 0.005, 0.010, 0.020, 0.050, 0.100, and 0.150 for the SSE at mass points 3, 5, 6, 8, 9, 10, 11, and 14 for motion in both the east-west and north-south directions. Response spectra for vertical motion were produced at mass point 14 and ground. Linear interpolation (see Section 3.7.2.5) was used to obtain vertical response spectra at intermediate points.

Auxiliary/Control Building

The idealized lumped mass model of the reinforced concrete structure is shown in Figure 3.7-9. Foundation-structure interaction was investigated by using a lumped mass-rock spring model, as discussed in Section 3.7.2.4. The results verified that a fixed base analysis may be used with no loss in accuracy. The dynamic analysis was performed by the response spectrum modal analysis technique. The results were computed for the OBE condition, and results for the SSE were obtained by doubling the values from the OBE. Element properties for the fixed base model are given in Table 3.7-9 and mass point properties in Table 3.7-10. Contributory weights to account for the soil contained within the wing walls at the north end of the structure were included in the total weights of the appropriate mass points. The effects of torsion and longitudinal motion were considered. Periods for the normal modes of vibrations are listed in Table 3.7-11.

Steel Containment Vessel

The containment vessel dynamic seismic analyses were performed using a lumped mass beam model. Structural and equipment masses were included, and structural properties were computed by hand calculations. The beam model and its properties are shown in Figure 3.7-7B.

Maximum overturning moments, shears, deflections, and shell stresses were computed by the response spectrum method. The site seismic design response spectra for 1% damping described in Sections 2.5.2.6 and 2.5.2.7 (both historical information) were utilized. The analyses were performed by CBI proprietary computer program 1017 described in Appendix 3.8C. Total response was computed by taking the absolute sum of modal responses.

A time-history analysis using the model in Figure 3.7-7B was performed in order to develop response spectra for equipment attached to the containment vessel. Four artificial earthquakes having an averaged response spectrum greater than the design response spectrum provided the seismic input. Using each of the artificial earthquakes individually, the beam model was analyzed and histories of acceleration were generated. For each of the acceleration histories, response spectra for various mass points and values of assumed damping were generated. The design spectra were the envelopes of the spectra generated from the four earthquakes and were used to design the vessel and the vessel's appurtenances in the scope of CBI, the vessel designer, fabricator, and erector. These calculations were performed by CBI computer programs 1017, 1044, and 1668, all of which are described in Appendix 3.8C.

As part of the review process and to provide response spectra for the design of equipment, piping and subsystems attached to and or supported by the containment vessel not supplied by the CBI, TVA performed an independent dynamic seismic analysis of the containment vessel. The ground motion input used to generate the floor response spectra consisted of the same four accelerograms of artificial earthquakes used by CBI.

The containment was idealized as a beam-type model consisting of lumped masses connected by massless elastic members. This lumped mass model is shown in Figure 3.7-7C. The element properties and inertial properties which were used in the analysis are shown in Table 3.7-5A and Table 3.7-5B, respectively.

<u>North Steam Valve Room</u>

The idealized lumped mass model of the reinforced concrete structure is shown in Figure 3.7-10. The structure is founded on bedrock and partially imbedded in soil. The effect of the soil restraint on the seismic response of the structure was included in the lumped mass model as soil spring restraints. The soil springs were calculated in accordance with the methodology given in Section 3.7.2.1.3. Element properties are shown in Table 3.7-12 and mass point properties in Table 3.7-13.

The dynamic analysis was performed using the response spectrum modal analysis technique. Response spectra were produced for selected elevations within the structure by the time history modal analysis method. Results were computed for the OBE with results for the SSE obtained by doubling those for the OBE. The frequencies for those modes considered important to the response of the structure are listed in Table 3.7-14.

Essential Raw Cooling Water Intake Pumping Station

The idealized lumped mass model of the reinforced concrete structure used in the analysis is shown in Figure 3.7-11. The dynamic analysis was performed by the response spectrum modal analysis technique. The results were computed for both the OBE and SSE with an assumed structural damping of 5% for each earthquake. Element properties are given in Table 3.7-15 and mass point properties in Table 3.7-16. The effects of torsion and soil restraint were considered. Periods for normal modes of vibration are listed in Table 3.7-17.

In addition, the effect on the building response of various water levels inside the pump wells was studied. The results of this study showed that the natural period of vibration was affected by variations in water level. Therefore, the structural responses used for design were those for the "worst-case" conditions. The amplitude of the response spectra peaks was not significantly affected by the water level variations. Only the location of the peak changed as the natural periods changed in response to water level variations. Accordingly, the response spectra peaks were broadened to account for the range of variations in natural period.

The response spectra for horizontal motion were produced for damping ratios of 0.005, 0.01, and 0.02 for both the OBE and SSE at mass points 6, 7, 8, and 10. Response spectra for vertical motion were produced for the base and mass points 8 and 10. Vertical spectra for intermediate mass points were developed using linear interpolation, as outlined in Section 3.7.2.5.

Essential Raw Cooling Water Intake Pumping Station-Retaining Walls

The reinforced concrete retaining walls were designed as a rigid structure subjected to the top of rock acceleration. Dynamic soil pressures on the retaining wall was determined in accordance with Reference [1].

3.7.2.1.2 Category I Rock - Supported Structures - Evaluation and New Design or Modification Analyses (Set B and Set B+C)

Analysis methodologies used in the original analyses (Set A) of Category I rock-supported structures were used in Evaluation (Set B) and New Design/Modification (Set B plus Set C) analyses except as noted in the remainder of this subsection. These exceptions provide for a seismic modeling approach which is consistent with current SRP Subsection 3.7.2 (NUREG 0800, Rev. 1) requirements.

Structures were represented as three-dimensional lumped mass stick models in the analyses except when coupling effects from omitted degrees of freedom were not significant. Actual centers of rigidity and actual mass eccentricities were modeled. Sufficient numbers of modes were included in the response to assure participation of at least 90% of the total mass. The simultaneous effects of three components of seismic input were considered by combining the co-directional responses resulting from the three components of input either by algebraic summation (for simultaneous inputs) or by the square-root-of-the-sum-of-squares (SRSS) method.

For design of structural elements, calculated seismic torsional moments were increased to account for accidental torsion. This increase is determined by multiplying the story shear force by the accidental eccentricity (defined as $\pm$ 5% of the structure dimension perpendicular to the direction of excitation.)

Rock-supported structures (ACB, IPS) were modeled as fixed base structures except where rock-structure interaction (Reactor Building) or structure-surrounding soil interaction (NSVR) effects were important. In these cases, three-dimensional finite element analysis was used to account for the interaction effects. The analyses were based on a foundation rock shear wave velocity of 5900 fps (Section 2.5.4.8, historical information). For rock-supported structures deeply embedded in soil, the effect of soil-structure interaction was considered and, where significant, included in the analysis model. The soil stiffness was determined as discussed in Subsection 3.7.2.1.4.

For WBN Unit 1 to support replacement of the steam generators, the same basic coupled model analytical approach used previously was adopted, while applying Replacement Steam Generator (RSG) parameters in the NSSS Reactor Coolant Loop (RCL) model. There was no change in the RCL seismic support arrangement. The RCL model was adjusted to account for an increase in the RSG mass and a lower center of gravity. The accuracy of the model and the Set B time history input motion was improved some, consistent with SRP guidance.

To support replacement of the WBN Unit 1 steam generators, the WBN Unit 1NSSS RCL was reanalyzed using the basemat Set B+C response spectra. Also, piping attached to the RCL was reanalyzed or evaluated for RCL Set B+C Amplified Response Spectra (ARS) and displacements. No piping or pipe support modifications were identified due to ARS or displacement changes. A comparison of the original Set B and B+C ARS with the new ARS was performed to determine whether or not reanalysis was required for other structures, systems or components in the Reactor Building. It determined that no other reanalysis was required due to ARS or displacement changes.

Structural damping values used in Set B and Set C analyses are given in Table 3.7-2. Where necessary, element associated damping was converted to modal damping using the strain energy composite modal damping approach. Damping values used in the new RCL analysis supporting the WBN Unit 1 steam generator replacementare consistent with the SRP, Regulatory Guide 1.61, and Table 3.7-2. The new RCL analysis damping values are for Set B+C instead of Set B, because replacement of the steam generators constitutes a significant modification.

Reactor Building Rock-Structure Interaction Analysis

The Reactor Building consists of the Shield Building (SB), Steel Containment Vessel (SCV), and interior concrete structure (ICS) including the NSSS piping, equipment, and components. The Set B and Set C Reactor Building model is a three branch, three dimensional (3-D) lumped mass model with branches representing the ICS, SB, and SCV. The ICS model was developed from a finite element analyses whereas the SB and SCV models were Set A models updated to include, in the vertical model, the fundamental vertical drumming mode of the dome for each of these structures. The ICS model used for the analyses also includes the NSSS model.

The Reactor Building is partially embedded in soils below the finished grade at Elevation 728.0 and in foundation rock below Elevation 702.78. In order to take into account the embedment effect on seismic responses, rock-structure interaction analyses were performed for the Reactor Building using the 3-D SSI analysis-computer program SASSI.

For the seismic response analysis, the input ground motion input was prescribed at the surface of the rock foundation (Elevation 702.78). This is the elevation of the top of the Reactor Building basemat where base fixity was provided for the structural models used in the original design basis seismic analysis (Set A). For rock-structure interaction analyses, the structural models for SB, SCV, and ICS were coupled together through the common Reactor Building basemat. The embedment of the Reactor Building basemat in the rock foundation was considered.

Using the SASSI computer program, time history response analyses for the Reactor Building were performed in the frequency domain using the Fast Fourier Transform (FFT) method.

Shield Building (SB)

The Set B and Set C dynamic model for the axisymmetric Shield Building structure is represented by a 3-D lumped mass single stick (Figure 3.7-5A) having the center of mass coincident with the center of rigidity for each lumped mass elevation. The model consists of 25 lumped masses interconnected with 25 elastic beam elements and a single-degree-of-freedom (SDOF) system located at the dome spring line elevation (Elevation 852.0) for simulating the fundamental vertical drumming mode of the dome. Except for the vertical SDOF system for the dome and the concrete modulus, the model configuration, lumped masses, and elastic beam element properties are the same as those used in the original design basis seismic analyses (Set A analysis). The vertical SDOF system for representing the fundamental vertical drumming mode of the dome was developed by matching the frequency and effective modal mass of the SDOF system with those of the fundamental vertical drumming mode of the dome obtained from a separate finite element modal analysis for the dome. The model geometry, lumped masses, and elastic beam element properties for SB used for Set B and Set C analyses are summarized in Table 3.7-4A.

Interior Concrete Structure (ICS)

The ICS consists of a complex assemblage of curved walls, columns and slabs which have some cross sections with significant asymmetry. In order to develop a seismic model, static, 3-D, finite element analyses were performed to determine the equivalent beam properties that simulate the seismic responses of the ICS. Consistency of equivalent stick model properties and response transfer functions with those of the finite element model demonstrated the adequacy of the 3-D equivalent stick model.

Since the equivalent beam model results in center-of-rigidity locations for axial and bending deformations different from those for shear and torsional deformations, the 3-D stick model for the ICS was represented by a combination of two sticks. One stick consists of elements with only axial areas of the structure located at the centers of rigidity for axial and bending deformations and another stick consists of elements with all other beam element properties, except the axial area, located at the centers of rigidity for shear and torsional deformations. The final configuration of the 3-D stick model for the ICS is shown in Figures 3.7-8A and 3.7-8B.

Mass and member element properties are summarized in Tables 3.7-6A and 3.7-6B. Mass

properties are unchanged from those of the original analysis (See Table 3.7-7).

Steel Containment Vessel (SCV)

The dynamic model for the SCV Set B and Set C analyses is represented by a 3-D lumped mass, concentric single stick model as shown in Figure 3.7-7A. The model consists of 23 lumped masses interconnected with 23 elastic beam elements and a vertical SDOF system located at the dome spring line elevation (Elevation 814.5) to represent the fundamental vertical mode of the dome. Mass and member element properties are defined in Table 3.7-5C. Except for the mass eccentricities and the SDOF vertical model, the model configuration, lumped masses, and elastic beam element properties are the same as those used in the original (Set A) design basis seismic analyses described in Section 3.7.2.1.1.

During the analysis of Set B and Set C, it was determined that the 5% accidental eccentricity will yield much higher eccentric responses than from the actual eccentricities which were used in Set A analysis. Therefore, the actual eccentricities were neglected in Set B and Set C analyses. However, 5% accidental eccentricity was used to calculate torsional moments. The SDOF vertical dome model for SCV was developed by matching the frequency and effective modal mass of the SDOF system with those of the fundamental vertical mode of the dome obtained from a separate finite element modal analysis.

Nuclear Steam Supply System (NSSS) Components

For Set B and Set C analyses, the dynamic model for the NSSS components is coupled with the Interior Concrete Structure (ICS) model, and the coupled model is used for seismic response analyses. The dynamic model for the NSSS components included in the coupled model for the ICS consists of the models for the Reactor Pressure Vessel (RPV), four primary reactor coolant loop piping (hot legs, cold legs, and cross-over legs), the steam generator (SG), and the reactor coolant pump (RCP) associated with each loop, as shown in Figures 3.7-8C through 3.7-8G. In coupling the NSSS model to the ICS stick model, the RCL attachment points are connected to the ICS model at the appropriate elevations of the attachment points through rigid links. The dynamic model data for the NSSS components are obtained from Westinghouse Electric Corporation.

Due to the presence of gaps and tension-only tie rods at the NSSS supports, these supports exhibit nonlinear behavior under dynamic loading conditions. For the purpose of linear response analysis, four linearized NSSS analysis cases, each with a unique set of linearized NSSS support stiffness, are used to represent the nonlinear behavior under various dynamic loading conditions. For each NSSS analysis case, a specific set of NSSS supports with their specified orientations are activated for a particular loading condition and linear support stiffnesses are developed and provided by Westinghouse Electric Corporation to represent the active supports for application to the particular analysis case.

The final acceleration response spectra and movements values are the envelope values resulting from the different NSSS cases. Furthermore, the ARS and movement values at the corresponding locations of the four loops are enveloped to obtain the enveloped ARS and movements applicable to all four loops.

The seismic analysis of the NSSS components which was performed by Westinghouse is discussed in Section 5.2.1.10.

Auxiliary Control Building (ACB)

The Set B and Set C three-dimensional lumped parameter fixed-base model of the Auxiliary Control Building is shown in Figure 3.7-9A. The centers of mass and centers of rigidity were modeled at their actual geometric locations as defined in Table 3.7-9A. The element properties and masses are unchanged from the original analysis, except for the concrete shear modulus, and are listed in Tables 3.7-9 and 3.7-10.

The dynamic analysis was performed by the time-history modal analysis technique. Structural responses were computed and floor ARS were generated for the same elevations as Set A. For Set C, since the structure damping ratios for OBE and SSE are the same (5%), the OBE responses were computed and the SSE responses were obtained by doubling the OBE responses. Separate OBE and site-specific SSE analyses were performed for Set B using structure damping ratios of 4% for OBE and 7% for site-specific SSE.

Essential Raw Cooling Water (ERCW) Intake Pumping Station (IPS)

The ERCW IPS original analysis model is updated to consider torsional effects. It incorporates rotatory inertia and the eccentricities between the centers of mass and centers of rigidity. No lateral soil springs were included as these had been determined from previous analyses to produce a negligible soil-structure interaction effect. The highest water level was used for both the Set C SSE and OBE earthquakes and Set B site-specific SSE and OBE earthquakes, since this condition yields the lowest frequency and hence would produce the highest response levels. The Set B and Set C IPS model is shown in Figure 3.7-11A. Table 3.7-15A presents the element properties. Tables 3.7-16A and 3.7-16B define the weight properties and coordinates of centers of mass and centers of rotation, respectively.

North Steam Valve Room (NSVR)

To account for the soil-structure interaction effects due to the presence of backfill surrounding the foundation walls, soil-structure interaction (SSI) analyses were performed for the NSVR. The methodology used for SSI analysis is the same as that used for Category I soil-supported structures described in Section 3.7.2.1.4.

The three-dimensional lumped mass model used in the seismic analysis of the NSVR superstructure is shown in Figures 3.7-10A and 3.7-10B, and the model properties are given in Tables 3.7-13A and 3.7-13B.

3.7.2.1.3 Category I Soil-Supported Structures - Original Analysis (Set A)

For structures founded on soil, the acceleration at top of rock was amplified or attenuated through the soil deposit using the techniques outlined in Section 3.7.2.4. The soil-supported structures were analyzed using lumped-mass and soil spring modeling techniques. A typical

model is shown in Figure 3.7-12.

Table 3.7-3 contains a tabulation of Seismic Category I soil-supported structures for the plant (small miscellaneous structures are not included in the table). Details of the supporting media and foundation characteristics are presented in Table 3.7-3 and Section 3.7.1.4. The horizontal and vertical translational soil springs and the rocking soil spring included in the lumped-mass model to simulate soil structure interaction are calculated using the procedures outlined by Whitman.[3] The damping ratio used for soil-supported structures depends on the predominant type of motion, as explained by Richard,[5] but is not permitted to exceed 10% in any case.

Embedment effects are accounted for by constructing a translational soil spring using Whitman's vertical spring expressions and attaching it to appropriate point or points on the structure. Specific features associated with the seismic analysis of the Category I soil-supported structures are discussed below.

<u>Diesel Generator Building</u>

The idealized lumped mass model of the reinforced concrete structure used in the analysis is shown in Figure 3.7-13. Element properties are given in Table 3.7-18 and mass point properties in Table 3.7-19. The effects of horizontal translation and rocking of the base were considered.

The soils investigation of Section 2.5.4 (historical information) revealed a soils profile from bedrock consisting of a firm silty gravel overlain by lean clays, silt of low plasticity, and sandy silt. In order to assure a firm foundation for the structure, the material between the top of firm gravel and the grade slab (a depth of approximately 17 feet) was excavated and replaced with compacted granular fill.

The Diesel Generator Building is founded on granular fill overlying firm gravel (see Figure 2.5-226). The shear wave velocity for the foundation material was determined to be 1650 ft/s and was used to calculate the value of the soil springs for the lumped-mass soil-structure interaction model. A parametric study was conducted to investigate the effects on building response of varying the shear wave velocity of the foundation material from 1150 ft/s to 2150 ft/s. The parametric study resulted in the structure being designed for earthquake loads from the peak of the amplified response spectrum for surface motion.

The predominant motion of the structure was a translatory rigid body motion. Motion of this type results in large damping; therefore, a damping ratio of 0.10 was used for the analysis. Longitudinal motion was also considered. Periods for the normal modes of vibrations are listed in Table 3.7-20.

Response spectra were produced for damping ratios of 0.005, 0.010, 0.020, 0.050, and 0.070 for mass points 1, 3, and 6 for motion in both east-west and north-south directions.

<u>Waste Packaging Area (WPA)</u>

The following two paragraphs describe the original design basis analysis for using Set A criteria performed for the WPA.

The idealized lumped mass model of the reinforced concrete structure is shown in Figure 3.7-14. Element properties are given in Table 3.7-21 and mass point properties in Table 3.7-22. The analysis indicated that the primary motion of the structure was in rocking and translation of

the base. Motion of this type results in large damping; therefore, a damping ratio of .10 was used in the analysis. Longitudinal motion was also considered. Periods for the normal modes of vibration are listed in Table 3.7-23.

Due to the extent of excavation for the Auxiliary Building and the results of the investigation for the Diesel Generator Building, all in situ material down to the top of rock was excavated and replaced with compacted granular fill for the WPA (see Figure 2.5-225). The shear wave velocity for the material was determined to be 1650 ft/s, and was used to calculate the value of the soil springs for the lump-mass soil-structure interaction model. A parametric study was conducted to investigate the effects on building response of varying the shear wave velocity from 1150 ft/s to 2150 ft/s. The parametric study resulted in the structure being designed for earthquake loads from the peak of the amplified response spectrum for surface motion. Additional studies beyond those described above have been performed to determine relative displacements between the WPA and the Auxiliary and Control Buildings.

Refueling Water Tanks and ERCW Pipe Tunnels

The refueling water tank and foundations were designed for seismic loads determined from the basic procedure outlined above. Soil property variations were considered in order to define conservative design loads, and ten-percent damping was used because of predominant translational soil spring motion. The adequacy of the design was later verified by more exact analytical techniques for soil-structure and fluid-structure interaction.

Pipe tunnels are analyzed as discussed under "Underground Electrical Concrete Conduit Banks" except axial loads are not considered due to the segmented configuration of the tunnel. Dynamic soil pressures on the walls are determined in accordance with Reference [1].

Underground Electrical Concrete Conduit Banks

The underground electrical concrete conduit banks which lead from the Auxiliary Building to the Diesel Generator Building and the Intake Pumping Station were seismically analyzed.

Utilizing the average values for the soil shear wave velocity and density, the ground deformation pattern in terms of wave length and amplitude is determined. The buried conduit banks are assumed to deform along with the surrounding soil layers. The average shear wave velocity of a single layer representation of a multi-layered soil system may be determined by:

$$V_{ST} = \sum \frac{V_s \, h'}{h}$$

Where,

V_{ST} = Average shear velocity in the soil, ft/sec
V_S = Shear velocity in each layer of soil, ft/sec
h' = Depth of each layer of soil, ft
h = Total depth of soil, ft

The fundamental period of the single layer is calculated from the following equation:

$$T = \frac{4h}{V_{ST}}(\text{seconds})$$

If the depth of the soil layer varies over the distance traversed by the buried conduit bank, both cases, for maximum and minimum depths, are considered.

The maximum amplitude of the sine wave which represents the maximum displacement of the conduit bank is:

$$A = \text{Displacement} = \left(\frac{T}{2\pi}\right)^2 * (\text{Accel})$$

Where,

T = Fundamental period, sec
Accel = Amplified soil acceleration value, in/sec^2

The wave length, L, is calculated as:

$$L = V_{ST}\ T$$

The bending moment resulting from the seismic disturbance, assuming the conduit bank follows the soil and deforms as a sine wave, is given by:

$$M = \frac{\pi^2 EIA}{(L/2)^2}$$

where,

M = Maximum bending moment, in-lb
E = Modulus of the conduit bank, psi
I = Moment of inertia of the conduit bank, in^4
A = Maximum amplitude, in
L = Wave length, in

The axial strain experienced by the conduit banks due to deformation of the soil is also evaluated. The axial strain due to seismic propagating waves is computed following the methods of Newmark, [15] [16] Yeh,[18] and Kuesel[17] which assume the soil is linearly elastic and homogenous, the conduit bank behaves as a slender beam, and the buried member deforms with the surrounding soil (this implies the strain in the soil equals the strain in the member).

The effect of soil strain from a seismic event on conduit bank turns in a buried system must be analyzed in greater detail than just calculating the axial strain. The effect of these strains on turns is more complex due to the turn trying to resist the strain. The complexity is a function of the conduit bank and backfill soil properties.

The basis for determining the effect of the strains on the conduit bank turns is described by Shah and Chu.[19] The Shah and Chu theory has been developed into an analysis procedure by Goodling.[20,21,22] The committee on Seismic Analysis of the ASCE Structural Committee on Nuclear Structures and Materials prepared a report "Seismic Response of Buried Pipes and Structural Components"[23] which explains and amplifies the referenced methodology[19] and analysis procedure.[20,21,22] These references shall be used for analysis of the effects of axial strain on buried conduit bank turns.

The magnitude of friction acting on the conduit banks to use in the analysis depends on several factors, such as surface condition, contact pressure, soil strengths, etc. The friction force acting on the conduit banks is determined in accordance with Reference [24].

Differential movement due to soil settlement or displacement during a seismic event was also evaluated in accordance with criteria given in Sections 2.5.4.10 and 2.5.4.8 (both historical information), respectively.

The conduit banks were evaluated for settlement due to the potential liquefaction of the underlying soil as discussed in Section 2.5.4.8 (historical information) (see Figures 2.5-576 through 2.5-578 for the potential settlement values). The banks were evaluated for potential settlements between manholes and at building/conduit interfaces. The only area of potential structural inadequacy was at the Intake Pumping Station (IPS). The conduit banks in this area (see Figure 3.8.4-46) required modification to accommodate the potential settlements. This modification consists of cutting 10 grooves on all 4 sides of the banks. The 2 inch wide by 2 inch deep grooves on top and sides and 3 inch wide by 2 inch deep grooves on the bottom begin 76 feet from the IPS and are spaced at 8 inch between centers for a distance of 6 feet along each bank. Settlement of the conduit banks will cause plastic hinges to develop at the grooves and at the pile supports farthest from the IPS. This results in a structural mechanism which will allow the conduit bank to settle without compromising the intended function of the encased conduits.

<u>Class 1E Electrical Systems Manholes</u>

These manholes are rigid structures which have the same motion as the soil deposits in which they are located. The soil deposits were analyzed as explained in Section 3.7.2.4. The accelerations obtained for the soil deposit at the level of the manholes were used to determine the inertia force on the structures and to calculate the increase in the static soil pressure using the shaking table experiments performed for the design of TVA's Kentucky Hydro Project,[1] as discussed in Section 3.7.2.1.1.

<u>Miscellaneous Yard Structures</u>

The ERCW discharge overflow structure, ERCW standpipe structures I and II, and other miscellaneous yard structures are normally rigid structures. These structures are designed for a rigid body acceleration. Dynamic soil pressures on the walls, if appropriate, are determined in accordance with Reference [1].

<u>Structure Interaction Analysis - WPA, CDWE, and ACB</u>

In the WPA Original Analysis (Set A) a decoupled, two-stage SSI analysis was used to determine conservative structural responses. An analysis, using the Set A Criteria and revised soil properties, confirmed that there is sufficient gap between the WPA and ACB to preclude

impact during a seismic event. For the CDWE, the Set A analysis was based on engineering judgments relating to the modeling of the supporting piles and on the assumption of full contact between the building's mat foundation and underlying soil. Additional analysis was performed to more accurately consider the stiffness of the pile groups and the postulated gap between the slab and soil. Results of this analysis confirmed that the gap between the buildings is sufficient for seismic separation and the design of the structure and piles is adequate.

3.7.2.1.4 Category I Soil-Supported Structures - Evaluation and New Design/Modification Analysis (Set B and Set B+C)

For Category I structures founded upon soil, the top-of-rock motions were considered to be amplified (or attenuated) through the soil. The value of amplification and the change in frequency content of the excitation were determined by a soil column analysis that incorporates strain-dependent soil properties. The soil properties were varied by the amount given in Tables 2.5-17A through 2.5-17D to obtain different soil surface motion time histories associated with mean, upper bound and lower bound shear moduli and bulk modulus for the horizontal and vertical analyses, respectively. For vertical motion, strain-compatible soil properties determined from the horizontal analysis were used. Using these surface motions as control motions, OBE and SSE and site-specific SSE and OBE Soil-Structure Interaction (SSI) analyses were performed and structural responses including floor acceleration time histories were obtained. The SSI analyses were performed using a 3-D flexible-volume substructuring technique and Fast Fourier Transform (FFT) method. From the floor acceleration time histories, ARS were developed. For Set C, the responses obtained from the four time history analyses were averaged. For Set B, the co-directional responses from the three component earthquake excitations were combined using the SRSS method for each of the three soil property cases. Responses from these three soil property cases were enveloped for Set B and C.

Details of the supporting media and foundation characteristics to be used in Set B and Set C analysis of Category I soil-supported structures are discussed in Section 2.5 (historical information). Additional details of seismic analyses specific to each of the Category I soil-supported structures are described in the following paragraphs.

Diesel Generator Building

The 3-D lumped parameter model used for the Diesel Generator Building is shown in Figures 3.7-13A and 3.7-13B, and the associated model properties are given in Tables 3.7-19A and 3.7-19B.

Refueling Water Storage Tank

The hydrodynamic effects were modeled considering the effects of tank flexibility. The 3-D lumped parameter model of the refueling water storage tank is shown in Figure 3.7-13C, and the associated model properties are given in Table 3.7-19C.

Waste Packaging Area

The waste packaging area does not house any safety systems and components. Therefore, Set B and Set C analyses were not performed.

ERCW Pipe Tunnels

Since the tunnels are embedded in soil, their response follows the response of the surrounding soil medium. Therefore, the ARS for the tunnels were obtained as the envelope of the ARS at the tunnel elevation from the soil column analyses considering mean, upper bound and lower bound shear moduli. For Set C, the ARS from the four time history analyses were averaged prior to enveloping. The horizontal ARS and the vertical ARS were determined from analysis of the appropriate soil column. The seismic analyses methodology used for the pipe tunnels is described in Section 3.7.2.1.3.

3.7.2.1.5 Category I Pile-Supported Structures - Original Analysis (Set A)

For structures founded on piles, the acceleration at top of rock was considered to be amplified through the soil as discussed in Section 3.7.2.4. The translational and rocking foundation springs included in the lumped mass model of the structure to characterize soil-structure interaction were calculated using Reference [3]. The damping ratio used for soil-supported structures depended upon the predominant type of motion as explained in Reference [5].

A more detailed description of the seismic analysis of Category I pile-supported structures is discussed below.

Additional Diesel Generator Building

Refer to Section 3.7.2.1.6.

Condensate Demineralizer Waste Evaporator Building (CDWE)

The CDWE Building is a pile supported, reinforced concrete structure. The building consists of two stories and is approximately 54 feet-9 inches by 41 feet-9 inches in plan and 59 feet high. The pile group supporting the CDWE Building consists of 104 vertical and 46 batter piles driven through 30 feet of soil to refusal in sound rock.

The seismic analysis of the CDWE Building was comprised of both a normal mode analysis using lumped mass models and a plane strain analysis using 2-dimensional models. The normal mode analysis was conducted for the north-south, east-west, and vertical directions. The plane strain analysis was conducted for the east-west and vertical directions assuming a unit depth in the north-south direction.

In the normal mode analysis, a model of the soil deposit was used to determine the acceleration time history at the top of ground from the specified bedrock acceleration records. The top of ground acceleration records were then used as input to a lumped mass model of the CDWE Building through a set of translational and rotational springs representing the pile group. The lumped mass models for the normal mode analysis are shown in Figure 3.7-15A.

The earthquake motion used in the analysis was determined by amplifying the four artificial earthquake input at top of rock through the supporting soil. The maximum top of rock horizontal accelerations for these earthquakes are 0.09g and 0.18g for the OBE and the SSE, respectively. The vertical motions are two-thirds of the horizontal.

The amplification of these earthquakes through the soil is performed by considering the soil as an elastic medium and making a dynamic analysis of a slice of unit thickness considering only the horizontal resistance of the soil. The soil deposit was divided into layers which would permit transmission of vibrational frequencies up to 30 Hz. An average value of the shear modulus was determined for each layer based on the effective vertical stress in the layer and then an

average for the entire deposit was calculated. To account for uncertainties in the soil properties, three soil profiles were considered in the normal mode analysis. The three profiles correspond to soil deposits having the calculated average value of shear modulus and variations of ± 50% in the shear modulus. Only the average profile was considered in the plan strain analysis. The values of shear modulus and corresponding shear wave velocities for the three soil profiles are shown in Table 3.7-23A. A damping ratio of 10% is used for the soil. From this analysis, four corresponding top of rock earthquake motions are obtained for use as input to the structural model. The vertical motion at top of ground is assumed to be two-thirds of the horizontal motion.

The lumped mass model of the building for the normal mode analysis consists of four mass points and four elements, the mass and inertia of the base, and translational and rotational springs representing the pile group. The mass points, elements, and spring properties are given in Table 3.7-23A.

The pile group is composed of 104 vertical and 46 batter piles. The pile group was modeled by equivalent translation and rocking springs in both horizontal directions and a vertical spring.

Once a set of spring constants were determined, the lateral and rocking springs were both modified by the same factor to produce a natural period for the structure of 0.15 second in each horizontal direction to correspond to the peak in the top of ground acceleration response spectrum. The spring constants representing the pile group are shown in Table 3.7-23A.

A normal mode time history analysis of the lumped mass model was conducted. A damping factor of 5% of critical was used in this step of the analysis for both soil springs and structural elements. The loads thus compared were considered to be overly conservative, and since the top of ground horizontal accelerations were approximately doubled by the base springs, the horizontal loads in the building were reduced by one-half. A plane strain analysis of the soil-structure system was then conducted for the SSE in the E-W and vertical directions to verify the reduction in the horizontal loads computed by the normal mode analysis. The input accelerations for the latter analysis were the top of rock acceleration records specified for the Watts Bar Nuclear Plant.

The plane strain analysis was conducted using a 2-dimensional model of the soil-structure system in order to verify reducing the results obtained in the normal mode analysis. The model included soil-structure interaction effects, and cases were run with and without the pile group stiffness included in the soil properties. Damping factors of 10% of critical for the soil elements and 5% of critical for the base mat and CDWE Building elements were used in the plane strain analysis. The soil properties are linear and elastic.

The time history accelerations specified for top of rock were applied at the base of the model, and the free field top of ground acceleration was compared to the lumped mass model top of ground motion. The plane strain analysis indicated the horizontal acceleration amplification through the soil and base springs in the lumped mass analysis was excessive and a reduction of the horizontal loads in the building by a factor of one-half was justified.

3.7.2.1.6 Category I Pile-Supported Structures - Evaluation and New Design/Modification Analyses (Set B and Set B+C)

Additional Diesel Generator Building (ADGB)

The original criteria for the ADGB design basis seismic analysis was based on NUREG-0800 and Regulatory Guide 1.60 ground design spectra. These criteria were incorporated into the UFSAR after the issuance of NUREG-0847, WBNP Safety Evaluation Report, Supplement 2,

1984. In order to bring the ADGB in line with the other Category I structure, the structure has been reanalyzed in accordance with Set B and Set C criteria. The seismic responses (ARS, accelerations, displacements, forces and moments) defined by Set B and the envelope of Set B and Set C (Set B + C) are used in evaluating the adequacy of existing structures, as well as new designs and modifications.

The 3-D lumped parameter model used for the ADGB is shown in Figures 3.7-15B and 3.7-15C, and the associated model properties are given in Tables 3.7-23B and 3.7-23C.

Condensate Demineralizer Water Evaporator Building (CDWE)

The CDWE Building does not house any safety-related systems and components. Therefore, Set B and Set C analyses were not performed.

3.7.2.2 Natural Frequencies and Response Loads for NSSS

The natural frequencies of Westinghouse supplied components are considered in the system seismic analysis. The natural frequencies are listed in detail in the component stress reports.

3.7.2.3 Procedures Used for Modeling

3.7.2.3.1 Other than NSSS

The procedures used to formulate original analysis mathematical models of each Category I structure have been discussed in Sections 3.7.2.1.1 and 3.7.2.1.3. The mass of supported equipment was considered in the lumped masses at the points of support. The stiffness of supported equipment was not considered in the lumped mass model of the structure.

For evaluation and new design or modification analyses, the stiffness and mass of a subsystem (supported equipment, a system, or a component) are included in the model if either Criteria 1 or 2 given below apply:

1) $0.01 \le R_m \le 0.10$ and $0.8 \le R_f \le 1.25$

2) $R_m \ge 0.1$

where,

$$R_m = \frac{\text{total mass of subsystem}}{\text{total mass of structure}}$$

$$R_f = \frac{\text{fundamental frequency of subsystem}}{\text{dominant frequency of structure}}$$

When the criteria given above for the inclusion of both stiffness and mass are not met the mass of a subsystem is included in the model if the subsystem is comparatively rigid in relation to the supporting structure and rigidly connected to the supporting structure.

3.7.2.3.2 For NSSS Analysis

The first step in any dynamic analysis for a system or component supplied by Westinghouse is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dash pots suitable for mathematical analysis. Essentially, the

procedure is to select mass points so that the displacements obtained will be a good representation of the motion of the system or component. Stated differently, the true inertia forces are not altered so as to appreciably affect the internal stresses in the structure or component. The mathematical model used for the dynamic analysis of the reactor coolant system is shown in Figure 5.2-1. Figure 5.2-2 shows the mathematical model of the reactor pressure vessel.

The determination as to whether the structure or component is analyzed as part of a system analysis or independently as a subsystem, is justified on a case by case basis.

3.7.2.4 Soil/Structure Interaction

3.7.2.4.1 Original Analysis (Set A)

For Category I structures founded upon soils, the rock motion was amplified to obtain the ground surface motion by considering the soil deposit as an elastic medium and making a dynamic analysis of a slice of unit thickness using only the horizontal shearing resistance of the soil. The four artificial earthquakes mentioned in Section 3.7.1.2 were considered as the input motion at top of rock. Once the time history of surface accelerations was known, a response spectrum was produced for the analysis of the soil-supported structure. The vertical surface motion was considered as two-thirds of the horizontal surface motion.

The soil amplification analysis is affected by the variations of onsite soil measurements, slanted soil layers, soil density, and depth of the soil deposit. Therefore, for structures supported on a soil deposit, the parameters of the soil deposit beneath the structure were varied to obtain a series of ground motion spectra. An envelope was drawn from these spectra resulting in the final ground motion spectrum used in analyzing the structure.

By following the procedure outlined, the maximum amplification of the ground response was obtained and the peak width of the ground response spectrum was wide enough to allow for variations in the frequencies of the structure due to variations in soil parameters.

3.7.2.4.2 Evaluation and New Design or Modification Analyses

For Category I structures founded upon soil, the top-of-rock motions were considered to be amplified (or attenuated) through the soil. The value of amplification and the change in frequency content of the excitation were determined by a soil column dynamic analysis that incorporates strain-dependent soil properties. Therefore, the soil properties beneath the structure were varied by the amounts given in Tables 2.5-17A through 2.5-17D to obtain different soil surface motion time histories.

For Set B analyses, the top-of-rock input motions are those defined by the Evaluation Site Design Response Spectra and Evaluation Site Design Time Histories of Section 3.7.1. For Set C analyses, the input motions are defined by the Original Site Design Response Spectra and the Original Site Design Time Histories (Section 3.7.1).

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.1 Original Analysis

Response spectra for use in computing the response of structural appurtenances, or of equipment attached to Category I structures were produced by the time-history modal analysis technique. The four artificially produced accelerograms (Section 3.7.1.2) were the input motion

at top of rock. To obtain a set of response spectra for one mass point for one direction of motion, the procedure outlined in Figure 3.7-37 was used.

Spectral values were computed for the periods using the distributions shown in Table 3.7-1, in addition to the natural frequencies of the structure. In all time-history calculations a time interval of 0.010 second was used.

Response spectra were computed for percentages of critical equipment damping of 0.5, 1.0, 2.0, 3.0, 4.0, 5.0 and 7.0. Response spectra were calculated for both the OBE and SSE; except, for those instances when the same percentage of critical structural damping was specified for both earthquakes, response was calculated for the OBE or SSE only (the SSE results equal twice the OBE).

Horizontal response spectra were produced at ground level, at major floors, and at other points of interest within the structure for both east-west and north-south directions, except where symmetry justifies the use of one direction.

For a direction in which torsion is considered, the time histories of accelerations used to produce the spectra will be computed where the maximum accelerations occur at that level (the farthest points on the structure from the shear center, on the axis perpendicular to motion).

Unless otherwise noted, vertical response spectra were produced at ground and at major floor elevations. The response spectra for ground was used throughout that portion of the structure where no structural amplification occurred. For other points, values were interpolated linearly between adjacent floors.

3.7.2.5.2 <u>Evaluation and New Design or Modification Analysis</u>

Response spectra for Set B and Set C analyses are produced by the time history modal analysis technique. For evaluation (Set B) analyses, the co-directional time history responses are either computed directly by simultaneous application of the directional seismic inputs or by the SRSS method. Set C co-directional responses are combined by the SRSS method only.

OBE (Set B) and OBE (Set C) constant damping response spectra are computed for damping ratios of 1, 2, 4, 5 and 7%. Site-specific SSE and SSE spectra are computed for 2, 3, 5 and 7%. Site-specific SSE, OBE (Set B), OBE and SSE variable damping response spectra are also computed for both Set B and Set C in accordance with ASME Code Case N411.

The ARS values were generated for the standard 75 spectral frequencies specified in Table 3.7.1-1 of the SRP plus the significant structure natural frequencies that are below the frequency limit of 33 Hz. For the ACB, UFSAR Table 3.7-1 spectral frequencies were used for Set C analyses. A study comparing the spectra obtained from the use of SRP and UFSAR frequencies concluded that the use of UFSAR frequencies for ACB Set C analyses is adequate.

Two solution methods were used to generate floor response spectra. These were time domain method of analysis and frequency domain method of analysis. For the time domain method of analysis, a time interval of 0.005 second was used for structural analysis, and time intervals of 0.005 and 0.0025 seconds were used for generation of floor response spectra. For the frequency domain method of analysis, a time interval of 0.01 second was used for structural analysis, and a time interval ranging from 0.01 to 0.0025 seconds was used for generation of floor response spectra.

The final Set B and Set C ARS include ± 15% and ± 10% peak broadening, respectively, for structures other than the ERCW tunnels. For Set C analyses, because of identical OBE and SSE structural damping, OBE ARS accelerations are one-half the corresponding SSE values. New design/modification ARS are defined by the envelope of Set B and Set C ARS.

The ERCW pipe tunnels are embedded in soil and their response follows the motion of the surrounding medium. The ARS at tunnel elevations were obtained from an envelope of the ARS generated from soil column analyses using the mean, upper, and lower bound soil shear moduli.

Vertical response spectra are calculated at the building extremities for the basemat and for all major floor elevations.

3.7.2.6 Three Components of Earthquake Motion

3.7.2.6.1 Original Analysis (Set A)

The seismic responses of Category I structures were computed by assuming the vertical earthquake to occur simultaneously with each of the two major horizontal directions separately. The derivation of the site response spectra and the design time histories for horizontal and vertical motion has been detailed in Sections 3.7.1.1 and 3.7.1.2, respectively.

3.7.2.6.2 Evaluation and New Design/Modification Analyses (Set B and Set C)

The seismic responses of the Category I structures are determined assuming that the three components of the earthquake occur simultaneously.

When the response spectrum method is used for seismic analysis of structures, the maximum structural response due to each of the three components of earthquake motion is combined by the SRSS of the maximum co-directional responses caused by each of the three components of earthquake motion at a particular point of the structure.

When the time history analysis method is used for Set B analysis of structures, the co-directional responses from each of the three components of earthquake motions are either combined algebraically at each time step or the maximum responses from each earthquake are combined by the SRSS method. For Set C time history analyses, only the SRSS method is used to combine co-directional responses.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Other Than NSSS

3.7.2.7.1.1 Original Analysis (Set A)

The responses of all Category I structures were computed by the response spectrum modal analysis method. The responses were calculated in each component mode. The total response was then calculated by determining the square root of the sum of the squares (SRSS) of the modal responses. For example, the total acceleration in any direction was calculated as:

$$a_T = \sqrt{a_1^2 + a_2^2 + \ldots + a_n^2}$$

Similar expressions exist for the other responses.

When the frequencies of two or more modes are found to be closely spaced (modes whose frequencies are within 10% of each other), the responses of these modes were combined in an absolute sum manner. The resulting total was treated as that of a pseudo-mode and combined with the remaining modes by the SRSS method.

The stresses in the structures were calculated assuming the vertical earthquake to occur simultaneously with either horizontal earthquake. For example, a typical expression for the stress x, caused by a horizontal earthquake in the x-direction and a vertical earthquake in the y-direction, would be:

$$\sigma_x = \pm\sigma_{xx} + \sigma_{xy}$$

3.7.2.7.1.2 Evaluation and New Design or Modification Analyses

The response spectrum method was used to determine the seismic responses for the Category I structures. The most probable response is obtained as the square root of the sum of the squares from the individual modes.

For Set B and Set C analyses, either the response spectrum or time history analysis methods were used to determine the seismic responses of Category I structures. When the response spectrum method was used, modal responses were combined in accordance with NRC Regulatory Guide 1.92, Rev. 1. Modal responses computed by the time history method were combined algebraically at each time step. For either analysis method, a sufficient number of modes were investigated to assure participation of all significant modes.

3.7.2.7.2 NSSS System

The total seismic response of systems and major components within Westinghouse scope of responsibility is obtained by combining the individual modal responses utilizing the SRSS method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10% of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor $\varepsilon_{k\ell}$. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{\ell=K+1}^{N_j} R_K R_\ell \varepsilon_{K\ell} \qquad (1)$$

Where,

RT	=	total response
Ri	=	absolute value of response of mode i
N	=	total number of modes considered
S	=	number of groups of closely spaced modes
Mj	=	lowest modal number associated with group j of closely spaced modes
Nj	=	highest modal number associated with group j of closely spaced modes
$\varepsilon_{\kappa l}$	=	coupling factor with

$$\varepsilon_{K\ell} = \left(1 + \left[\frac{\omega_{K'} - \omega_{\ell'}}{(\beta_{K'}\omega_K + \beta_{\ell'}\omega_\ell)}\right]^2\right)^{-1} \tag{2}$$

$$\beta'_j = \beta_j + \frac{2}{\omega_j t_d} \tag{3}$$

$$\omega'_j = \omega_j[1 - (\beta'_j)^2]^{1/2} \tag{4}$$

Where,

ϖj	=	frequency of closely spaced mode j (rad/sec)
βj	=	fraction of critical damping in closely spaced mode j
t_d	=	duration of the earthquake (sec.)

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely with modes 2, 3, 4 and 6, 7. Therefore,

S	=	2 number of groups of closely spaced modes
M_1	=	2 lowest modal number associated with group 1
N_1	=	4 highest modal number associated with group 1
M_2	=	6 highest modal number associated with group 2
N_2	=	7 highest modal number associated with group 2
N	=	8 total number of modes considered

The total response for this system is, as derived from the expansion of Equation (1):

$$\begin{aligned} R_T^2 &= [R_1^2 + R_2^2 + R_3^2 + \ldots + R_8^2] + 2 R_2 R_3 \in_{23} \\ &+ 2 R_2 R_4 \in_{24} + 2 R_3 R_4 \in_{24} + 2 R_6 R_7 \in_{67} \end{aligned} \quad (5)$$

3.7.2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

All interfaces between Category I and non-Category I structures were designed to withstand the displacement and/or dynamic loads produced by both the Category I and non-Category I structures and equipment. The Turbine Building, Service Building and Old Steam Generator Storage Facility (OSGSF) are the only non-Category I structures for which this section applies.

The Turbine and Service Buildings were analyzed for a total lateral base shear computed as the product of the mass of the structure and the ground acceleration for the SSE. The total lateral shear was distributed in the height of the structure according to the provisions of the Uniform Building Code.

The OSGSF seismic design is in accordance with the 2000 edition of the International Building Code.[31]

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

To account for variations in structural frequencies owing to variations in material properties of the structure and soil and to approximations in modeling techniques used in seismic analyses, the computed floor response spectra are smoothed and peaks associated with the structural frequencies are broadened $\pm$ 10% for Set A and Set C and $\pm$ 15% for Set B.

For the soil-supported structures in which floor response spectra were produced, the soil properties were varied to account for variations in soil properties. Soil-structure interaction was considered as discussed in Sections 3.7.2.1 and 3.7.2.4.

3.7.2.10 Use of Constant Vertical Load Factors

3.7.2.10.1 Other Than NSSS

3.7.2.10.1.1 Original Analysis (Set A)

A vertical lumped mass dynamic analysis using the techniques outlined in Section 3.7.2.1.1 was performed for all of the Category I structures to determine the vertical loads. The results for each horizontal earthquake analysis were separately added on an absolute basis to the loads from the vertical earthquake analysis. Static vertical load factors were not used unless the dynamic analysis indicated the structure behaved as a rigid body in the vertical direction.

3.7.2.10.1.2 Evaluation and New Design or Modification Analyses

The Category I structures, when analyzed for vertical motion, used lumped-mass dynamic techniques as discussed in Section 3.7.2.1.2. For Evaluation Analyses (Set B), the co-directional time history responses are either computed by simultaneous application of the seismic input in three directions or by the SRSS method. For Set C, co-directional responses are combined by SRSS only. For systems and components the appropriate floor response spectra was used in the analysis. Static load factors were not used for either Set B or Set C analysis.

3.7.2.10.2 For NSSS

Static vertical load factors are not used as the vertical floor response load for the seismic design of safety-related systems and components within Westinghouse scope of responsibility.

3.7.2.11 Methods Used to Account for Torsional Effects

The dynamic analysis of structures is discussed in Section 3.7.2.1. In original or Set A analyses, torsional effects were considered by using a lumped-mass cantilever beam model to represent stiffness and inertial characteristics. The torsional moment of inertia, eccentricity, and mass moment of inertia were included in the analyses.

In the process of preparing lumped-mass mathematical models for the Set A analyses, the location of both the center of rotation and center of mass for each floor were computed. Accelerations and deflections were calculated where their maximum values occurred (at the farthest points on the structure from the shear center, on the axis perpendicular to the direction of motion).

For Set B and Set C analyses, modeling of torsional effects was refined by three-dimensional modeling.

The models described above were subjected to seismic excitations and the resultant responses in the form of frequencies, mode shapes, moments, and forces were obtained.

3.7.2.12 Comparison of Responses - Set A versus Set B

The comparison of Set A and Set B responses showed that, in general, Set A responses were higher. In making the ARS comparisons, the applicable damping ratios of Set A and Set B were used. In certain frequency ranges, Set B responses were higher than Set A responses. An evaluation was performed on a building by building basis to assess the impact of Set B response. Adequacy of structures, systems, and components for Set B effects has been documented in calculations.

As a sample comparison of Set A and Set B responses, the ARS comparisons for Auxiliary Control Building, which is a rock-supported structure, and for the Diesel Generator Building, which is a soil-supported structure, are presented. The ARS for north-south, east-west and vertical directions are compared. The comparison at Elevation 692.0 and Elevation 814.25 of the Auxiliary Control Building are presented in Figures 3.7-15D through 3.7-15I.

3.7.2.13 Methods for Seismic Analysis of Dams

Since no dams are utilized to impound bodies of water to serve as heat sinks, this section is not applicable to this site.

3.7.2.14 Determination of Category I Structure Overturning Moments

3.7.2.14.1 Original Analysis

From the dynamic analyses of the structures, the seismic moments, shears, and vertical loads were determined at the base of the structure. These loads were used in combination with other appropriate loads in determining total overturning effects as discussed in Section 3.8.

3.7.2.14.2 Evaluation and New Design or Modification Analysis

From the dynamic earthquake, analyses total moments, shears, and vertical loads were computed.

The earthquake moment, shear, and vertical load were used in combination with other appropriate loads in determining total overturning effects as discussed in Section 3.8.

3.7.2.15 Analysis Procedure for Damping

The damping values used in the dynamic earthquake analyses of Category I structures are given in Table 3.7-2.
For Set A analysis, the Category I structural models were not coupled together, therefore, the structural damping values used in the seismic analyses are as shown in Tables 3.7-2 and 3.7-24.

For Set B and Set C analyses, either composite modal damping or structural damping were used in the seismic analyses of Category I structures. The damping values used for the various structures and components are given in Tables 3.7-2 and 3.7-24. The damping used in the seismic analysis of systems and components are also given in Tables 3.7-2 and 3.7-24.

The new Unit 1 RCL analysis damping values (from Table 3.7-2) are for Set B+C instead of Set B. Under the Westinghouse standard scope of supply and analysis, the lowest damping value associated with each element of the system is used for all modes.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods for Other Than NSSS

The seismic analysis of Category I piping systems is described in detail in Section 3.7.3.8.

In the analysis of piping subsystems there are two distinct approaches to seismic analysis. A detailed analysis is discussed in Section 3.7.3.8.2 and a simplified analysis is discussed in Section 3.7.3.8.3.

The general seismic analysis of Category I equipment and components is discussed in Section 3.7.3.16. Additional details applicable for simplified analysis are discussed in Sections 3.7.3.5 and 3.7.3.10.

The seismic analyses of HVAC and conduit/cable tray subsystems are discussed in Sections 3.7.3.17 and 3.10.3, respectively.

The detailed seismic analyses of Category I subsystems is based upon dynamic analyses using the lumped mass normal mode method with idealized mathematical models. The inertial properties of the models are characterized by mass, eccentricity, and mass moment of inertia of each mass point. Mass points are located at carefully selected points in order to accurately model the subsystem as described in Section 3.7.3.3.1. The stiffness properties are characterized by the moment of inertia, area, torsion constant, Young's modulus, and shear modulus.

The response of Category I subsystems are computed by the response spectrum modal analysis method for designs. All significant modes of vibration are considered in determining the total response. Subsystem response is calculated in three orthogonal directions.

Seismic responses of the Category I subsystems, equipment, and components are determined and combined in accordance with Sections 3.7.3.6 and 3.7.3.7. The damping ratios used in the dynamic analyses of the structures, subsystems, and equipment/components are shown in Table 3.7-2.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Category I Systems and Components Other Than NSSS

During the design life of the plant (40 years), two earthquakes of OBE magnitude and one SSE are postulated to occur. This was based upon a study of seismic history in the Southern Appalachian Province over a 100-year period. Based on this study, each occurrence is conservatively assumed to have a time duration of 15 seconds of strong excitation.

For Class A Category I components, an evaluation of predominant frequencies revealed that the most significant response of components is conservatively considered using an average frequency of 20 Hz. Therefore, the total number of cycles considered for the OBE and SSE are 600 and 300, respectively.

The seismic qualification testing of Category I equipment considers the number of events and durations described above in accordance with IEEE 344-1975.

ASME Section III Class 1 Piping Analysis - Since the piping in this scope has been reanalyzed in accordance with SRP requirements, the piping analysis has assumed the occurrence of 5 OBEs and 1 SSE. The number of peak stress cycles may be obtained from the synthetic time history used for the analysis (with a minimum duration of 10 seconds), or a minimum of 10 peak stress cycles per event assumed.

3.7.3.2.2 NSSS System

Where fatigue analysis of mechanical systems and components is required, Westinghouse specifies in the equipment specification that 20 occurrences of OBE having 20 cycles of maximum response for each occurrence, be analyzed. The fatigue analyses are performed as part of the stress report.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Other Than NSSS

3.7.3.3.1.1 Modeling of Piping Systems for Detailed Rigorous Analysis

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at the nodes connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points are such that the adequate simulation of the dynamic characteristics of the piping system is ensured. All concentrated weights on the piping system such as main valves, relief valves, pumps, motors, and effects of support mass on piping system when found to be significant are modeled as lumped masses unless isolated from the system by positive anchorage. The torsional effects of the valve operators and the other line-mounted equipment with offset center of gravity with respect to center line of the pipe, is included in the analytical model.

3.7.3.3.1.2 Modeling of Equipment

For seismic analysis, Seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

1. The number of modes of a dynamic system is controlled by the number of masses used. The number of masses is chosen so that all significant modes are included. The modes are considered as potentially significant if the corresponding natural frequencies are less than 33 Hz. For modes greater than 33 Hz the rigid response contribution is considered.

2. Mass is lumped at points where significant concentrated weight and continuous mass are located.

3.7.3.3.1.3 Modeling of HVAC, Conduit, and Cable Tray Subsystems

Runs of HVAC, conduit, and cable tray subsystems (including supports) are modeled by continuous or discrete mass models with the interconnecting elements represented by their effective stiffness properties. Additional lumped masses are applied at or near significant concentrated weights such as from fittings or other in-line or attached commodities. Significant concentrated weights are those which cannot be adequately represented by smearing their effect as part of the overall uniform mass. Mass eccentricities and torsional stiffnesses are considered. Where models are truncated, at least one span and the next support on either side of the contiguous span(s) and support(s) of interest for evaluation are modeled. Alternately, the contiguous span(s) and support(s) of interest are evaluated with one half of the adjacent spans on either side modeled with symmetry boundary conditions such that no artificial stiffening is introduced.

A sufficient number of masses (or degrees of freedom) are modeled such that additional masses would not increase the predicted responses by more than 10%. Alternately, the number of masses are modeled to be at least twice as many as the number of modes with frequencies less than 33 Hz. The dynamic analysis considers all modes with significant mass participation such that inclusion of additional modes would not increase the predicted responses by more than 10%. Alternately, the dynamic analysis considers all modes up to 33 Hz and includes an additional check for any missing mass.

3.7.3.3.2 Modeling of NSSS Subsystems

The criteria and procedures used for modeling of NSSS subsystems is given in Section 3.7.2.3.

3.7.3.4 Basis for Selection of Frequencies

3.7.3.4.1 Other Than NSSS

The method used to analyze systems for dynamic loading is the modal response spectrum method.

Frequencies of the subsystems are selected such that all significant modes of vibration are included in the analysis. Frequencies of simplified analysis models are determined by solutions of closed form expressions. Frequencies of detailed analysis models are determined by computerized solutions.

The subsystem or component model is subjected to loadings in the form of accelerations that represent the seismic environment of its supports. Since the response spectrum employed is representative of the building elevation at the equipment/system location considered, structural amplifications are reflected in the spectra. Therefore, the input acceleration values taken from the building response spectra and utilized as input to the dynamic analysis of the subsystem or component assures the model is loaded in a representative manner and the proper amplifications determined. The subsystem or component was analyzed and designed for the amplified loading.

3.7.3.4.2 NSSS Basis for Selection of Forcing Frequencies

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports based upon the mass and stiffness characteristics of the system, will determine the seismic accelerations which the equipment must withstand.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low-period region of the floor response spectra.

2. If the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant and the equipment behaves as though supported on the ground.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

In addition, an equipment/support system is considered to be rigid if the fundamental natural frequency is greater than 33 Hz.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

3.7.3.5.1 Other Than NSSS

For discussion of the equivalent static load method as applied to equipment/components, see Sections 3.7.3.10.1, 3.7.3.16.1, 3.7.3.16.2, and 3.7.3.16.3.

For other Category I subsystems, the following discussion applies:

Simplified seismic analysis by the equivalent static load method may be used as an alternative to detailed computer analysis when the subsystem being analyzed is adequately represented by an effective one degree-of-freedom system with multi-mode effects accommodated by the use of a multi-mode factor. A modal participation factor of 1.0 is used for the equivalent static load method. If the subsystem is determined to be rigid (fundamental frequency $\geq$ 33 Hz), then the acceleration of the building at the elevation of the subsystem attachment (floor zero period acceleration) is used with a multi-mode factor of 1.0; i.e., the subsystem is evaluated for rigid-body response. When no frequency evaluation of the subsystem is made, the peak acceleration of the applicable floor response spectrum is used multiplied by a multi-mode factor of 1.5 except where a lower factor is justified. When a frequency evaluation is made and the subsystem is determined to be flexible, the highest acceleration at or above the determined frequency is used for evaluation multiplied by a multi-mode factor of 1.5 except where a lower factor is justified. For HVAC, conduit, and cable tray subsystems a multi-mode factor of 1.2 has been justified.

3.7.3.5.2 Use of Equivalent Static Load Method of Analysis for NSSS

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient, which is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as a single-degree-of-freedom system are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems, which may be in the resonance region of the amplified response spectra curves, are increased by 50 percent to account conservatively for the increased modal participation.

3.7.3.6 Three Components of Earthquake Motion

Seismic responses of Category I subsystems, equipment, and components are analytically computed or simulated by qualification tests for the applicable Set A, B, and C seismic inputs in three orthogonal directions. The Set A, B, and C inputs for original analysis/qualification, evaluation, and new design/modification are described in Section 3.7.2.

3.7.3.6.1 Piping Subsystems

The seismic responses of Category I piping subsystems are determined assuming that the three components of the earthquake motion occur simultaneously. The maximum response due to each of the three components of earthquake motion is combined by SRSS of the maximum directional responses caused by each of the three components of earthquake motion.

3.7.3.6.2 HVAC Ducting, Conduit, and Cable Tray Subsystems

The seismic responses of HVAC ducting, cable tray, and conduit subsystems are determined by two dimensional seismic analysis and associated testing of representative duct, cable tray, and conduit spans. Seismic input in each major horizontal direction is applied separately but simultaneously with vertical input. Horizontal and vertical responses are analytically combined by absolute summation.

3.7.3.6.3 Other Than NSSS Equipment and Components

The seismic responses of Category I equipment and components were determined by analysis or test in accordance with the guidelines of IEEE 344-1971 for procurements initiated prior to September 1, 1974. After that date procurement, evaluation, and modification activities applied the guidance of IEEE 344-1975 to determine the seismic responses.

Floor or wall mounted equipment and components and their supports and anchorage are seismically analyzed or tested by application of the required seismic response spectra described in Section 3.7.2.5, in a two-dimensional manner. Seismic input in each major horizontal direction is applied separately but simultaneously with vertical input. Horizontal and vertical responses are analytically combined by absolute summation.

Seismic responses of line-mounted equipment and components are determined by device analysis or testing techniques from IEEE 344-1971 or IEEE 344-1975, as applicable. These techniques are applied in a two-dimensional manner relative to the three orthogonal local axes of the line-mounted equipment and component. Calculated seismic response of the subsystem at the equipment and component location is maintained at a level which is less than or equal to the device seismic qualification level.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Other Than NSSS

Modal responses of the piping subsystems are combined in accordance with Regulatory Guide 1.92, Revision 1. Modal responses of other subsystems are analytically combined by the techniques described in Section 3.7.2.7.1 for structures.

Category I equipment and components are seismically analyzed or tested by IEEE Standard 344-1971 or -1975 techniques, as described in Section 3.7.3.6. In accordance with these standards, modal responses are analytically combined by SRSS techniques except for closely-spaced modes whose responses are combined by absolute summation.

3.7.3.7.2 Combination of Modal Responses of NSSS

For the NSSS procedure for the combination of modal responses see Section 3.7.2.7.2.

3.7.3.8 Analytical Procedures for Piping Other Than NSSS

3.7.3.8.1 General

The analysis of classified fluid system components other than the reactor coolant system considers both static and dynamic loadings. The loading combinations considered and the allowable stress limits are discussed in Section 3.9.3.1. Thermal expansion, dead load, and normal operational stresses due to system pressurization for Category I piping systems are analyzed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1 Nuclear Power Plant Components, 1971 Edition up to and including the Summer 1973 Addenda. Non-nuclear safety classes of pipe are analyzed in conformance with ANSI B31.1, Power Piping Code, 1973 Edition up to and including Summer 1973 Addenda as shown in Table 3.2-5. In addition, TVA Class M (chilled water) piping conforms to ANSI B31.5, 1974. Stresses due to all loadings are appropriately combined with the seismic stresses in accordance with Code requirements.

As permitted by NA-1140 of applicable ASME Code, the following sections of more recent editions and addenda of the ASME Boiler and Pressure Vessel Code and ASME Code Cases are used. All related requirements were met.

A. CODE EDITIONS AND ADDENDA

1. Stress Intensification Factors

   a. 1974 Code; used for Stress Intensification Factors for Class 2 and 3 piping.

2. Nozzle Dimensions

   a. Figure NB-3686.1-1 for nozzle dimensions from the Summer 1975 Addenda.

3 Material Properties

   a. 1980 Edition - including Summer 1980 Addenda, Appendix I, Table I-4.0; for thermal conductivity and thermal diffusivity of materials.

   b. 1983 Edition - including Winter 1983 Addenda, Appendix I, Table I-5.0; for coefficient of Thermal Expansion of materials which are not available in the Code of Record.

   c. 1983 Edition - including Summer 1985 Addenda, Appendix I, Table I-6.0; for Modulus of Elasticity of materials which are not available in the Code of Record.

d. 1983 Edition - including Summer 1985 Addenda, Appendix I, Tables I-1.1, I-1.2, I-1.3, I-2.1, I-2.2, I-3.1, I-3.2, I-7.1, I-7.2, I-7.3, and I-9.1 for materials which are not available in the Code of Record.

4. <u>Stress Qualification</u>

a. 1980 Edition - up to and including Winter 1982 Addenda, Section III, Subsection NB; May be used for the stress qualification of Class 1 piping (NB-3600).

b. 1974 Edition - Summer 1976 Addenda, Section III, Paragraph NB-3630 (d); used for Class 1 piping which can be analyzed per requirements of Subsection NC.

c. 1974 Edition - Winter 1976 Addenda, Section III, Paragraph NC/ND-3611.2.

d. 1977 Edition - Section III, Paragraph NC/ND-3652.3.

e. 1974 Edition - Summer 1975 Addenda, Section III, paragraph NC/ND-3651.

f. 1974 Edition - Section III, Paragraph NC/ND-3652.4.

5. <u>Welded Attachments</u>

a. 1980 Edition - Winter 1980 Addenda, Section III, Paragraph NB-4433 which permitted the use of continuous fillet or partial penetration welds for welded structural attachments (Lugs) to the pipe.

6. <u>Flange Qualification</u>

a. 1983 Edition - up to and including Winter 1983 Addenda, Section III; Used for Class 1 Flange qualification per NB-3658; Used for Class 2 and 3 Flange qualification per NC-3658 and ND-3658.

7 <u>Relief and Safety Valve Thrust</u>

a. 1977 Edition - Winter 1978 Addenda, Section III, Paragraph NC/ND-3622.5 and Appendix O.

B. CODE CASES

1. Half-Coupling Branch Connections

a. Code Case N-313, November 28, 1986, Alternate Rules for Half Coupling Branch Connections, Section III, Division 1, Class 2.

2. Response Spectra

a. Code Case N-411-1, February 20, 1986, Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Systems, Section III, Division 1, may be used.

3. Stress Qualification

a. Code Case 1606-1, December 16, 1974, Stress Criteria, Section III, Classes 2 and 3 Piping Subject to Upset, Emergency, and Faulted Operating Conditions.

b. Code Case N-319, July 13, 1984, Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping. (Applicable to Unit 1 Only)

c. Code Case N-319, July 13, 1981, Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping. (Applicable to Unit 2 Only)

4. Welded Attachments

a. Code Case N-122, January 21, 1988, Stress Indices for Integral Structural Attachments, Section III, Division 1, Class 1. (Applicable to Unit 1 Only)

b. Code Case N-318-3, September 5, 1985, Procedure for Evaluation of the Design of Rectangular Cross Section Welded Attachments on Class 2 or 3 Piping, Section III, Division 1.

c. Code Case N-391, November 28, 1983, Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1.

d. Code Case N-392, November 28, 1983, Procedure for Evaluation of the Design of the Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping, Section III, Division 1.

e. Code Case N-122, January 21, 1982, Stress Indices for Integral Structural Attachments, Section III, Division 1, Class 1. (Applicable to Unit 2 Only)

Category I piping is classified into two analytical categories. These categories are defined below.

Rigorous Analysis (Detailed Seismic Analysis)--A comprehensive computer-aided analysis of the piping system to ensure that the system design meets all the ASME Section III requirements for stress in the piping.

<u>Alternative (Simplified) Analysis</u>--A conservative method for locating supports and determining support loads, using computer generated data, hand calculations and/or computer aided analysis to ensure that the ASME Section III code requirements are met.

<u>Systems Rigorously Analyzed</u>

TVA evaluates the necessity of performing a Rigorous Analysis on all piping systems and identifies the limits of the analysis using the following guidelines:

1. Class A piping systems not analyzed by the NSSS vendor.

2. TVA Class B, C and D lines 2-1/2inches in diameter and larger.

3. Piping in Category I structures larger than 1-inch diameter that has a maximum operating temperature of 200°F or greater and a maximum operating pressure of 275 psig or greater unless it is determined that there is not a potential for unacceptable pipe rupture interactions.

4. Piping which, due to high temperature or other extraordinary loading conditions, cannot be supported using alternate analysis methods.

<u>Systems Analyzed by Alternate (Simplified) Methods</u>

Piping requiring seismic qualification, but not requiring rigorous analysis as outlined above, may be evaluated according to the alternate methods.

3.7.3.8.2 <u>Detailed Seismic Analysis (Rigorous) for Piping Systems</u>

A detailed seismic analysis is performed on applicable piping systems by the response spectrum method. Each pipe run is idealized as a mathematical model consisting of lumped masses connected by weightless elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe, the flexibility matrix for the pipe is determined. The flexibility calculations include the effects of the torsional, bending, shear, and axial deformations. The stiffness of curved members, valves, branch connections, etc., is also taken into consideration.

Once the flexibility and mass matrices of the mathematical model are determined, the frequencies and mode shapes for all significant modes of vibration are determined. All modes having a period greater than 0.0303 seconds (natural frequencies < 33 Hz) are used in the analysis. The mode shapes and frequencies are solved in accordance with the following equation:

$$(K - w_n^2 M) \phi_n = 0$$

where: K = Square stiffness matrix of the pipe loop

M = Mass matrix for the pipe loop

w_n = Frequency for the nth mode

$ø_n$ = Mode shape matrix of the nth mode

After the frequency is determined for each mode, the participation factors can be calculated by the following equation:

$$\Gamma_{njk} = \frac{\phi_n^T M \gamma_{jk}}{\phi_n^T M \phi_n}$$

Where:

Γ_{njk} = Participation factor for mode n in the jth direction of support zone k.

γ_{jk} = Displacement matrix of all nodes due to a unit displacement of the jth direction restrained degrees of freedom in support zone k.

Support zone = A set of restrained nodes which move together during a dynamic event.

Using these results and the corresponding spectral accelerations of the mode for the direction and support zone being excited, the response for each mode is determined by the following equation:

$$(V_{in})_{jk} = \frac{\Gamma_{njk} \phi_{in} S_{anjk}}{W_n^2}$$

Where:

$(V_{in})_{jk}$ = Displacement of mass for mode n for an earthquake in the jth direction of support zone k.

$ø_{in}$ = Value associated with mass i in $ø_n$

S_{anjk} =Spectral acceleration for mode n for an earthquake in the jth direction of support zone k.

Using these results, the maximum displacements for each mode are calculated for each mass point in accordance with the following equation:

$$(V_{in})_j = \sum_{k-1}^{NZ} \left| (V_{in})_{jk} \right|$$

where:

$(V_{in})_j$ = Displacement of mass i for mode n for an earthquake in the jth direction

NZ = Number of support zones used for the pipe loop. However, if ASME Code Case N-411 damping values are used then all supports are in a single support zone.

The maximum displacements for each mode are calculated as follows:

$$V_{in} = \sqrt{\Sigma (V_{in}^2)_j}$$

where:

(V$_{in}$ = maximum displacement of mass i for mode n.

j = x, y, and z

The maximum displacement for each mass is determined by combining the maximum deflection for each mode by the method described in Section 3.7.3.7. The contribution from higher frequency modes (period less than 0.0303 seconds) are combined with lower frequency modes by the SRSS rule.

With the displacements known, the associated member forces/moments can be obtained by standard structural techniques. The forces for each mode and each earthquake direction will be combined using the conventions described above.

3.7.3.8.3 Alternate (Simplified) Analysis for Piping Systems

Section 3.7.3.8.1 defines alternate analysis and specifies the piping for which it may be applied. Various methods are used to perform alternate analysis. These methods may involve the use of simple beam equations, computer generated data and/or computer assisted analysis. For each method, the following general requirements are observed.

1. Deadweight

   Supports are located such that adequate rigidity is assured and pipe sagging is minimized.

2. Seismic

   Seismic effects are approximated using accelerations from the applicable building response spectra. Response spectra accelerations at the frequency computed for the piping system are used except that if the computed frequency is below the frequency corresponding to the peak of the response spectra, the peak accelerations are used. The response spectra accelerations are increased by at least 50 percent to account for multimode response, unless justification is provided for using a lesser increase.

3. Thermal Expansion and Anchor Movement

Thermal expansion and anchor movement are evaluated using conventional hand calculation methods, the results of computer analysis of typical configurations and/or computer aided thermal flexibility analysis.

4. Pipe Stress

Pipe stress resulting from applicable load sources are evaluated and combined in accordance with applicable code requirements. Details of load combinations and stress limits are provided in Section 3.9.3.1.2.

5. Support Loads

Support loads resulting from applicable load sources are evaluated and combined as specified in Section 3.9.3.4.2.

3.7.3.8.4 Seismic Analysis of Piping Systems That Span Two or More Seismic Support Zones Such as Buildings, Portions of Buildings, or Primary Components

Each building, portion of building, or primary component may be considered a separate support zone. The worst enveloped response spectrum for which any portion of the pipe located in that zone is subjected is used to represent the input motion in that zone.

For the evaluation of relative support motions in the seismic analysis of piping systems interconnecting two or more seismic support zones, the maximum relative movement between component supports is assumed and the piping system is subjected to movements through the piping system supports and restraints. Separate cases for each of the three orthogonal directions are considered. Support movements are based on the maximum of the floor movements immediately above and below the support location.

3.7.3.9 Multiple Supported Equipment and Components with Distinct Inputs

3.7.3.9.1 Other Than NSSS

The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are similar to those described for piping in Section 3.7.3.8.4. When the equipment is supported at two or more points located at different elevations in the building, the response spectrum for the most severe single point of attachment is chosen as the design spectra.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the additional stresses due to support displacements.

3.7.3.9.2 Multiple Supported NSSS Equipment and Components with Distinct Inputs

When response spectrum methods are used to evaluate reactor coolant system primary components interconnected between floors, the procedures of the following paragraphs are used. There are no components in Westinghouse scope of analysis which are interconnected between buildings. The primary components of the reactor coolant system are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis is the worst floor response spectra. Any deviation from this position will be subject for NRC review on a case-by-case basis. The reactor coolant loop analysis described below only includes response spectra at the basemat.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis.

Per ASME Code rules, this stress caused by differential seismic motion is clearly secondary for piping (NB 3650) and component supports (NF 3231). For components, the differential motion will be evaluated as a free end displacement, since, per NB 3213.19, examples of a free end displacement are motions 'that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping'. The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB 3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses. Since the reactor coolant loops and primary supports are coupled with the reactor building interior concrete structure in the reactor coolant loop seismic analysis, there is no separate consideration of seismic anchor motions.

3.7.3.10 Use of Constant Vertical Load Factors

3.7.3.10.1 Use of Constant Load Factors for Equipment Other Than NSSS

With respect to equipment, static analysis for seismic loading is recognized as an acceptable approach with restrictions as follows:

1. The analysis method is consistent with the 'static coefficient method' as prescribed in IEEE 344-1975, Paragraph 5.3. The peak acceleration values of the applicable floor response spectra are multiplied by a factor of 1.5 if natural frequencies are not determined. The increased acceleration values are used as equivalent static load factors applied to the entire mass of the equipment being evaluated. Lower multiplication factors (between 1.0 and 1.5) are only used as justified by frequency analysis.

2. The static coefficient analysis method is used only for the evaluation of structural integrity of equipment. It is recognized that the static analysis method alone is not sufficient for the qualification of safety-related active equipment where the demonstration of operability is required.

3.7.3.10.2 Use of Constant Vertical Load Factors for NSSS

Constant vertical load factors are not used as the vertical floor response load for the seismic design of NSSS safety-related systems and components.

3.7.3.11 Torsional Effects of Eccentric Masses

3.7.3.11.1 Piping Other Than NSSS

The torsional effects of eccentric masses such as valve operators are modeled in the piping mathematical model as lumped masses at the free end of cantilevered rods with a length equal to the distance from the center of gravity of the mass to the pipe flow axis. The stiffness of the rod is used to simulate the valve extended structure flexibility.

3.7.3.11.2 Torsional Effects of Eccentric Masses of NSSS

The effect of eccentric masses, such as valves and valve operators, is considered, when applicable, in the seismic piping analyses. These eccentric masses are modeled in the system analysis and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of piping.

3.7.3.12 Buried Seismic Category I Piping Systems

Buried piping complies with the ASME Boiler and Pressure Vessel Code, Section III and is analyzed seismically as follows:

> The soil is considered to be a horizontal 1-layer system which responds to the earthquake by moving in a continuous sinusoidal plane wave and supported by a second layer or base material. The top layer is assumed to pick up accelerations from the base material.

Utilizing the average values for the shear wave velocity and density for the top layers, the ground deformation pattern in terms of wave length and amplitude is determined. The buried pipes are assumed to deform along with the surrounding soil layers.

The average shear wave velocity of a single layer representation of a multi-layered soil system may be determined by:

$$V_{ST} = \frac{\Sigma V_S h'}{h}$$

where,

V_{ST} = Average shear velocity in the top layers of soil, ft/sec
V_S = Shear velocity in each layer of soil, ft/sec
h' = Depth of each layer of soil, ft

h = Total depth of top layers of soil, ft

The fundamental period of the single layer is calculated from the following equation:

$$T = \frac{4h}{V_{ST}} \text{ (seconds)}$$

If the depth of the soil layer varies over the distance traversed by the buried pipe, both cases, for maximum and minimum depths, are considered.

The maximum amplitude of the sine wave which represents the maximum displacement of the pipe is:

$$A = \text{Displacement} = \left(\frac{T}{2\pi}\right)^2 * (Accel)$$

Where:

T = Fundamental period, sec
Accel = Amplified soil acceleration value, in/sec^2

The wave length, L, is calculated as:

$$L = V_{ST}\ T$$

The bending moment resulting from the seismic disturbance, assuming the pipe follows the soil and deforms as a sine wave, is given by

$$M = \frac{\pi^2 EIA}{(L/2)^2}$$

Where:

M = Maximum bending moment, in-lb
E = Modulus of the pipe, psi
I = Moment of inertia of the pipe, in4
A = Maximum amplitude, in.
L = Wave length, in.

The corresponding bending stress is obtained by dividing the moment by the section modulus of the pipe. The above bending stress is combined with bending stresses due to other loads according to the applicable loading combinations.

The axial strain experienced by the pipe due to deformation of the soils is also evaluated. The axial strain due to seismic propagating waves is computed following the methods of Newmark,[15] and [16] Yeh,[18] and Keusel,[17] which assume the soil is linearly elastic and homogenous, the pipe behaves as a slender beam, and the buried member deforms with the surrounding soil (this implies the strain in the soil equals the strain in the member).

The effect of soil strain from a seismic event on elbows or turns in a buried pipe system must be analyzed in greater detail than just calculating the axial strain. The effect of these strains on elbows/turns is more complex due to the pipe elbow/turn trying to resist the strain. The complexity is a function of the pipe and backfill soil properties.

The basis for determining the effect of the strains on the piping elbows/turns is described by Shah and Chu.[19] The Shah and Chu theory has been developed into an analysis procedure by Goodling.[20], [21], [22] The committee on Seismic Analysis of the ASCE Structural Committee on Nuclear Structures and Materials prepared a report "Seismic Response of Buried Pipes and Structural Components,"[23] which explains and amplifies the referenced methodology[19] and analysis procedure.[20], [21],[22] These references shall be used for analysis of the effects of axial strain on buried piping.

The magnitude of friction acting on the pipe used in the analysis depends on several factors, such as pipe surface conditions, contact pressure, soil strengths, etc. The friction force acting on the pipe is determined in accordance with Reference [24].

<u>Differential Movement</u>

Differential movement between the piping and a structure/feature occurs from two sources. The first is vertical, which can be caused by differential soil consolidation below the pipe or structure/feature. The second source is horizontal movement due to differential movement during a seismic event.

Where practical, seismic classed buried piping is routed to avoid areas of weak soils. Where weak soils are encountered, the bad material is removed and replaced by backfill. The backfill is placed to standards that ensure suitable bearing conditions; therefore, the transition from one material to another, i.e., in situ soil to backfill, should not be a problem. In lieu of the above, in some cases an analysis is performed to show that the pipe has sufficient strength to bridge the discontinuity and support the soil above the pipe without exceeding the allowable stress of the piping material.

Category I buried piping which penetrates structures where fill settlement or seismic movements are expected to be high is protected from differential movement of the soil and structure by Category I concrete slabs or encasements. The slab or encasement is supported by a bracket on the structure on one end and on undisturbed or Class A backfill at the other end. Bearing piles are used if required to support the slab. The encased pipes are insulated to prevent bonding between the pipes and concrete. For details of the slab at the intake pumping station and the encasement at the Diesel Generator Building, refer to Section 3.8.4.4.

For seismic classed buried piping that penetrates structures in areas where very little fill is involved and seismic movements are low, protection from differential movement of the soil and structure is provided by an oversized opening in the structure. The annular space between the pipe and opening is filled with a resilient material. The first support inside the structure is located to allow for relative movement of the pipe and structure. The soil- structure interface is treated as an anchor, and stresses are limited to code allowables.

Soil consolidation is determined in conformance with criteria given in Section 2.5.4.10 (static settlement) and 2.5.4.8 (dynamic settlement - soil liquefaction).

The ERCW piping was evaluated for potential settlement due to soil liquefaction as discussed in Section 2.5.4.8. The potential settlements used for the evaluation were determined in the liquefaction evaluation using the strain criteria specified by the NRC staff which are shown on Figures 2.5-571 through 2.5-575. The effect of these potential settlements was evaluated for the entire length of pipe and also at all building interfaces. The evaluation of the effect of these potential settlements was done in two phases.

The first phase was a preliminary screening which involved calculations to identify areas of the pipe which may undergo excessive settlement. In the preliminary screening, the boundaries of the pipe system, the pipe sizes, and pipe materials were determined. Because of the size and length of pipe involved, a 60 foot length was chosen as sufficient to model the system. A fixed-fixed end model was assumed to describe the piping for the initial calculations. Using the standard equation for maximum deflection for a fixed-fixed end model:

$$Y_{max} = \frac{ML^2}{32EI}$$

M = Resultant moment
L = Span length
E = Young's modulus
I = Moment of inertia

The settlement can be determined if the resulting-moment were known. ASME Code Section III (1977 edition) states that the effects of any single nonrepeated anchor movement is governed by Equation 10A:

$$\frac{iM}{Z} \leq 3.0S_c$$

i = Stress intensification factor
Z = Section modulus
S_c= Allowable stress at room temperature

To expand this equation to include thermal effects (assuming Mc = 0) would involve adding it to Equation 11 (1971 ASME III Code, Summer 1973 Addenda, NC-3652.3) thus;

$$\frac{iM}{Z} \leq 3.0\ S_c + S_A$$ S_A = Allowable stress for expansion

Since the pipe sizes and materials are known, and the stress intensification factor can be calculated, the resultant moment at any point on the pipe can be determined. Thus the potential settlement can be found by using the standard equation for the fixed-fixed end model. The results from these preliminary screening calculations were used in conjunction with the potential settlement evaluation, Section 2.5.4.8, to identify potential areas of excessive settlement, either at the buildings or along the pipeline.

The second phase of the evaluation consisted of making rigorous piping analyses at the potential areas of excessive settlement. There were three areas along the pipeline with apparent problems that were modeled into the TPIPE piping analysis program. These areas were modeled for a distance on both sides of the potential high settlement area. The areas that were modeled were: (1) from the intake pump station to boring SS-131; (2) from boring SS-141 to boring SS-90; and (3) from boring SS-163 to boring SS-159.

At these areas the potential settlements were used as input in the phase II analysis to give the most conservative results. In all cases, the stress levels are below the ASME Code allowable for settlement induced loads (Reference 1977 ASME Code).

Cement-mortar lined carbon steel pipe is used in the buried portion of the ERCW yard piping system. The reason for the mortar lining is given in Section 9.2.1.6. The seismic qualification of the cement-mortar lining is provided by testing. This testing is described below.

A full-scale testing program consisting of laboratory tests, field tests, and vibration measurements was conducted for seismic qualification of the cement-mortar lined carbon steel pipes. A total of 100 feet of 30-inch diameter pipe, 20 feet of 18-inch diameter pipe, and a 90-degree elbow of 30-inch diameter were lined. Pipe sections tested were: one 30-foot pipe of 30-inch diameter, one 40-foot pipe of 30-inch diameter, one 90-degree elbow of 30-inch diameter with a 5-foot pipe welded to each end, 14 two-foot sections of 30-inch diameter, and 10 two-foot sections of 18-inch diameter. Cement-mortar samples were taken from the mixer before lining application began. Density and moisture content tests were performed on the compacted backfill material surrounding the pipe for field tests. Lining materials and procedures were conforming to American Water Works Association Standard C602-76, 'Cement-Mortar Lining of Water Pipelines - 4 Inches and Larger - In Place.'

Cement-mortar specimens were tested for compressive, tensile, and flexural strength, modulus of elasticity, and density. The two-foot pipe sections were subjected to three-edge-bearing, cyclic loading, torsion, drop, and impact tests. The 30-foot pipe was subjected to bending, cyclic loading, and drop tests. The 90-degree elbow was subjected to bending tests. The 40-foot pipe was installed in a trench and after backfilling it was subjected to a dynamic loading of 36,000 pounds at 28 hertz (Hz) from a vibratory roller with a smooth drum of 60-inch diameter by 84-inch width. Two accelerometers were mounted on two of the 30 inch pipes to monitor vibrations experienced by the pipes during the 100-mile trip from the Phipps Bend construction site near Kingsport, Tennessee, to Singleton Materials Engineering Laboratory near Knoxville, Tennessee. The vibrations of the 30-foot pipe (bottom) and a two-foot section (top) were measured and recorded on tape for later analyses. It was expected that the difference in dimension and difference in physical location of the pipes would result in different vibration magnitudes and frequency contents. Comparison between the recorded vibrations and the design earthquake was also made.

The acceleration time histories and their corresponding Fourier amplitude spectra at certain high acceleration locations on the record were processed. The acceleration time histories are

recorded data and the Fourier amplitude spectra are calculated from the recorded data. This transform of data from time domain to frequency domain reveals the frequency content of the vibration data. The maximum acceleration experienced by the bottom pipe (30 feet long) was 0.6g and that experienced by the top pipe (two-foot section) was 2.1g. Both values are higher than the SSE accelerations for the design of TVA nuclear plants. The recorded maximum peak-to-peak accelerations were 1.2 g and 3.8 g, respectively. Dominant frequencies ranged from 15 to 70 Hz, mostly concentrated in the range of 15 to 50 Hz.

For most large earthquakes the dominant frequencies are in the range of 0.5 to 10 Hz. Lower frequencies indicate that a buried pipe would experience less number of cycles of vibration during real earthquakes. Since a pipe has to move with its surrounding soil, vibration amplification due to structure properties is minimal.

No crack due to vibration was found in any of the lining after unloading. It is concluded that the linings had experienced more severe vibrations than any recorded earthquakes in terms of magnitude and number of cycles. The vibration measurements were considered as effective as shaking table tests.

The three-edge-bearing tests showed that the cement-mortar linings were flexible. The lining underwent considerable cracking prior to separation and falling of the linings. Linings only fell after the formation of the plastic hinges in the steel.

The testing program covers a much broader range in types of loading than earthquake loadings. They simulated dead load (loading from roller without vibration, three-edge-bearing test, torsion, and bending tests), low frequency load (cycle tests), large dynamic load at 28 Hz (loading from roller with vibration), large acceleration load with a major frequency content of 0-100 Hz (vibration measurements during shipping), line load with very short duration (drop test), and point load with very short duration (impact test).

From these tests, it is concluded that the test loadings applied to the cement-mortar lining were much more severe and broad-ranged than the design seismic loadings. Therefore, the cement-mortar lining in the underground ERCW pipes is seismically qualified.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

The analysis of a Category I piping system may be terminated at the interface of a nonnuclear safety class piping run by either of the following methods.

1. Terminate the analysis at an in-line anchor designed to prevent transfer of rotations and deflections. The design of the anchor will be sufficient to accommodate reactions from all adjacent piping runs.

2. Extend the analysis and support of the Category I system far enough into the nonnuclear safety class system to ensure that the effects of this adjacent system have been imposed on the Category I system.

Normally, a valve serves as a seismic-nonseismic boundary in a fluid system. The valve capability to maintain a pressure boundary in the event of a seismic event is assured by seismically designing piping on the nonclassified side as described above.

3.7.3.14 Seismic Analyses for Fuel Elements, Control Rod Assemblies, Control Rod

Drives, and Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling. The time history floor response based on a standard seismic time history normalized to SSE levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in References [7] and [9].

The Control Rod Drive Mechanisms (CRDM) are seismically analyzed to confirm that system stresses under seismic conditions do not exceed allowable levels as defined by the ASME Boiler and Pressure Vessel Code Section III for 'upset' and 'faulted' conditions. Based on these stress criteria, the allowable seismic stresses in terms of bending moments in the structure are determined. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation, and the resultant seismic bending moments along the length of the CRDM are calculated. These values are then compared to the allowable seismic bending moments for the equipment, to ensure adequacy of the design.

The seismic qualification of Watts Bar reactor vessel internals is demonstrated using a generic basis for a four loop plant. The generic basis or analysis consists of generic design response spectra and generic reactor vessel supports which envelope the analogous specific Watts Bar values.

The generic seismic analysis of the reactor internals is conducted in accordance with the guidelines specified in Regulatory Guide 1.92. The seismic analysis determines the response of the reactor internals to OBE and SSE vertical and horizontal seismic shock components. The horizontal and vertical seismic analysis use the modal response spectrum method and the WECAN general purpose finite element program to determine the internals response. The method used to obtain the combined response of the modal spectral responses is square-root-of-the-sum-of-the squares (SRSS).

The effect of closely spaced modes is considered using the Ten Percent Method (Regulatory Guide 1.92, Paragraph 1.2.2); however, the effect has been shown to be insignificant. The maximum or total seismic response value of the reactor internals is obtained by taking the SRSS of the maximum values of the co-directional responses due to the three components of earthquake motion. In general, this combination is made in the Stress Analysis section of the particular structural component.

When appropriate (e.g., simple beam analysis) LOCA and SSE loads are combined on a reactor internals structural component basis per the SRSS method, the resultant stress intensities calculated. For more complex structural geometries (e.g., core barrel shell) the stress components due to LOCA and SSE are combined either by absolute sum or SRSS, preserving the appropriate signs. These stress components are used to determine the stress intensity for the structural component. For the LOCA, the maximum stresses from the time history response are used. Since the seismic stresses are calculated using response spectrum techniques, the responses are unsigned; therefore, when the LOCA and SSE stresses are combined, the most unfavorable sign convention for the SSE is assumed. The horizontal and vertical seismic models contain 118 and 27 active dynamic degrees of freedom, respectively. Results from the modal analysis of the horizontal and vertical systems indicates, in general, 17 and 3 modes

present with frequencies less than 33 Hz.

In developing the seismic model of the reactor vessel and internals, a systematic approach was used to ensure that basic fundamental frequencies, i.e., both component and system frequencies are described and inherent in the mathematical models. The approach used to verify the mathematical modeling of reactor vessel and internals was to compare and require that the system frequencies and mode shapes from the mathematical models to be in agreement with plant test and scale model test data.

In determining the seismic response of the reactor system due to the excitation of unidirectional shock spectrum, those modes contributing to the first 80-90% of total system mass was considered in the solution.

Hydrodynamic mass effects, for both horizontal and vertical directions, was included in the reactor vessel-internals system models. The numerical values for the various hydrodynamic masses effects within the reactor system is based on scale model and plant tests and applicable analytical expressions, e.g., Fritz, Fritz & Kiss, etc.

The effect of significant nonlinearities in the reactor system, i.e., gaps between reactor vessel and internals on the seismic response is considered in the system analysis. The nonlinearities due to the gaps are included by determining an effective stiffness at the gap location. The validity of this approach has been investigated and found to be conservative for the frequency response range of the reactor internals.

The structural damping values used in the system seismic analysis are in accordance with Regulatory Guide 1.61; i.e., 2 and 4 percent for OBE and SSE, respectively.

In addition, the stiffness of the primary piping and the stiffness of reactor vessel supports are considered in the analysis. Coupling effects between the horizontal and vertical directions are insignificant and are not considered in the analysis.

The frequency response for the Watts Bar reactor vessel internals system is enveloped by the frequency response of the four loop reactor internals which uses the generic vessel support stiffness. The generic frequency response of four loop reactor internals results in acceleration values on the generic response spectra curve. The generic spectra envelopes the specific Watts Bar spectra by a considerable margin and therefore, the loads for the four loop generic analysis envelope the loads for Watts Bar. Consequently, seismic qualification of the Watts Bar reactor internals is demonstrated since the four loop reactor internals have been qualified on a generic basis.

3.7.3.15 Analysis Procedure for Damping

The specific percentages of critical damping value used for Category I structures, systems, and components are provided in Tables 3.7-2 and 3.7-24.

3.7.3.16 Seismic Analysis and Qualification of Category I Equipment Other Than NSSS

All seismic Category I floor or wall-mounted mechanical and electrical equipment was analyzed or tested and designed to withstand seismic loadings in the horizontal and vertical directions. The floor response spectra obtained from the analysis of structures were used in the analyses. Each procurement specification for equipment contained the particular floor response spectra curve for the floor on which the equipment is located. Depending on the relative rigidity and/or the complexity of the equipment being analyzed, the vendor could use one of the following four methods to qualify the equipment:

1. Dynamic analysis method,
2. Simplified dynamic analysis method,
3. Equivalent static load method,
4. Testing method.

The basis used for selection of the appropriate accelerations used in the above paragraph is described in further detail in Section 3.7.3.16.2. Table 3.7-25 identifies how each Seismic Category I item was qualified.

Equipment is considered to be rigid for seismic design if the first natural frequency is equal to or more than 33 cycles per second.

The Watts Bar Category I electrical and mechanical equipment seismic qualification program is consistent with the guidance provided by the NRC Standard Review Plan (NUREG-0800), Revision 2, July 1981, Section 3.10, acceptance criteria for plants with Construction Permit applications docketed before October 27, 1972. The equipment has been seismically qualified either in direct compliance with IEEE Std. 344-1975/Regulatory Guide 1.100 (equipment procured after September 1, 1974), or in accordance with a program which provided as a minimum, qualification to the requirements of IEEE 344-1971 and in addition addressed the guidelines of SRP 3.10.

3.7.3.16.1 Dynamic Analysis Method For Equipment and Components

Equipment that is rigid and rigidly attached to its support structure was analyzed for a g-loading equal to the acceleration of the supporting structure at the appropriate elevation.

For nonrigid, structurally simple equipment, the dynamic model consisted of one mass and one spring. Keeping the values of the mass and the spring constant, the natural period of the equipment was determined. The natural period, together with the appropriate damping value, was used to enter the appropriate acceleration response spectrum to obtain the equipment acceleration in units of g's. The corresponding inertia force was obtained by multiplying the weight times the acceleration.

If the equipment is structurally complex to the extent that a single-degree-of-freedom-system model does not adequately represent the action of the structure to dynamic loads, then a multi-degree-of-freedom model was used with a complete multi-degree-of-freedom analysis. Enough modes were considered to adequately represent the response of the equipment.

3.7.3.16.2 Simplified Dynamic Analysis Method For Equipment and Components

In the simplified dynamic analysis method, the acceleration value corresponding to the maximum shown on the response spectrum curve is used in qualifying the equipment. The forces on the equipment are determined by multiplying the equipment weight times the acceleration. This provides an acceptable method of analysis providing one of the following criteria is met:

1. The item of equipment is simple enough to be adequately modeled by a simple one-degree-of-freedom spring-mass system.

2. The item of equipment is not simple but its fundamental frequency is greater than the rigid frequency. The rigid frequency is defined as that frequency of the floor response spectrum above which there is no acceleration amplification.

3. The item of equipment is not simple and its fundamental frequency is lower than the rigid frequency but its other frequencies are higher than the rigid frequency.

If the equipment can be shown to meet one of these criteria, any amplification due to internal dynamics will not cause stresses greater than those obtained by using the peak value of the floor response spectrum. All of the equipment listed in Table 3.7-25 as having been analyzed by the simplified dynamic analysis method has been reviewed to verify that it meets one of these criteria.

The method described above is conservative since the maximum acceleration, regardless of the frequency of the equipment, is used.

3.7.3.16.3 Equivalent Static Load Method

The description of equivalent load method and its applicability are detailed in Section 3.7.3.5.

3.7.3.16.4 Testing Method

Equipment that did not lend itself to mathematical modeling and structural analysis to determine no loss of function was evaluated by actual vibration testing. The seismic qualification of mechanical equipment, instrumentation and electric equipment are described in Sections 3.9 and 3.10, respectively.

3.7.3.16.5 Equipment and Component Mounting Considerations

Seismic loads for vendor-supplied floor or wall mounted Category I equipment and fluid system component (equipment/component) assemblies and their TVA-designed supports and/or anchorages are determined with consideration of the damping values and stiffness of each. Damping values for these equipment/component assemblies and their bolted or welded structural steel supports and/or anchorages are as indicated in Table 3.7-2. Most of the TVA-designed supports and/or anchorages are effectively rigid; e.g., they do not result in significant amplification of the building structure seismic input. When a TVA-designed support and/or anchorage is not effectively rigid a coupled analysis of the equipment and/or component assembly and its support and/or anchorage is performed using composite modal damping response spectrum analysis techniques.

Examples of vendor-supplied floor or wall mounted mechanical equipment/ component assemblies include: tanks, heat exchangers, diesel generator sets, air handling units, chiller units, compressor assemblies, fan assemblies, and pumps. Electrical equipment assemblies include: transformers, battery racks, instruments and control (I/C) cabinets, I/C panels, and I/C racks.

Seismic loads for line-mounted Category I equipment/components and their mountings are determined from analysis of the subsystems on which they are mounted. The line-mounted equipment/component is tested or analyzed using device qualification techniques as described in Section 3.7.3.6.3. Mass and stiffness characteristics of the equipment/components are included in the subsystem analysis when significant to its seismic response. For example, Section 3.7.3.11.1 describes the modeling of valves in Category I piping subsystems. The subsystem response at the equipment/component location is kept below the device qualification level of the equipment/component. Local mounting brackets for line-mounted equipment/components are seismically qualified with the equipment/component (as part of the device) or they are designed to be effectively rigid. In this case, effectively rigid means the local mounting brackets do not result in significant amplification of the seismic input from the subsystem.

Examples of line-mounted mechanical and electrical equipment/components include: valves, HVAC dampers, and locally-mounted I/C devices of all types.

The techniques described in this section ensure compatibility of the seismic loads for qualification of the Category I equipment/components and the predicted seismic responses of structures and subsystems to which they are mounted.

3.7.3.17 Seismic Analysis and Design of HVAC Duct and Duct Support Systems

This section addresses the analysis and design of Category I and I(L) (see Sections 3.2.1 and 3.2.2.7) HVAC duct and duct support subsystems.

3.7.3.17.1 Description of HVAC Duct and Duct Support Subsystems

HVAC duct and duct support subsystems consist of continuous runs of round and rectangular sheet metal ducts multiple supported along their lengths by structural steel support frames or rod hangers. Scheduled pipe and pipe supports functionally used for an HVAC purpose are treated as piping subsystems in accordance with Section 3.9.

For purpose of analysis, an HVAC duct and duct support subsystem is regarded as any continuous portion of a total duct run and its supports which may be conservatively modeled for evaluation of the loads and stresses within the portion of interest. Significant mass and mass eccentricities of in-line or attached mechanical and electrical components are accounted for in the subsystem model to represent their effects in structural qualifications of the ducts and duct supports in accordance with Sections 3.7.3.17.2 through 3.7.3.17.6. Qualification of the in-line or attached Category I mechanical or electrical equipment and components are in accordance with Sections 3.7.3.6, 3.7.3.16.5, 3.9, and 3.10.

3.7.3.17.2 Applicable Codes, Standards, and Specifications

The following codes, standards, and specifications are applicable to various portions of the HVAC duct and duct support subsystems:

1) SMACNA High Velocity Duct Construction Standards, 2nd Edition, 1969

2) ANSI/ASME N509 Standard, "Nuclear Power Plant Air Cleaning Units and Components," 1976

3) ASTM Standards

4) AISI Specifications for the Design of Cold-Formed Steel Structural Members, 1986 Edition

5) AISC Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th and 8th Editions except welded construction is in accordance with Item 7 below.

6) Manufacturer's Standardization Society of the Valve and Fittings Industry, Standard Practice MSS-SP-58, "Pipe Hangers and Supports - Materials and Design," 1967 Edition

7) American Welding Society, AWS D1.1 Structural Welding Code (See Section 3.8.1.2, Item 4)

8) American Welding Society, AWS D1.3 Structural Welding Code for Sheet Metal

9) American Welding Society, AWS D9.1 Specifications for Welding Code for Sheet Metal

10) NRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2.

3.7.3.17.3 Loads and Load Combinations

HVAC duct and duct support subsystems are designed for the following loads:

DL - Dead loads

OBE - Operating basis earthquake loads

SSE - Safe shutdown earthquake loads

To - Thermal effects and loads during normal operating or shutdown conditions based on the most critical transient or steady-state conditions

Ta - Time varying thermal loads under conditions generated by the design basis accident condition and including To

Note: The maximum value of Ta need not be considered simultaneously with the DBA if time phasing evaluation shows that less than Ta maximum occurs during the DBA transient.

Po - Operating pressure in the duct

Pj - Accident pressure external to the duct due to jet impingement loads from a pipe break. The ducts shall be protected against possible Pj loadings; therefore, this load need not be considered.

Pa - Compartmental pressure loads resulting from a design basis accident.

DBA - Design basis accident dynamic loads due to pressure transient response

F - Airflow induced dynamic loads acting on turning vanes inside the ducts (dependent on the mean airflow velocity). These loads are considered in the following combinations for the duct and duct support elements of the subsystems:

Ducts	Duct Supports
(1) DL + Po + F + OBE	(1) DL + OBE
(2) DL + Po + To + F + OBE	(2) DL + To + OBE
(3) DL + Po + To + F + SSE	(3) DL + To + SSE
(4) DL + Po + Ta + F + OBE + DBA + Pa	(4) DL + Ta + OBE + DBA
(5) DL + Po + Ta + F + SSE + DBA + Pa	(5) DL + Ta + SSE + DBA

3.7.3.17.4 Analysis and Design Procedures

Existing HVAC duct and duct support subsystems that were originally analyzed and designed to Set A seismic response spectra are reevaluated to Set B response spectra as the basis for their qualification. New designs and modification designs to existing subsystems are based on the envelope of Set B+C response spectra.

3.7.3.17.5 Structural Acceptance Criteria

The various elements of the HVAC duct and duct support subsystems are qualified for structural acceptance based on allowable stress criteria.

Allowable stresses for the duct supports are specified in Table 3.7-26.

Allowable stresses for the ducts involve a number of specialized considerations to address both overall and local stresses. Overall stress allowables for duct plate (membrane) elements are developed based on AISI equations. These equations are modified where necessary to adjust for large height-to-thickness and width-to-thickness ratios beyond the normal AISI limits. These adjustments are based on correlations to results of testing, large displacement finite element analyses, and/or industry literature. Additional specialized considerations are made for local stress evaluations. Stress evaluations of the duct stiffeners (including companion-angles) and bolting between these stiffeners are based on AISC allowables. Stress evaluations of the tinners rivets connecting the companion-angles to the duct plate are based on correlation to test results.

In general, unfactored duct stress allowables are used in evaluations of loading combination (1). These stress allowables are multiplied by 1.5 in evaluations of loading combinations (2), (3), (4), and (5). Critical elements of the duct necessary to maintain overall cross section stability are limited to 0.90 Fy except shear is limited to 0.52 Fy and buckling is limited to 0.90 Fcr. Local plate stresses are maintained within 0.90 Fy for mid-plane membrane stresses although surface stresses may exceed yield. The effective cross section of a duct is evaluated based on the post-buckled membrane strength of the duct panels between stiffeners.

3.7.3.17.6 Materials and Quality Control

Some HVAC sheet metal materials installed prior to March 1990 were not always specified and controlled sufficiently to assure known mechanical properties. Samples of these materials were taken from the installed ducts and tested to determine their mechanical properties. The following mechanical properties are used for designs with these materials:

Duct Construction Type	Yield Strength, Fy	Tensile Strength, Fu
SMACNA rectangular (ASTM A525/A527 galvanized sheet)	33 ksi	45 ksi
Specially formed round or rectangular welded (ASTM A570 sheet)	30 ksi	49 ksi
Spiral-welded pipe (ASTM A211)	30 ksi	40 ksi
SMACNA round spiral-lock or longitudinal-lock	20 ksi	37 ksi

HVAC duct sheet metal materials specified after March 1990 and the associated mechanical properties used for designs are as follows:

Specified Material	Yield Strength, Fy	Tensile Strength, Fu
ASTM A527 galvanized steel sheet with ASTM A446 Grade A(minimum) base metal	33 ksi	45 ksi
ASTM A570 Grade 30 (minimum) steel sheet (also used for ASTM A211 spiral-welded pipe)	30 ksi	49 ksi

HVAC structural steel supports are fabricated of ASTM A36 or equivalent or stronger material and are evaluated as having mechanical properties of Fy=36 ksi and Fu=58 ksi.

All steel materials used in the fabrication of HVAC ducts and duct supports are evaluated with a Young's Modulus of E=29 x 10^3 ksi except for those areas within the Reactor Building where reductions must be taken due to extreme accident thermal conditions.

3.7.3.18 Seismic Qualification of Main Control Room Suspended Ceiling and Air Delivery Components

Flexible ducting, triangular ducts, and air bar linear diffusers deliver air flow from the sheet metal ducts located above the Main Control Room (MCR) suspended ceiling to the air space below the ceiling. These air delivery components have been seismically qualified to ensure position retention and structural integrity such that pressure boundary and air flow delivery is maintained during and after the Safe Shutdown Earthquake (SSE).

Seismic qualification of the suspended ceiling and the air delivery components has been accomplished by rigorous time history analysis using the ANSYS computer code. The analysis models non-linear response due to gaps, friction, ceiling support wires, and geometric effects of the ceiling grid work. The seismic time histories correspond to the Control Building response to the Set B SSE at the floor elevation above the suspended ceiling. The combined time histories were then adjusted to account for ±15 percent frequency uncertainty. A factor of safety of at least 1.3 for seismic qualification of the ceiling and air delivery components was demonstrated by increasing the time history motions by 30 percent and verifying that the seismic demand is less than the capacity of the ceiling grid members (including air bars), support wires, and flexible and triangular ducts. The ceiling grid member and support wire capacities are based on classical structural analysis formulas. The flexible and triangular duct capacities were based on analysis for potential failure modes, industry precedents, and the analytical determination that the ceiling grid work remains stable. Other suspended ceiling components, including luminous panels, were shown to retain their position during and after the SSE.

3.7.4 Seismic Instrumentation Program

Seismic instrumentation is provided in order to assess the effects on the plant of earthquakes which may cause exceedance of the Operating Basis Earthquake (OBE = 0.09g horizontal and 0.06g vertical ground acceleration). The seismic monitoring system (SMS) is not safety-related, nor does it have any effect on safety-related systems or components. The components of the SMS are selected to emphasize accuracy and reliability. The instrumentation program is described in the following subsections.

3.7.4.1 Comparison with Regulatory Guide 1.12

The instrumentation is described in Section 3.7.4.2 below and meets the requirements of Regulatory Guide 1.12.

3.7.4.2 Location and Description of Instrumentation

The seismic instrumentation locations are shown in Figures 3.7-39 through 3.7-45.

Instrumentation consists of the following:

1. A strong motion triaxial accelerometer at each of the following locations:

   a. Elevation 702.78, Unit 1 Reactor Building, on the floor slab in the annulus between the Shield Building and the Steel Containment Vessel as shown in Figure 3.7-39.

   b. Elevation 756.63, Unit 1 Reactor Building, on the floor slab as shown in Figure 3.7-40.

   c. Elevation 742.0, Diesel Generator Building, on the base slab as shown in Figure 3.7-41.

   These accelerometers are connected to digital recorders (See Item 3). The recording system is located in the Control Building. The full scale range of the transducers is 0 to 1.0g with a bandwidth of 0 Hz to 50 Hz and a temperature effect of less than 2% per 100°F change.

2. A triaxial strong motion accelerograph with a range of 0g to 2g at Elevation 757 in the Auxiliary Building contains an internal battery backup and is capable of digitally recording a minimum of 25 minutes of data with a minimum of 3 seconds of pre-event memory. An internal seismic trigger with a bandwidth of 0.1 to 12.5 Hz actuates the recording system when a threshold acceleration level is sensed.

3. A seismic instrumentation panel board located at Elevation 708 in the Control Building as shown in Figure 3.7-42. The panel board houses a centralized SMS consisting of a recorder panel, a central controller assembly, a display panel, an alarm panel, and a printer panel. A description of each item mounted on the panel board is given below.

   a) Two recorder panels containing a total of three digital recorders capable of 18-bit resolution. The three strong motion accelerometers of Items 1a, 1b, and 1c above provide input to the recorders. Each digital recorder contains three channels and is capable of recording a minimum of 25 minutes of data with a minimum of 3 seconds of pre-event memory. Each recorder has an internal trigger with a bandwidth of 0.1 to 12.5 Hz which constantly monitors its interconnected triaxial accelerometer. When one of the recorders senses a seismic event, an interconnected network causes the other recorders to trigger and record data at the same time to ensure time-synchronized event-data files. The trigger threshold is set to initiate recording when the acceleration at the containment foundation exceeds 0.01g. A signal is also sent to the alarm panel to indicate that the system is recording (See Item 3c). The recorders can operate for up to 36 hours on internal batteries.

   b) A central controller consisting of an industrial computer and custom software which provides a user interface in a multi-task operating system that supports simultaneous acquisition and interrogation. The controller is powered by 120V AC power.

The central controller retrieves data files from the digital recorders after an event and performs automatic analysis on the data. The event analysis capabilities include calculation of the spectral content of the recorded data and comparison to the site OBE design basis response spectrum. The results of the analysis are displayed on the LCD display panel, sent to a printer, and saved to disk for later off-line analysis. The central controller's software capabilities also include automatic event alarm and annunciation, as well as configurable built in tests of the components comprising the centralized system.

c) An alarm panel containing visual alarms to locally indicate that a seismic event has been recorded, that the OBE site design response spectrum has been exceeded in a damaging frequency range, and to indicate either loss of AC or DC power. The seismic event alarm is triggered by the recorder panels; while the OBE exceedance alarm (See Item 4) is triggered by the central controller. Activation of either the event alarm or exceedance alarm also causes corresponding windows on an annunciator panel in the Main Control Room to illuminate.

d) A display panel to provide a visual display for operation of the centralized system.

e) A printer panel to provide a permanent copy of operational data and event analysis results.

4. Annunciator lights mounted on a window box located on Panel 1-M-15, Main Control Room, Control Building, as shown by reference in Figure 3.7-43. The messages displayed on the annunciator windows in the Main Control Room are 'Seismic Recording Initiated,' 'OBE Spectra Exceeded,' and 'Seismic Instrumentation Loss of Power.'

The basis for the selection of the Reactor Building for installation of seismic instrumentation is that it is the rock-supported building most important to safety. The basis for the selection of the Diesel Generator Building is that it is the soil-supported building most important to safety. The basis for the selection of the Auxiliary Building is that it is a rock-supported structure outside containment.

Steps for utilization of the data recorded by the above described instrumentation are provided in Sections 3.7.4.4 and 3.7.4.5 below.

3.7.4.3 Control Room Operator Notification

The operator receives three annunciation signals in the Main Control Room.These annunciations are independent of each other. The first annunciation is ‘Seismic Instrumentation Loss of Power,’ which serves to provide warning of equipment operability problems under normal conditions as well as following a seismic event. The next annunciation is provided by the recorder panel described in Item 3a, Section 3.7.4.2, which informs the operator that a seismic event is being recorded. This annunciation indicates that one of the triggers for the digital recorders sensed seismic motion in excess of 0.01g.

The final annunciation signal (‘OBE Spectra Exceeded’) is received later and is provided by the central controller described in Item 3b, Section 3.7.4.2, and is only received if the event-analysis software indicates that the site OBE site design response spectrum has been exceeded in a potentially damaging frequency range, i.e., at any frequency between 2 to 10 Hz, or the design response spectral velocity has been exceeded between 1 to 2 Hz.

The basis for establishing the OBE design response spectrum for the levels at which control room operator notification is required is that the design of structures, systems, and components for loading combinations, which include OBE, is to code allowable stress levels which are well within the elastic limit of the materials.

3.7.4.4 Controlled Shutdown Logic

The operator will utilize input from multiple sources to determine the need for a controlled shutdown following the seismic event. The decision for a controlled shutdown will be based primarily on an assessment of the actual damage potential of the event. The event analysis data from the SMS will be reviewed to confirm the 'OBE Spectra Exceeded' alarm. The operator may also confirm that ground motion was sensed by plant personnel and/or confirm the occurrence of the seismic event with the National Earthquake Center. Walkdowns of key plant structures, systems, and components will be performed following the seismic event. The walkdowns will be performed using the guidance of Reference [26], and will include checks of the neutron flux monitoring sensors and containment isolation system. If the 'OBE Spectra Exceeded' alarm is confirmed by analysis and the event is confirmed by plant personnel, data from these other sources will be used to determine the best manner in which to proceed with plant shutdown. If a seismic event occurs which does not result in an OBE exceedance (as determined either by annunciation or subsequent analysis), a plant walkdown may be performed to confirm plant condition, however plant shutdown will not be required unless it is determined to be necessary by the operator based on consideration of available information.

The assessment of the damage potential will be made using the OBE Exceedance Criteria developed by the Electric Power Research Institute (EPRI).[25], [26], [27], [28], [29], and [30] As noted above, the indication of damage potential will be provided by event analysis software installed on the centralized SMS described in Section 3.7.4.2. The analysis will be performed for the uncorrected accelerograms recorded from the strong motion triaxial accelerometer located on the base slab in the annulus of the Unit 1 Reactor Building (Item 1a of Section 3.7.4.2). Use of the uncorrected accelerograms is known to be conservative. The basis for use of the seismic motion on the base slab of this structure is that the site OBE design response spectrum is defined at top-of-rock, which corresponds to the base slab location. An engineer will confirm the event analysis results from the SMS.

The EPRI OBE Exceedance Criteria uses two indicators of damage potential. The first indicator of damage potential is specified as the cumulative absolute velocity (CAV), of the accelegram. A meaningful usage of the CAV requires that the recorded data be obtained by an accelerometer mounted in the free-field. As noted above, the OBE design spectrum for WBN is defined as occurring at top-of-rock (i.e., foundation level of the rock-supported structures); whereas, free-field is defined as top-of-soil at sufficient distance from nearby structures to preclude interference/interaction effects. The SMS does not have a free-field accelerometer. Therefore, the shutdown logic adopted will concede CAV exceedance and base the decision on the need for a controlled shutdown solely on the second indicator, as discussed below.

In the absence of data from a free-field accelerometer, the second indicator is an evaluation of the frequency at which the OBE spectrum is exceeded. This criterion is based on research indicating that exceedances above a frequency of 10 Hz are not damaging to nuclear plant structures, systems and components. Two measures of damage potential are used for this second indicator. The OBE design response spectrum is considered exceeded if the 5% damped response spectra generated for any one of the three components of the uncorrected accelerograms from the Containment Building base slab is larger than:

1. The corresponding OBE design response spectral acceleration in a frequency range between 2 - 10 Hz, or,

2. The corresponding OBE design response spectral velocity for frequencies between 1 - 2 Hz.

Basing shutdown logic on the actual damage potential reduces shutdown risk by avoidance of unnecessary shutdowns while ensuring that the operator has the information on plant status necessary to make an informed shutdown decision.

3.7.4.5 Comparison of Measured and Predicted Responses

The steps to be followed after the initiation of a controlled shutdown due to OBE exceedance are discussed in the following sections.

3.7.4.5.1 <u>Retrieval of Data</u>

The digital records for the Reactor Building and Diesel Generator Building accelerometers and the strong motion accelerograph in the Auxiliary Building will be retrieved. The accelerometers and the accelerograph will be recalibrated to confirm the accuracy of the recorded area.

3.7.4.5.2 <u>Evaluation of Recorded Earthquake</u>

Corrected accelerograms and corresponding response spectra, will be prepared for event data recorded by the aforementioned accelerometers and accelerograph. The response spectra for the recorded motion at Elevation 757 in the Reactor Building, Elevation 745 in the Diesel Generator Building, and Elevation 757 in the Auxiliary Building will be compared to the corresponding design spectra for the OBE.

The structural response of these buildings to the recorded earthquake will be compared with the OBE design structural response, and if less, no further analysis will be required. If the structural response of these buildings to the recorded earthquake is greater than the OBE design structural response, then floor response spectra, for the same mass points in these buildings as used in the equipment design, will be produced for use in evaluation of mechanical and electrical equipment response.

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25. EPRI Report NP-5930, “A Criterion for Determining Exceedance of the Operating Basis Earthquake,” July 1988.

26. EPRI Report NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989.

27. EPRI Report TR-100082, "Standardization of the Cumulative Absolute Velocity," December 1991.

28. EPRI Report TR-104239, "Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation," June 1994.

29. NRC Memorandum from Stuart A. Treby, OGC, to Goutam Bagchi, NRR, concerning "Interpretaion of Part 100, Appendix A Regarding: Proposed Guidelines for Determining when Operating Basis Earthquake is Exceeded," dated May 3, 1988.

30. Nuclear Regulatory Commission Regulatory Guide 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions," March 1997.

31. International Building Code, International Code Council, Inc., March 2000.

TABLE 3.7-1

PERIODS FOR SPECTRAL VALUES[(1)]

SET A

Range of Periods, T (sec)	Increment, T (sec)
0.03 to 0.10	0.005
0.11 to 0.30	0.010
0.32 to 0.50	0.020
0.55 to 1.0	0.050

SET B AND SET C[(2)]

Frequency Range (hertz)	Increment (hertz)
0.2 - 3.0	10
3.0 - 3.6	15
3.6 - 5.0	20
5.0 - 8.0	25
8.0 - 15.0	50
15.0 - 18.0	1.0
18.0 - 22.0	2.0
22.0 - 34.0	3.0

NOTES:

(1) Spectral values were computed for the periods/frequencies shown above in addition to the natural frequencies of the structure.

(2) Except for the Auxiliary-Control Building where Set A periods were used in Set C analysis.

TABLE 3.7-2

STRUCTURAL DAMPING RATIOS USED IN

ANALYSIS OF CATEGORY I STRUCTURES, SYSTEMS AND COMPONENTS

	Set A		Set B[8]		Set C	
CATEGORY I STRUCTURES	OBE	SSE	OBE	SSE	OBE	SSE
Reactor Building -						
Interior Concrete Structure	2	5[1]	4	7	2	5[1]
Steel Containment Vessel	1	1	2	4	1	1
Shield Building	2	5[1]	4	7	2	5[1]
Additional Diesel Generator Bldg	N/A	N/A	4	7	5	5
Other Concrete Structures	5	5[1]	4	7	5	5[1]
Refueling Water Storage Tank	2	2	2	4	2	2
Other Welded Steel Structures[4]	2	2[2]	2	4	2	2[2]
Other Bolted Steel Structures[4]	5	5[1]	4	7	5	5[1]

	Set A		SET B[8]		Set B+C	
CATEGORY I SYSTEMS AND COMPONENTS	OBE	SSE	OBE	SSE	OBE	SSE
Piping -						
12" or Larger	0.5	1	2	3	2	3
Less than 12"	0.5	1	1	2	1	2
Optional (Code Case)	N/A	N/A	Note 7	Note 7	Note 7	Note 7
Cable Tray	4	5	4	7	4	7
Conduit	Note 5	2	4	7	4	7
HVAC -						
Companion Angle	Note 6	7	4	7	4	7
Pocket Lock	Note 6	7	7	7	7	7
Welded Duct	Note 3	Note 3	2	4	2	4
Equipment/Components	2	3	2	3	2	3

NOTES:

(1) Damping value of 7% may be used when stress levels are at or near yield.
(2) Damping value of 5% may be used when stress levels are at or near yield.
(3) Not addressed.
(4) Includes TVA-designed supports and anchorage for equipment and component assemblies.
(5) Design is based on SSE only.
(6) OBE loads are assumed to be ½ of SSE loads.
(7) N-411-1--Damping values from ASME Code Case N-411-1.
(8) For Set B, OBE and SSE are site-specific OBE and SSE

TABLE 3.7-3

SUPPORTING MEDIA FOR CATEGORY I STRUCTURES

Rock-Supported Structures (Set A, Set B, and Set C Analyses)

Structure	Shear Wave Velocity of Bedrock, fps
Shield Building	5900
Interior Concrete Structure	5900
Auxiliary-Control Building	5900
Steel Containment Vessel	5900
North Steam Valve Room	5900
ERCW Intake Pumping Station	5900

Soil-Supported Structures

Structure	Shear Wave Velocities (fps)[1] Set A Analysis	Shear Wave Velocities (fps)[1] Set B and Set C Analyses
Diesel Generator Building	1650	Note 2
Waste-Packaging Area	1650	N/A
Refueling Water Storage Tank	1008	Note 2
ERCW Pipe Tunnels	1150	Note 2

Pile-Supported Structures

Structure	Shear Wave Velocities (fps)[1] Set A Analysis	Shear Wave Velocities (fps)[1] Set B and Set C Analyses
Condensate Demineralizer Waste Evaporator Building	761	N/A
Additional Diesel Generator Building	N/A	Note 2

NOTES:

(1) Shear wave velocities are defined at zero shear strain.
(2) Shear wave velocities for Set B and Set C analyses are related to the soil layer, overburden, shear strain, etc. See Section 2.5 (historical information) for a description of the supporting media dynamic soil properties.

WBN

TABLE 3.7-4

SHIELD BUILDING STRUCTURAL PROPERTIES (SET A)

$E = 545,000\ K/FT^2$ $G = 218,000\ K/FT^2$

Element No.	Length Ft	Area Ft^2	Moment of Inertia. Ft^4	Mass Pt. No.	Weight Kips
1	6.67	1194	2435×10^3	1	789.83
2	2.27	1194	2435×10^3	2	556.11
3	4.06	1194	2435×10^3	3	710.40
4	4.06	1174	2435×10^3	4	704.40
5	4.06	1174	2398×10^3	5	710.40
6	4.06	1194	2398×10^6	6	974.30
7	6.92	1194	2435×10^3	7	1590.40
8	10.88	1194	2435×10^3	8	1298.50
9	3.62	1194	2435×10^3	9	648.34
10	3.62	1194	2435×10^3	10	649.24
11	3.63	1194	2435×10^3	11	608.03
12	3.33	1133	2202×10^3	12	565.93
13	3.33	1133	2202×10^3	13	566.78
14	3.43	1133	2202×10^3	14	570.53
15	3.42	1148	2250×10^3	15	573.43
16	3.42	1148	2250×10^3	16	574.29
17	3.43	1148	2250×10^3	17	622.79
18	4.28	1194	2435×10^3	18	750.43
19	4.28	1194	2435×10^3	19	750.43
20	4.25	1194	2435×10^3	20	1500.90
21	12.57	1194	2435×10^3	21	2251.30
22	12.57	1194	2435×10^3	22	2252.20
23	12.58	1194	2435×10^3	23	2253.10
24	12.58	1194	2435×10^3	24	2253.10
25	12.58	1194	2435×10^3	25	6893.50

TABLE 3.7-4A (Sheet 1 of 3)

LUMPED-MASS MODEL PROPERTIES
OF SHIELD BUILDING MODEL (SET B AND SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertias (10^4 x $k\text{-}ft\text{-}sec^2$) J_z	Axial Area (ft^2) A	Shear Areas (ft^2) $A_x = A_y$	Moment of Inertias (10^4 x ft^4) J	$I_{xx} = I_{yy}$
852.1	(Note 1)	42.52				
			1194	597	487	243.5
839.5	69.97	28.52				
			1194	597	487	243.5
826.9	69.97	28.52				
			1194	597	487	243.5
814.3	69.94	28.52				
			1194	597	487	243.5
802.1	69.92	28.52				
			1194	597	487	243.5
789.8	46.61	19.01				
			1194	597	487	243.5
785.6	23.31	9.51				
			1194	597	487	243.5
781.3	23.31	9.51				
			1194	597	487	243.5
777.0	19.34	8.39				
			1148	574	450	225
773.6	17.84	7.26				
			1148	574	450	225

Note 1: Horizontal Mass = 214.1
Vertical Mass = 134.3

TABLE 3.7-4A (Sheet 2 of 3)

LUMPED-MASS MODEL PROPERTIES
OF SHIELD BUILDING MODEL (SET B AND SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertias (10^4 x $k\text{-}ft\text{-}sec^2$) J_z	Axial Area (ft^2) A	Shear Areas (ft^2) $A_x = A_y$	Moment of Inertias (10^4 x ft^4) J	$I_{xx} = I_{yy}$
770.1	17.81	7.26				
			1148	574	450	225
766.7	17.72	7.23				
			1133	567	440.4	220.2
763.3	17.60	7.16				
			1133	567	440.4	220.2
760.0	17.58	7.16				
			1133	567	440.4	220.0
756.6	18.88	7.70				
			1194	597	487	243.5
753.0	20.16	8.21				
			1194	597	487	243.5
749.4	20.13	8.21				
			1194	597	487	243.5
745.8	40.33	16.45				
			1194	597	487	243.5
734.9	49.39	20.15				
			1194	597	487	243.5
728.0	30.26	12.34				
			1194	597	487	243.5

TABLE 3.7-4A (Sheet 3 of 3)

LUMPED-MASS MODEL PROPERTIES
OF SHIELD BUILDING MODEL (SET B AND SET C)

Elevation (ft)	Masses (k-sec^2/ft)	Mass Moment of Inertias (10^4 x k-ft-sec^2) J_z	Axial Area (ft^2) A	Shear Areas (ft^2) $A_x = A_y$	Moment of Inertias (10^4 x ft^4) J	$I_{xx} = I_{yy}$
723.9	22.06	9.01				
			1174	587	479.6	239.8
719.8	21.88	8.94				
			1174	587	479.6	239.8
715.8	22.06	9.01				
			1194	597	487.6	243.5
711.7	17.27	7.04				
			1194	597	487	243.5
709.5	24.53	10.00				
			1194	597	487	243.5
702.8	18.30	7.46				

Dome Vertical SDOF Oscillator

Mass = 79.81 (k-sec^2/ft)
Spring Stiffness = 806 x 10^3 (k/ft)

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

WBN

TABLE 3.7-5

SHIELD BUILDING NAUTRAL PERIODS

	CYLINDRICAL SHELL				DOME	
	Translation Motion		Vertical Motion		Vertical Motion	
Mode No.	Period (Seconds)	Participation Factor	Period (Seconds)	Participation Factor	Period (Seconds)	Participation Factor
1	0.1868	1.326	0.0671	1.232	0.063	2.237
2	0.0951	0.046			0.040	-2.213
3	0.0552	0.580			0.033	1.207
4	0.0313	0.008			0.026	-0.676
5					0.020	1.281

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TABLE 3.7-5A

STEEL CONTAINMENT VESSEL
ELEMENT PROPERTIES

$E = 4,176,000\ K/Ft^2$ $G = 1,670,400\ K/Ft^2$

Element No.	Length, Ft	Area (Ft^2)	Torsion Constant (Ft^4)	North-South Motion: Moment of Inertia (Ft^4)	North-South Motion: Shear Factor	East-West Motion: Moment of Inertia (Ft^4)	East-West Motion: Shear Factor
1	1.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
2	6.22	45.55	152×10^3	75.3×10^3	2	75.3×10^3	2
3	6.50	45.55	152×10^3	75.3×10^3	2	75.3×10^3	2
4	8.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
5	9.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
6	11.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
7	9.50	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
8	3.50	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
9	6.00	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
10	4.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
11	5.00	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
12	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
13	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
14	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
15	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
16	3.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
17	2.50	41.30	137×10^3	68.3×10^3	2	68.3×10^3	2
18	12.00	41.30	129×10^3	64.5×10^3	2	64.5×10^3	2
19	12.46	41.30	108×10^3	54.0×10^3	2	54.0×10^3	2
20	9.54	24.40	456×10^2	22.8×10^3	2	22.8×10^3	2
21	9.00	24.50	268×10^2	13.4×10^3	2	13.4×10^3	2
22	9.00	24.46	81×10^2	40.5×10^2	2	40.5×10^2	2
23	3.00	28.23	36×10	18.2×10	2	18.2×10	2

TABLE 3.7-5B

STEEL CONTAINMENT VESSEL MASS POINT PROPERTIES

Elevations, Ft	Total Horizontal Weight, Kips	Total Vertical Weight, Kips	Weight of Inertia WR^2 $K\text{-}ft^2$	Eccentricity Used in Dynamic Analysis, Ft
703.78	91.87	91.87	305×10^3	0.0
710.00	147.60	147.60	491×10^3	2.43
716.50	227.64	227.64	754×10^3	0.995
724.50	393.44	393.44	$1,301 \times 10^3$	0.0
733.50	335.23	335.23	$1,108 \times 10^3$	-0.033
744.50	424.10	409.28	$1,402 \times 10^3$	0.57
754.00	220.28	190.94	728×10^3	-1.53
757.50	158.12	137.75	523×10^3	-0.82
763.50	310.98	288.44	$1,028 \times 10^3$	0.99
768.00	145.52	125.07	481×10^3	1.23
773.00	222.08	191.02	734×10^3	0.25
782.50	407.87	367.35	$1,349 \times 10^3$	-0.13
792.00	295.51	254.97	977×10^3	-0.105
801.50	318.18	283.31	$1,052 \times 10^3$	-0.052
811.50	216.89	205.01	717×10^3	-0.075
814.50	69.60	64.17	229×10^3	-0.036
817.00	192.21	185.28	630×10^3	0.0
829.00	302.84	302.84	933×10^3	0.0
841.46	183.10	183.10	485×10^3	0.0
851.00	114.66	114.66	227×10^3	0.0
860.00	155.67	155.67	192×10^3	0.0
869.00	84.00	84.00	386×10^2	0.0
872.00	23.25	23.25	186×10	0.0

TABLE 3.7-5C (Sheet 1 of 3)

LUMPED-MASS MODEL PROPERTIES OF STEEL CONTAINMENT VESSEL MODEL

Elevation (ft)	Masses ($K\text{-}sec^2/ft$) $M_x = M_y$	M_z	Mass Moment of Inertias (10^2 k-ft-sec^2) J_z	Axial Area (ft^2) A	Shear Areas ($10^2 \times ft^2$) $A_x = A_y$	Moment of Inertias ($10^2 \times ft^4$) J	$I_{yy} = I_{xx}$
872.0	0.72	0	0.58				
				28.23	14.11	3.64	1.82
869.0	2.61	0	11.99				
				24.46	12.23	80.98	40.49
860.0	4.83	0	59.54				
				24.50	12.25	268.0	134.0
851.0	3.56	0	70.41				
				24.4	12.2	456.0	228.0
841.5	5.69	0	150.70				
				41.30	20.65	1080.0	540.0
829.0	9.40	0	289.67				
				41.30	20.65	1290.0	645.0
817.0	5.96	0	195.89				
				41.30	20.65	1356.0	682.5
814.5	2.16	28.49	71.02				
				45.16	22.58	1439.0	746.6
811.0	6.74	6.37	222.70				
				45.16	22.58	1439.0	746.6
801.50	9.88	8.80	326.71				
				45.16	22.58	1439.0	746.6

TABLE 3.7-5C (Sheet 2 of 3)

LUMPED-MASS MODEL PROPERTIES OF STEEL CONTAINMENT VESSEL MODEL

Elevation (ft)	Masses ($K\text{-}sec^2/ft$)		Mass Moment of Inertias (10^2 k-ft-sec^2)	Axial Area (ft^2)	Shear Areas ($10^2 \times ft^2$)	Moment of Inertias ($10^2 \times ft^4$)	
	$M_x = M_y$	M_z	J_z	A	$A_x = A_y$	J	$I_{yy} = I_{xx}$
792.0	9.18	7.92	303.43				
				45.16	22.58	1439.0	746.6
782.5	12.67	11.41	418.82				
				45.16	22.58	1439.0	746.6
773.0	6.83	5.93	228.03				
				45.16	22.58	1439.0	746.6
768.0	4.52	3.88	149.43				
				45.16	22.58	1439.0	746.6
763.5	9.66	8.96	319.32				
				45.16	22.58	1493.0	746.6
757.5	4.91	4.28	162.36				
				41.55	20.77	1373.7	686.8
754.0	6.84	5.93	226.20				
				41.55	20.77	1373.7	686.8
744.5	13.17	12.71	435.47				
				41.55	22.77	1373.7	686.8
733.5	10.41	10.41	344.19				
				41.55	20.77	1373.7	686.8
724.5	12.22	12.22	403.98				
				41.55	20.77	1373.7	686.8

TABLE 3.7-5C (Sheet 3 of 3)

LUMPED-MASS MODEL PROPERTIES OF STEEL CONTAINMENT VESSEL MODEL (cont'd)

Elevation (ft)	Masses ($K\text{-}sec^2/ft$) $M_x = M_y$	M_z	Mass Moment of Inertias (10^2 k-ft-sec^2) J_z	Axial Area (ft^2) A	Shear Areas ($10^2 \times ft^2$) $A_x = A_y$	Moment of Inertias ($10^2 \times ft^4$) J	$I_{yy} = I_{xx}$
716.5	7.07	7.07	234.22				
				45.55	22.77	1516.0	752.9
710.0	4.56	4.56	152.55				
				45.55	22.77	1516.0	752.9
703.8	2.85	2.85	94.81				
				41.55	20.77	1374.0	686.8

Dome Vertical SDOF Oscillator

Mass = 6.74 ($k\text{-}sec^2/ft$)
Spring Stiffness = 287×10^3 (k/ft)

Steel Properties

Modulus of Elasticity = 4,176,000 k/ft^2
Poisson's Ratio = 0.25

+X = EAST
+Y = NORTH

WBN

TABLE 3.7-6

INTERIOR CONCRETE ELEMENT PROPERTIES

E_C = 720000 K/Ft^2 G_C = 288000 K/Ft^2

				East-West Motion		North-South Motion	
Element No.	Length, Ft	Area, Ft^2	Torsion Constant, Ft^4	Moment of Inertia, Ft^4	Shear Factor	Moment of Inertia, Ft^4	Shear Factor
1	12.22	1779	1840 x 10^3	1024 x 10^3	1.76	1021 x 10^3	1.79
2	10.00	2107	1700 x 10^3	1849 x 10^3	1.70	1281 x 10^3	2.27
3	9.96	1796	1610 x 10^3	1829 x 10^3	1.49	1271 x 10^3	2.20
4	9.96	1796	1610 x 10^3	1829 x 10^3	1.49	1271 x 10^3	2.20
5	5.36	880	249 x 10^3	990 x 10^3	1.07	320 x 10^3	1.75
6	5.35	880	249 x 10^3	990 x 10^3	1.07	320 x 10^3	1.75
7	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
8	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
9	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
10	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
11	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
12	6.72	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
13	11.82	816	1510 x 10^3	755 x 10^3	2.00	755 x 10^3	2.00
14	11.82	816	1510 x 10^3	755 x 10^3	2.00	755 x 10^3	2.00

TABLE 3.7-6A (Sheet 1 of 2)

LUMPED-MASS MODEL PROPERTIES
OF INTERIOR CONCRETE STRUCTURE-HORIZONTAL MODEL - SET B AND SET C

Elevation (ft)	Masses (K-sec^2/ft)	Mass Moment of Inertias (10^4 k-ft-sec^2)	Shear Center (ft)		Shear Areas (ft^2)		Moment of Inertias (10^3 x ft^4)		
		J_z	e_x	e_y	A_x	A_y	I_{xx}	I_{yy}	J
819.0	38.9	4.2	0.0	0.0					
					335	408	755	625	1510
807.8	45.5	8.3	0.0	0.0					
					335	408	755	625	1510
796.0	123.7	13.0	-33.67	3.06					
					445	200	1070	655	840
789.3	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
782.6	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
775.8	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
769.1	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
762.4	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
755.6	89.8	18.4	28.21	0.90					
					575	540	555	800	250
750.3	40.0	4.4	28.21	0.90					
					575	540	555	800	250

TABLE 3.7-6A (Sheet 2 of 2)

LUMPED-MASS MODEL PROPERTIES
OF INTERIOR CONCRETE STRUCTURE-HORIZONTAL MODEL - SET B AND SET C

Elevation (ft)	Masses ($K\text{-}sec^2/ft$)	Mass Moment of Inertias (10^4 k-ft-sec^2) J_z	Shear Center (ft) e_x	e_y	Shear Areas (ft^2) A_x	A_y	Moment of Inertias ($10^3 \times ft^4$) I_{xx}	I_{yy}	J
744.9	135.2	14.7	-0.25	0.85					
					915	580	1140	1460	1650
735	110	12.6	-0.25	0.85					
					915	580	1140	1460	1650
725	114.5	15.2	3.63	1.76					
					1075	835	1170	1455	1830
715	160.2	23.2	0.54	0.13					
					940	1185	965	825	1815
702.8	1160	230	0.0	0.0					

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

TABLE 3.7-6B (Sheet 1 of 2)

LUMPED-MASS MODEL PROPERTIES OF
INTERIOR CONCRETE STRUCTURE-VERTICAL MODEL - SET B AND SET C

Elevation (ft)	Masses (k-sec^2/ft)	Mass Moment of Inertia (10^4k-ft-sec^2) J	Location of Centroid (ft) d_x	d_y	Areas (ft^2) A
819.6	38.9	4.2			
			0.0	0.0	816
807.8	45.5	8.3			
			0.0	0.0	816
796.0	123.7	13.0			
			-4.2	0.61	1154
789.3	58.4	8.0			
			-4.2	0.61	1154
782.6	58.4	8.0			
			-4.2	0.61	1154
775.8	58.4	8.0			
			-4.2	0.61	1154
769.1	58.4	8.0			
			-4.2	0.61	1154
762.4	58.4	8.0			
			-4.2	0.61	1154
755.6	89.8	18.4			
			16.60	0.29	880
750.3	40.0	4.4			
			16.60	0.29	880

TABLE 3.7-6B (Sheet 2 of 2)

LUMPED-MASS MODEL PROPERTIES OF
INTERIOR CONCRETE STRUCTURE-VERTICAL MODEL - SET B AND SET C

Elevation (ft)	Masses (k-sec^2/ft)	Mass Moment of Inertia (10^4k-ft-sec^2) J	Location of Centroid Areas (ft)(ft^2) d_x	d_y	A
744.9	135.2	14.7			
			8.56	0.60	1796
735	110	12.6			
			8.56	0.60	1796
725	114.5	15.2			
			6.52	0.98	2107
715	160.2	23.2			
			0.25	0.08	1779
702.8	1160	230			

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

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TABLE 3.7-7

INTERIOR CONCRETE STRUCTURE - MASS POINT PROPERTIES (SET A)

Point No.	Total Wt. Kips	Equip. Wt. Kips	WR^2 $K\text{-}Ft^2$	Eccentrictiy, Ft E-W Motion	Eccentrictiy, Ft N-S Motion
1	8203	3588	7.48×10^6	0.0	3.8
2	4539	1619	4.89×10^6	0.0	7.2
3	3574	894	4.07×10^6	0.0	11.0
4	4352	1211	4.74×10^6	0.0	- 3.4
5	1288	578	1.41×10^6	0.0	-12.6
6	5451	3397	5.92×10^6	0.0	21.6
7	1879	714	2.56×10^6	0.0	43.7
8	1879	714	2.56×10^6	0.0	43.7
9	1879	714	2.56×10^6	0.0	43.7
10	1879	714	2.56×10^6	0.0	43.7
11	1879	714	2.56×10^6	0.0	43.7
12	3983	1734	4.16×10^6	0.0	17.7
13	1464	14	2.67×10^6	0.0	0.0
14	1253	588	1.34×10^6	0.0	0.0

WBN

TABLE 3.7-8

INTERIOR CONCRETE STRUCTURE - NORMAL MODES OF VIBRATION (SET A)

Mode No.	East-West Motion Frequency, cps (Period, sec)	East-West Motion Participation Factor	North-South Motion Frequency, cps (Period, sec)	North-South Motion Participation Factor	Vertical Motion Frequency, cps (Period, sec)	Vertical Motion Participation Factor
1	8.81 (0.114)	1.665	4.96 (0.202)	1.600	22.68 (0.044)	1.408
2	23.82 (0.042)	-0.950	9.64 (0.104)	2.329		
3			14.88 (0.067)	0.032		
4			22.91 (0.044)	-0.852		
5			24.88 (0.040)	0.887		
6			32.34 (0.031)	1.760		

TABLE 3.7-9

AUXILIARY BUILDING ELEMENT PROPERTIES (SET A, SET B, AND SET C)

E_C =590,000 k/ft^2; G_C = 236,000 k/ft^2 (For Set A)

E_C = 590,000 k/ft^2; G_C = 252,800 k/ft^2 (For Set B and Set C)

				North-South Motion		East-West Motion	
Elevation	Length (ft)	Area (ft^2)	Torsion Constant (ft^4)	Moment of Inertia (ft^4)	Shear Area (ft^2)	Moment of Inertia (ft^4)	Shear Area (ft^2)
692.00	7.62	11,172	2,893x10^4	11,914x10^4	4,968	5,728x10^4	4,860
699.62	7.63	11,172	2,893x10^4	11,914x10^4	4,968	5,728x10^4	4,860
707.25	4.25	11,172	2,893x10^4	11,914x10^4	4,968	5,728x10^4	4,860
711.50	8.38	8,410	2,125x10^4	8,570x10^4	2,244	5,708x10^4	3,592
719.88	8.37	8,410	2,125x10^4	8,570x10^4	2,244	5,708x10^4	3,592
728.25	8.25	7,902	2,302x10^4	8,178x10^4	2,108	4,801x10^4	3,340
736.50	9.25	7,340	2,640x10^4	8,052x10^4	2,174	4,645x10^4	2,867
745.75	9.75	7,340	2,640x10^4	8,052x10^4	2,174	4,645x10^4	2,867
755.50	16.00	5,609	2,746x10^4	5,961x10^4	1,503	4,469x10^4	2,310
771.50	10.00	4,269	1,820x10^4	3,782x10^4	1,242	2,460x10^4	1,609
781.50	4.00	4,269	1,820x10^4	3,782x10^4	1,242	2,460x10^4	1,609
800.50	15.00	1,495	286x10^4	1,037x10^4	432	672x10^4	570
814.25	13.75	781	233x10^4	601x10^4	319	86x10^4	201

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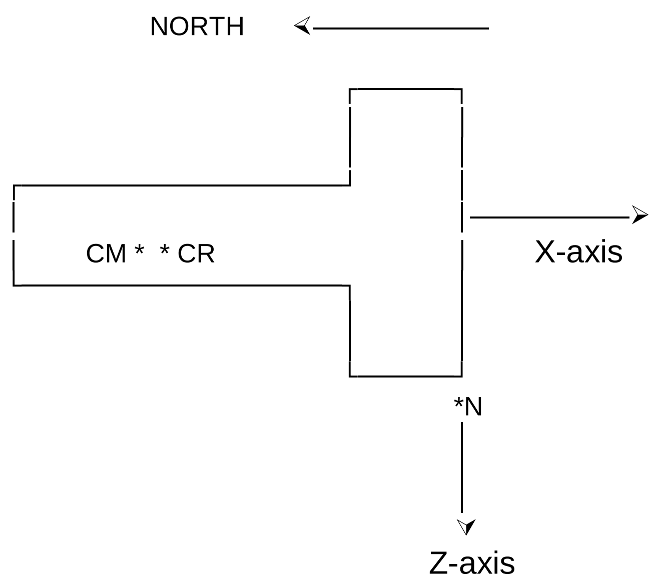
TABLE 3.7-9A

AUXILIARY BUILDING NODAL COORDINATES (SET B AND SET C)

Legend

CM - Center of mass

CR - Center of rigidity



Elev. Y-Coord	CM Mode X-Coord	CR Mode X-Coordinate H-Model	CR Mode X-Coordinate V-Model	CM&CR Mode Z-Coord
814.25	-213.75			0.00
		-230.28	-218.36	0.00
800.50	-207.42			0.00
		-178.72	-205.68	0.00
785.50	-135.51			0.00
		-84.47	-139.18	0.00
781.50	-154.65			0.00
		-84.47	-139.18	0.00
771.50	-96.66			0.00
		-67.32	-123.15	0.00
755.50	-140.46			0.00
		-79.42	-158.35	0.00
745.75	-156.75			0.00
		-79.42	-158.35	0.00
736.50	-145.28			0.00
		-84.25	-160.35	0.00
728.50	-155.67			0.00
		-85.92	-162.52	0.00
719.88	-149.55			0.00
		-85.92	-162.52	0.00
711.50	-137.02			0.00
		-65.17	-161.35	0.00
707.25	-125.61			0.00
		-65.17	-161.35	0.00
699.62	-158.00			0.00
		-65.17	-161.35	0.00
692.00	Fixed base	-65.17	-161.35	0.00

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TABLE 3.7-10

AUXILIARY BUILDING MASS POINT PROPERTIES (SET A, SET B, SET C)

	EAST-WEST MOTION		NORTH-SOUTH MOTION		VERTICAL MOTION		
Elevation	Total Weight (kips)	Equip. & Added Soil Weight (kips)	Total Weight kips	Equip. & Added Soil Weight (kips)	Total Weight kips	Equip. & Added Soil Weight (kips)	WR^2 $(K\text{-}Ft^2)$
699.62	18209	2471	20320	4582	22791	7053	2.22×10^8
707.25	18036	1925	19681	3570	21605	5494	2.64×10^8
711.50	29461	4466	30450	5455	32496	7501	2.90×10^8
719.88	18620	3534	21718	6632	24431	9347	2.11×10^8
728.25	23473	2886	24559	3972	25510	4923	3.02×10^8
736.50	21840	895	21840	895	21840	895	2.52×10^8
745.75	13131	905	13131	905	13131	905	1.91×10^8
755.50	25921	273	25921	273	25921	273	3.85×10^8
771.50	20797	146	20797	146	20797	146	3.36×10^8
781.50	7311	0	7311	0	7311	0	0.97×10^8
785.50	7870	399	7870	399	7870	399	0.90×10^8
800.50	4676	41	4676	41	4676	41	0.45×10^8
814.25	5023	352	5023	352	5023	352	0.30×10^8

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TABLE 3.7-11

AUXILIARY BUILDING NATURAL PERIODS (SET A)

Model No.	North-South Motion Frequency (cycles/sec.)	North-South Motion Mass Participation Factor	East-West Motion Frequency (cycles/sec.)	East-West Motion Mass Participation Factor	Vertical Motion Frequency (cycles/sec.)	Vertical Motion Mass Participation Factor
1	8.17	-2.157×10^3	6.05	1.324×10^3	23.25	-2.300×10^3
2	17.60	0.897×10^3	10.11	1.762×10^3		
3	24.84	0.747×10^3	16.00	0.645×10^3		
4			18.77	-0.696×10^3		

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TABLE 3.7-12

NORTH STEAM VALVE ROOM ELEMENT PROPERTIES

$E_c = 720000\ k/Ft^2$ $G_c = 300000\ K/Ft^2$

				North-South Direction		East-West Direction	
Element No.	Length (Ft)	Area (Ft^2)	Torsion Constant (Ft^4)	Moment of Inertia (Ft^4)	Shear Factor	Moment of Inertia (Ft^4)	Shear Factor
1	9.375	830	4423	102117	1.23	289089	5.32
2	9.375	830	4423	102117	1.23	289089	5.32
3	9.375	830	4423	102117	1.23	289089	5.32
4	9.375	830	4423	102117	1.23	289089	5.32
5	7.000	1960	422400	235500	1.00	542700	1.00
6	8.580	317	1352	20587	1.61	139759	2.64
7	8.420	318	1263	19471	1.49	151177	3.03
8	7.000	373	1717	22339	1.39	159900	3.55
9	5.000	593	6160	33462	1.97	216100	2.03
10	5.000	593	6160	33462	1.97	216100	2.03
11	8.000	400	1861	28528	1.33	184401	4.05
12	5.250	230	422	17895	1.83	110963	2.21

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TABLE 3.7-13

NORTH STEAM VALVE ROOM MASS POINT PROPERTIES

Mass Point No.	Total Weight (Kips)* N-S Direction	Total Weight (Kips)* E-W Direction
1	1976	2796
2	1976	2796
3	1976	2796
4	2723	3443
5	1149	1369
6	658	658
7	475	475
8	477	477
9	552	552
10	473	473
11	331	331
12	330	330

*Includes the weight of contained fill material for mass points 1-4.

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TABLE 3.7-13A

LUMPED-MASS MODEL PROPERTIES
OF UNIT 1 NORTH STEAM VALVE ROOM (NSVR) - HORIZONTAL MODEL (SET B, Set C)

		Mass Moment of Inertia (10^3 k-ft-Sec^2)		Shear Center (ft)		Shear Areas (ft^2)		Moment of Inertias (10^3 x ft^4)		
Elevation (ft)	Masses (K-sec^2/ft)	J_K	J_Y	e_X	e_y	A_X	A_Y	J	I_{x-x}	I_{y-y}
777	15.39	0.60	3.75							
				8.72	4.14	106	257	43.8	21.5	102.3
763	28.4	1.69	10.87							
				8.68	3.69	319	295	60.8	38.2	177.6
753	30.75	1.52	10.43							
				6.18	3.83	110	259	54.9	18.7	132.4
738	27.82	1.16	8.54							
				1.49	-1.32	130	228	89.1	14.6	71.9
728	9.9	0.48	3.25							

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

TABLE 3.7-13B

LUMPED-MASS MODEL PROPERTIES OF
UNIT 1 NORTH STEAM VALVE ROOM (NSVR) - VERTICAL MODEL (SET B, SET C)

Elevation (ft)	Masses (k-sec^2/ft)	Mass Moment of Inertia (10^3 k-ft-sec^2) J_z	Location of Centroid (ft) d_x	d_y	Areas (ft^2) A
777	15.39	7.37			
			7.19	-8.68	363
763	28.4	13.34			
			1.20	-5.91	613
753	30.75	12.78			
			2.54	-8.75	369
738	27.82	10.72			
			-.28	-9.62	358
728	9.9	3.74			

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2

Poisson's Ratio = 0.15

+X = EAST

+Y = NORTH

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TABLE 3.7-14

NORTH STEAM VALVE ROOM NAUTRAL FREQUENCIES

Case	Mode No.	North-South Motion Frequency (Hz)	North-South Motion Participation Factor	East-West Motion Frequency (Hz)	East-West Motion Participation Factor	Vertical Motion Frequency (Hz)	Vertical Motion Participation Factor
	1	2.63	-0.0465	2.62	-0.2666		
	2	6.76	0.2428	6.04	-0.9882		
	3	9.03	2.6640	9.59	2.4961		
	4	11.28	-0.6918	11.30	1.7919		
0.5 G	5	16.25	0.4464	16.12	0.6909		
	6	22.22	0.2433	20.04	-0.9738		
	7	24.33	1.5204	22.32	0.3151		
	8	25.88	-1.3297	25.43	-0.0209		
	1	2.64	-0.0476	2.62	-0.2543	34.37	1.7273
	2	6.79	0.2278	6.10	-0.9356	---	----
G	3	9.19	2.6917	9.96	2.7548	---	----
	4	11.29	-0.7362	11.36	2.2774	---	----
	5	16.26	0.4458	16.14	0.7316	---	----
	6	22.25	0.2179	20.25	-1.0336	---	----
	7	24.56	1.6430	22.33	0.3650	---	----
	8	25.98	-1.5517	25.43	-0.0218	---	----
	1	2.65	-0.0487	2.63	-0.2444		
	2	6.81	0.2154	6.14	-0.8933		
1.5 G	3	9.33	2.7216	10.28	-3.3180		
	4	11.29	-0.7817	11.43	2.8117		
	5	16.27	0.4452	16.17	0.7708		
	6	22.25	0.1981	20.47	-1.0920		
	7	24.76	1.6990	22.33	0.4278		
	8	26.12	-1.7493	25.43	-0.0229		

TABLE 3.7-15

PUMPING STATION ELEMENT PROPERTIES

E_C = 590,000 k/ft^2; G_C = 246,000 k/ft^2

Element No.	Length Ft	Area Ft^2	Motion about X-X: Moment of Inertia Ft^4	Motion about X-X: Shear Factor	Motion about Y-Y: Moment of Inertia Ft^4	Motion about Y-Y: Shear Factor
1	10.25	2338	1410 x 10^3	1.565	3061 x 10^3	2.769
2	8.00	2825	1833 x 10^3	1.865	3248 x 10^3	2.155
3	10.00	2825	1833 x 10^3	1.865	3248 x 10^3	2.155
4	10.00	2825	1833 x 10^3	1.865	3248 x 10^3	2.155
5	10.00	2825	1833 x 10^3	1.865	3248 x 10^3	2.155
6	9.50	2825	1833 x 10^3	1.865	3248 x 10^3	2.155
7	11.00	2747	1779 x 10^3	2.005	3389 x 10^3	1.995
8	5.75	2602	1774 x 10^3	2.124	3305 x 10^3	1.889
9	6.50	1932	497 x 10^3	1.807	2710 x 10^3	2.238
10	6.50	1932	497 x 10^3	1.807	2710 x 10^3	2.238

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TABLE 3.7-15A

INTAKE PUMPING STATION BEAM ELEMENT PROPERTIES (SET B, SET C)

Elev.	A_x	A_y	A_z	Moment of Inertia J_{xx}	I_{yy}	I_{zz}
	←	ft^2	→	←	ft^4	→
754.00						
	1217	541.6	675.4	$0.512x10^6$	1337353	294476
739.50						
	2167	847.3	1319.7	$1.13x10^6$	3158004	572896
733.00						
	2167	847.3	1319.7	$1.13x10^6$	3158004	572896
726.50						
	2772	1297.3	1474.7	$2.15x10^6$	3730181	1745654
720.75						
	3105	1639.4	1465.6	$2.15x10^6$	4070156	1763585
709.75						
	3148	1520.5	1627.5	$2.05x10^6$	3903497	1904018
700.25						
	3000	1302.0	1698.0	$2.05x10^6$	3674971	1896620
690.25						
	3000	1302.0	1698.0	$2.05x10^6$	3674971	1896620
680.25						
	2958	1260.1	1697.9	$2.05x10^6$	3601451	1889580
670.25						
	2945	1248.7	1696.3	$2.05x10^6$	3654683	1860185
662.25						
	2204	894.8	1309.2	$1.89x10^6$	2981336	1344534
652.00						

Notes: 1. x, y and z are local coordinate axes, i.e.,
x:vertical (global Z)
y:transverse (global Y)
z:longitudinal (global X)

2. A_x is the cross-sectional area, and A_y and A_z are the shear areas in the transverse and longitudinal directions respectively. J_{xx} is torsional moment of inertia, and I_{yy}, and I_{zz} are the bending moments of inertia about the transverse and longitudinal axes respectively.

3. E_c for beam elements: 590000 k/ft^2

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TABLE 3.7-16

PUMPING STATION MASS POINT PROPERTIES

	1/2 SSE PROPERTIES		SSE PROPERTIES	
Mass Point No.	Total Wt Kips	Equip. & Water Wt Kips	Total Wt Kips	Equip. & Water Wt Kips
1	7871	4378	7871	4378
2	7804	3990	7804	3990
3	7448	3210	7448	3210
4	7448	3210	7454	3216
5	7262	3130	7467	3335
6	8637	4357	8306	4026
7	6856	3467	6384	2995
8	3217	1153	3217	1153
9	1884	0	1884	0
10	4593	3652	4593	3652

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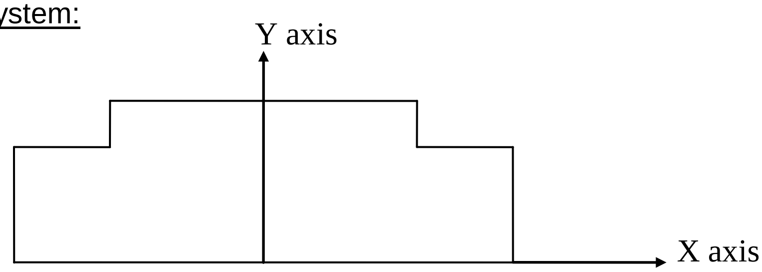
TABLE 3.7-16A

INTAKE PUMPING STATION NODAL WEIGHT PROPERTIES (SET B and SET C)

WEIGHTS

	WEIGHTS			MOMENTS OF INERTIA		
Elev.	W_x	W_y	W_z	W_{xx}	W_{yy}	W_{zz}
feet	<-----------	kips	------------->	<------------	10^6 kips-ft^2	------------>
754.00	1696	1696	1696	0.38878	1.8942	2.2830
739.50	5046	5046	5046	1.0626	6.5340	7.5965
733.00	2160	2160	2160	0.60628	3.1329	3.7392
726.50	3595	3595	3368	1.9182	3.7848	5.7031
720.75	6206	6206	5123	2.3578	7.1056	9.4634
709.75	7709	7709	5571	3.2726	6.5889	9.8615
700.25	7856	7856	4493	2.7792	5.5375	8.3167
690.25	7994	7994	4499	2.8449	5.5125	8.3574
680.25	7977	7977	4469	2.8397	5.4573	8.2970
670.25	7543	7543	5108	2.8676	6.2701	9.1378
662.25	7562	7562	4564	2.5089	5.8541	8.3630

Coordinate System:



X axis - Longitudinal direction
Y axis - Transverse direction
Z axis - Vertical direction

WBN

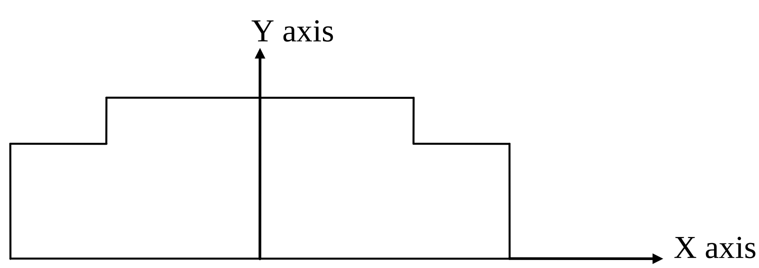
TABLE 3.7-16B

INTAKE PUMPING STATION NODAL COORDINATES (SET B AND SET C)
(Feet Units)

Elev.	CM Node		CR Node	
Z-Coord	X-Coord	Y-Coord	X-Coord	Y-Coord
754.00	0.38	23.15		
			0.00	24.02
739.50	0.06	23.31		
			0.00	25.41
733.00	0.74	21.55		
			0.00	25.41
726.50	0.03	36.61		
			0.00	42.40
720.75	-0.01	31.20		
			0.00	42.97
709.75	0.00	36.24		
			0.00	44.71
700.25	0.00	33.21		
			0.00	45.54
690.25	0.00	33.04		
			0.00	45.54
680.25	0.00	33.13		
			0.00	46.00
670.25	0.00	34.30		
			0.00	45.87
662.25	0.00	33.13		
			0.00	38.01
652.00			0.00	38.01
648.00	Fixed Base==>		0.00	43.00

Coordinate System:

CM - Center of Mass
CR - Center of Rigidity



X axis - Longitudinal direction
Y axis - Transverse direction
Z axis - Vertical direction

WBN

TABLE 3.7-17

PUMPING STATION NATURAL PERIODS

1/2 SAFE SHUTDOWN EARTHQUAKE

	Motion About X-X		Motion About Y-Y		Vertical Motion	
Mode No.	Period (Sec)	Participation Factor	Period (Sec)	Participation Factor	Period (Sec)	Participation Factor
1	0.1085	1.4529	0.1091	1.3786	0.0420	1.2995
2	0.0353	-0.6714	0.0374	-0.5608	0.01454	-0.4914
3	0.0254	0.3298	0.0222	0.2693	0.00939	0.2946

SAFE SHUTDOWN EARTHQUAKE

	Motion About X-X		Motion About Y-Y		Vertical Motion	
Mode No.	Period (Sec)	Participation Factor	Period (Sec)	Participation Factor	Period (Sec)	Participation Factor
1	0.1075	1.4579	0.1082	1.3830	0.0417	1.3035
2	0.0353	-0.6750	0.0374	-0.5652	0.0145	-0.4958
3	0.0205	0.3259	0.0222	0.2671	0.00938	0.2924

TABLE 3.7-18

DIESEL-GENERATOR BUILDING ELEMENT PROPERTIES

E_C = 590,000 K/FT^2 G_C = 236,000 K/FT^2

			North-South Motion		East-West Motion	
Element No.	Length, Ft.	Area, Ft^2	Moment of Inertia, Ft^4	Shear Factor	Moment of Inertia, Ft^4	Shear Factor
1	6.00	1060	119 x 10^4	1.77	235 x 10^4	2.30
2	6.00	1060	119 x 10^4	1.77	235 x 10^4	2.30
3	5.75	1162	137 x 10^4	1.91	252 x 10^4	3.10
4	3.75	1259	117 x 10^4	2.10	225 x 10^4	3.00
5	5.00	992	64 x 10^4	1.65	190 x 10^4	6.52
6	2.75	1259	117 x 10^4	2.10	225 x 10^4	3.00

WBN

TABLE 3.7-19

DIESEL-GENERATOR BUILDING MASS POINT PROPERTIES

Mass Point No.	Total Weight (Kips)	Equipment Weight (Kips)	Weight Moment of Inertia ($K\text{-}Ft^2$)	
			N-S Motion	E-W Motion
Base	14,800	650	178×10^5	273×10^5
1	960	-	107×10^4	212×10^4
2	980	-	113×10^4	215×10^4
3	3,250	205	223×10^4	472×10^4
4	920	-	57×10^4	135×10^4
5	800	-	48×10^4	118×10^4
6	2,250	-	100×10^4	223×10^4

WBN

TABLE 3.7-19A

LUMPED-MASS MODEL PROPERTIES

OF DIESEL GENERATOR BUILDING - HORIZONTAL MODEL (SET B and SET C)

Elevation (ft)	Masses (K-sec^2/ft)	Mass Moment of Inertias (10^4 k-ft-sec^2) J_z	Shear Center (ft) e_x	Shear Center (ft) e_y	Shear Areas (ft^2) A_x	Shear Areas (ft^2) A_y	Moment of Inertias (10^3 x ft^4) J	Moment of Inertias (10^3 x ft^4) $I_{y\text{-}y}$	Moment of Inertias (10^3 x ft^4) $I_{y\text{-}y}$
773.5	107.58	27.85							
			0	10.49	430.8	645.3	1295	868	1445
768.5	-	-							
			0	2.27	182.7	635.9	996	711.5	1003
763.5	-	-							
			0	10.08	434.7	642.1	1295	926.5	1333
759.75	188.89	49.40							
			0	4.51	562	589.6	1290	765.5	1653
754	-	-							
			0	-17.1	697.05	715.97	1707	1522	2013
748	-	-							
			0	-19.41	729.1	765.8	1765	1731	2098
742	43.73	15.67							
			0	0	Rigid Link				
Basemat	540.87	130.59*							

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2

Poisson's Ratio = 0.15

+X = EAST

+Y = NORTH

*Rocking Mass Moment of Inertia: $I_{X\text{-}X}$ = 85.6 x 10^4 k-ft-S^2; $I_{Y\text{-}Y}$ = 137.92 x 10^4 k-ft-S^2

WBN

TABLE 3.7-19B

LUMPED-MASS MODEL PROPERTIES OF
DIESEL GENERATOR BUILDING - VERTICAL MODEL (SET B and SET C)

Elevation (ft)	Masses ($K\text{-}sec^2/ft$)	Mass Moment of Inertia (10^4k-ft-sec^2) J_z	Location of Centroid (ft) d_x	d_y	Areas (ft^2) A
773.5	107.58	27.85			
			0	10.14	1086.03
768.5	-	-			
			0	7.90	818.6
763.5	-	-			
			0	9.48	1076.8
759.75	188.89	49.40			
			0	7.91	1151.6
754	-	-			
			0	-3.54	1413.02
748	-	-			
			0	-6.27	1494.9
742	43.73	15.67			
Base	540.87	130.59*			

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

*Rocking Mass Moment of Inertia:

$I_{X\text{-}X} = 85.6 \times 10^4$ k-ft-S^2; $I_{Y\text{-}Y} = 137.92 \times 10^4$ k-ft-S^2

TABLE 3.7-19C

LUMPED-MASS MODEL PROPERTIES
OF REFUELING WATER STORAGE TANK - SEISMIC MODEL (SET B and SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)		Mass Moment of Inertias (10^4 $k\text{-}ft\text{-}sec^2$)		Axial Area A (ft)	Shear Areas (ft^2)		Moment of Inertia (ft^4)		
	M_X=M_Y	M_Z	I_X=I_Y	J		A_x	A_y	J	$I_{x\text{-}x}$	$I_{y\text{-}y}$
767.20	0.96	0.96	-	-						
					3.551	1.884	1.884	1678	839	839
763.20	3.94	0.31	-	-						
					3.551	1.884	1.884	1678	839	839
759.66	7.86	.34	-	-						
					4.233	2.246	2.246	2000	1000	1000
755.85	8.17	0.37	-	-						
					4.233	2.246	2.246	2000	1000	1000
752.04	8.17 28.32*	0.37	-	-						
					4.643	2.463	2.463	2192	1096	1096
748.23	8.17	0.37	-							
					4.643	2.463	2.463	2192	1096	1096
744.42	8.23	0.43	-	-						
					6.007	3.187	3.187	2836	1418	1418
740.61	8.26	0.47	-	-						
					6.007	3.187	3.187	2836	1418	1418
736.80	8.29	0.50	-	-						
					7.507	3.982	3.982	3542	1771	1771
733.0	8.32	0.56	-	-						
					7.507	3.982	3.982	3542	1771	1771
729.20	4.16	28.98 69.22**	1.24	2.48						
Base										

Young's Modulus E = 30,000 ksi

Shear Modulus G = 11,540 ksi

*Sloshing-induced horizontal mass, $M_X = M_Y = 28.32$ $k\text{-}sec^2/ft$
associated horizontal spring, $K_X = K_Y = 76.2$ k/ft

**Seismic-induced vertical effective mass, $M_Z = 69.22$ $k\text{-}sec^2/ft$
associated vertical spring $K_Z = 246120$ k/ft

WBN

TABLE 3.7-20

DIESEL-GENERATOR BUILDING NATURAL PERIODS

V_S = 1150 FPS

Mode No.	N-S Motion: $K_T = 147 \times 10^4$ K/Ft, $K_R = 300 \times 10^7$ K-Ft RAD — Period, Second	E-W Motion: $K_T = 141 \times 10^4$ K/Ft, $K_R = 425 \times 10^7$ K-Ft RAD — Period, Second
1	0.154	0.156
2	0.103	0.111
3	0.029	0.035

V_S = 1650 FPS

Mode No.	N-S Motion: $K_T = 308 \times 10^4$ K/Ft, $K_R = 614 \times 10^7$ K-Ft RAD — Period, Second	E-W Motion: $K_T = 294 \times 10^4$ K/Ft, $K_R = 887 \times 10^7$ K-Ft RAD — Period, Second
1	0.108	0.110
2	0.072	0.077
3	0.028	0.034

V_S = 2150 FPS

Mode No.	N-S Motion: $K_T = 517 \times 10^4$ K/Ft, $K_R = 1031 \times 10^7$ K-Ft RAD — Period, Second	E-W Motion: $K_T = 493 \times 10^4$ K/Ft, $K_R = 1490 \times 10^7$ K-Ft RAD — Period, Second
1	0.085	0.087
2	0.056	0.059
3	0.028	0.033

TABLE 3.7-21

WASTE-PACKAGING AREA ELEMENT PROPERTIES

E_C = 590,000 K/FT^2 G_C = 236,000 K/FT^2

			North-South Motion		East-West Motion	
Element No.	Length (Ft)	Area (Ft^2)	Moment of Inertia (Ft^4)	Shear Factor	Moment of Inertia, Ft^4	Shear Factor
1	9.00	573.2	184100	2.36	556500	1.63
2	6.50	573.2	184100	2.36	556500	1.63
3	6.50	573.2	184100	2.36	556500	1.63
4	5.75	573.2	184100	2.36	556500	1.63
5	5.75	319.2	78630	1.59	393500	2.56
6	5.75	237.7	72700	1.95	267000	1.91
7	5.75	156.4	66728	3.89	136000	1.25

TABLE 3.7-22

WASTE-PACKAGING AREA MASS POINT PROPERTIES

Mass Point Mo.	Total Weight Kips	Weight Moment of Inertia, K-ft^2	
		N-S Motion	E-W Motion
Base	3108	6.04×10^5	2.08×10^6
1	971	1.86×10^5	5.63×10^5
2	629	1.79×10^5	5.43×10^5
3	512	1.69×10^5	5.11×10^5
4	418	1.13×10^5	4.10×10^5
5	597	0.65×10^5	2.85×10^5
6	535	0.60×10^5	1.74×10^5
7	444	0.29×10^5	0.59×10^5

WBN

TABLE 3.7-23

WASTE-PACKAGING AREA NATURAL PERIODS

1/2 SSE

Mode No.	N-S Motion $K_T = 8.54 \times 10^5$ K/Ft $K_R = 4.24 \times 10^8$ K-Ft/Rad Period, Second	E-W Motion $K_T = 8.30 \times 10^5$ K/Ft $K_R = 1.65 \times 10^9$ K-Ft/Rad Period, Second
1	0.143	0.116
2	0.072	0.065

SSE

Mode No.	N-S Motion $K_T = 5.34 \times 10^5$ K/Ft $K_R = 6.02 \times 10^7$ K-Ft/Rad Period, Second	E-W Motion $K_T = 8.30 \times 10^5$ K/Ft $K_R = 1.65 \times 10^9$ K-Ft/Rad Period, Second
1	0.313	0.116
2	0.108	0.065

Table 3.7-23A

CDWE BUILDING

SOIL DEPOSIT SHEAR MODULI AND SHEAR WAVE VELOCITIES

Soil Profile	Shear Modulus (ksf)	Shear Wave Velocity (f/s)
Average	2409	761
-50% Variation	1205	538
+50% Variation	3613	932

SPRING CONSTANTS FOR PILE GROUP

Direction	Spring Constant
N-S Translation	5.61×10^5 k/f
E-W Translation	5.31×10^5 k/f
Rocking About E-W Axis	1.61×10^9 k-f/rad
Rocking About N-S Axis	2.57×10^9 k-f/rad
Vertical	3.41×10^6 k/f

MASS POINT PROPERTIES - LUMPED MASS MODEL

Mass Point	Weight (kips)	Center of Gravity (f)		Mass Moment About CG (k-f)		Mass Moment About Geometric Center (k-f)		
		X	Y	X	Y	X	Y	Z
Base	2669.5	-1.054	-0.829	386095	659047	370909	621259	955811
1	742.3	0.0	0.0	220746	331660	220746	331660	530274
2	2031.9	-1.850	0.050	357010	612045	357455	611943	932527
3	742.3	0.0	0.0	220746	331660	220746	331660	530274
4	1120.3	0.600	0.0	227799	346548	227126	346548	570708

ELEMENT PROPERTIES - LUMPED MASS MODEL

Element	Length (f)	Area (f^2)	Moment of Inertia (f^4)		Torsion Constant (f^4)	Shape Factor	
			X	Y		X	Y
1	14.75	370.0	104513	159798	5718000	1.69	2.22
2	14.0	370.0	104513	159798	5718000	1.69	2.22
3	13.0	370.0	104513	159798	5718000	1.69	2.22
4	14.0	370.0	104513	159798	5718000	1.69	2.22

E = 720000 ksf
E/G = 2.50

TABLE 3.7-24

DAMPING RATIOS FOR FLUID SYSTEM PIPING AND THEIR SUPPORTS ANALYZED BY NSSS VENDOR (SET A)

Item	Damping Ratio Percentage of Critical Viscous Damping	
	OBE	SSE
Reactor Coolant Loop	1	1
Auxiliary Piping Systems(2)	0.5	1
Welded Steel Structures	1	1-2(1)
Bolted Steel Structures	2	2-5(1)

Notes:

(1) Damping value used when stress levels are at or near yield.

(2) An as option, for some cases or piping response spectrum seismic analysis, variable damping of 5% to 10 hertz decreasing linearly to 2% at 20 hertz and remaining at 2% to 33 hertz was used for both OBE and SSE as described in ASME Code Case N-411.

TABLE 3.7-25 (Sheet 1 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Reactor Coolant System						
Reactor Vessel	X				See Section 5.2	
Full-length CRDM housing		X			"	
Part-length CRDM housing		X			"	
Reactor coolant pump		X			"	
Steam generator		X			"	
Pressurizer		X			"	
Reactor coolant piping to pressure boundary		X			"	
RC system supports		X			"	
Surge pipe and fittings		X			"	
Bypass manifold	X					
RC Thermowells				X	See Section 5.2	
Safety valves	X				"	
Relief valves	X				"	
Valves to RC system boundary	X				"	

TABLE 3.7-25 (Sheet 2 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
CRDM head adapter plugs		X			See Section 5.2	
Chemical and Volume Control System						
Regenerative HX		X			See Section 3.9	
Letdown HX		X			"	
Mixed-bed demineralizer		X			"	
Cation bed demineralizer		X			"	
Reactor coolant filter		X			"	
Volume control tank		X			"	
Charging/high head safety injection pump	X			X	See Section 3.9	Tests were run to determine natural frequency of the foundation system to meet seismic criteria.
Seal water injection filter		X			"	
Excess letdown HX		X			"	
Seal water return filter		X			"	
Seal water HX		X			"	
Boric acid tanks		X			"	
Boric acid filter		X			"	
Boric acid transfer pump	X				"	
Boric acid blender		X			"	

TABLE 3.7-25 (Sheet 3 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Chemical and Volume Control System						
Reactor makeup water storage tank		X			w/o exceeding 90% of yield stresses and/or loss of function	
Emergency Core Cooling System						
Accumulators		X			"	
Boron injection tank		X			"	
BIT recirculation pump	X				"	
Boron injection surge Tank		X			See Section 3.9	
Residual Heat Removal System						
Residual heat removal/low head safety injection pump	X				"	
Residual heat exchanger		X			"	
Containment Spray System						
Spray additive tank		X			"	
Containment spray pump	X				"	
Containment Isolation System						
Valves	X				See Section 3.9	

TABLE 3.7-25 (Sheet 4 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Containment Cooling System						
Fans		X			See Section 3.9	
Heat Exchanger		X			"	
Component Cooling System						
Pumps	X				"	
Heat exchangers		X			"	
Surge Tank		X			"	
Spent Fuel Pool Cooling System						
Spent fuel pool heat exchanger		X			"	
Spent fuel pool pump	X				"	
Boron Thermal Regeneration Subsystem						
Moderating HX		X			See Section 3.9	
Letdown chiller HX		X			"	
Letdown reheat HX		X			"	
Thermal regeneration demineralizer		X			"	

TABLE 3.7-25 (Sheet 5 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Liquid Recycle and Waste Subsystem						
Recycle holdup tank		X			See Section 3.9	Per API 650
Recycle evaporator feed pump	X				"	
Recycle evaporator feed demineralizer		X			"	
Recycle evaporator feed filter	X				"	
Recycle evaporator	X				"	
R.C. drain tank HX		X			"	
Waste holdup tank		X			"	
Waste evaporator feed filter	X				"	
Waste evaporator	X				"	
Spent resin storage tank		X			"	
Spent resin sluice pump	X				"	
Spent resin sluice filter	X				"	
Floor drain tank		X			"	
ES room sump pump	X				"	

TABLE 3.7-25 (Sheet 6 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Gas Handling Subsystem						
Gas compressor	X			X	See Section 3.9	Vibration tests were conducted to determine seismic capability
Gas decay tanks		X			"	
Emergency Diesel Fuel Oil System						
Transfer pumps		X			"	
Fuel oil tanks	X				"	
Service Water System						
Pumps		X			"	
Fuel Handling System						
Fuel manipulator crane		X			See Section 3.9	
Fuel transfer tube	X				"	
Underwater fuel conveyor car and rail system	X				"	
Fuel pool bridge crane		X			"	

TABLE 3.7-25 (Sheet 7 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Fuel Handling System						
Polar crane	X				See Section 3.9	
Crane supports	X				"	
Refueling Water System						
Storage tank		X			w/o exceeding 90% of yield stresses and/or loss of function	
Auxiliary Building Ventilation System						
ES air cooling units: Heat exchanger		X			w/o exceeding 90% of yield stresses and/or loss of function	
Fan		X			"	
Penetration Room Filtration System						
Fans		X			"	
Filters (HEPA and charcoal)		X			"	
Control Room Ventilation System						
Fans		X			"	
Filters		X			"	

TABLE 3.7-25 (Sheet 8 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Control Room Ventilation System						
Air Conditioning unit		X			See Section 7.1	
Heat exchanger		X			"	
Diesel Building Ventilation System						
Fans				X	w/o exceeding 90% of yield stresses and/or loss of function	
Filters				X	"	
Main Steam System						
Isolation valves	X				"	
Auxiliary Feedwater System						
Auxiliary feedwater pumps motordriven, steam turbine driven		X			"	
Condensate storage tank		X			"	
Steam Dump Systems						
Relief Valves	X				"	
Safety Valves	X				"	

TABLE 3.7-25 (Sheet 9 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Electrical Components and Systems						
6.9 kV shutdown boards (engineered safe-guard buses)				X	w/o exceeding 90% of yield stresses and/or loss of function	
6.9 kV to 480 V transformers (associated with engineered safeguard systems)		X			"	
480 V shutdown boards (engineered safeguard systems buses)				X	"	Test on prototype
480-V motor-control centers (associated with engineered safeguard systems)				X	"	Test on prototype
125-Vdc vital batteries				X	"	Test on prototype
120-Vac vital inverters, (associated with vital instrument buses)				X	See Section 7.1	
125-Vdc battery boards				X	See Section 7.1	Tests on two panels selected at random
120-Vac vital instrument power boards				X	"	

TABLE 3.7-25 (Sheet 10 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Electrical Components and Systems						
125-v dc switchgear				X	See Section 7.1	Tests on prototype
125-v dc battery chargers				X	"	Test on one charger
Solid-state protection system cabinets				X	"	
Reactor trip switchgear				X	"	
Nuclear instrumentation system cabinets				X	"	
Process protection and control system cabinets				X	"	
Cable tray supports (associated with engineered safeguard system)		X			"	
Auxiliary relay racks				X	"	
Containment penetration assemblies				X	w/o exceeding 90% of yield stresses and/or loss of function	Test on one medium voltage penetration assembly plus test on a composite assembly comprised of 1 1000-v dc power and 600-v control and instrument cables

TABLE 3.7-25 (Sheet 11 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Electrical Components and Systems						
Emergency power board		X		X	w/o exceeding 90% of yield stresses and/or loss of function	Instruments and switches are tested
Direct-current emergency lighting				X	"	Test on prototype
Diesel generators		X			"	
Diesel generator control panels				X	"	
Diesel generator sequencers				X	"	Test on one panel
Boric acid heat-tracing equipment		X			"	
Balance of plant instrument cabinets and equipment contained therein				X	w/o loss of function	
Equipment contained within balance of plant instrument cabinets		X		X	w/o loss of function	
Containment purge radiation monitors		X		X	w/o loss of function	
Fuel handling area radiation monitors		X		X	w/o loss of function	

TABLE 3.7-25 (Sheet 12 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Sampling System						
1. Cabinet		X			w/o exceeding 90% of yield stresses and w/o loss of function	
2. Tubing, valves, coolers, sample vessels	X				w/o loss of function	
Electrical Components and Systems						
Balance of plant field mounted instruments	X			X	w/o loss of function	
Instrument valves for field mounted instruments	X				w/o loss of function	
Instrument lines for field mounted instruments	X				w/o exceeding code allowable stresses	

TABLE 3.7-26

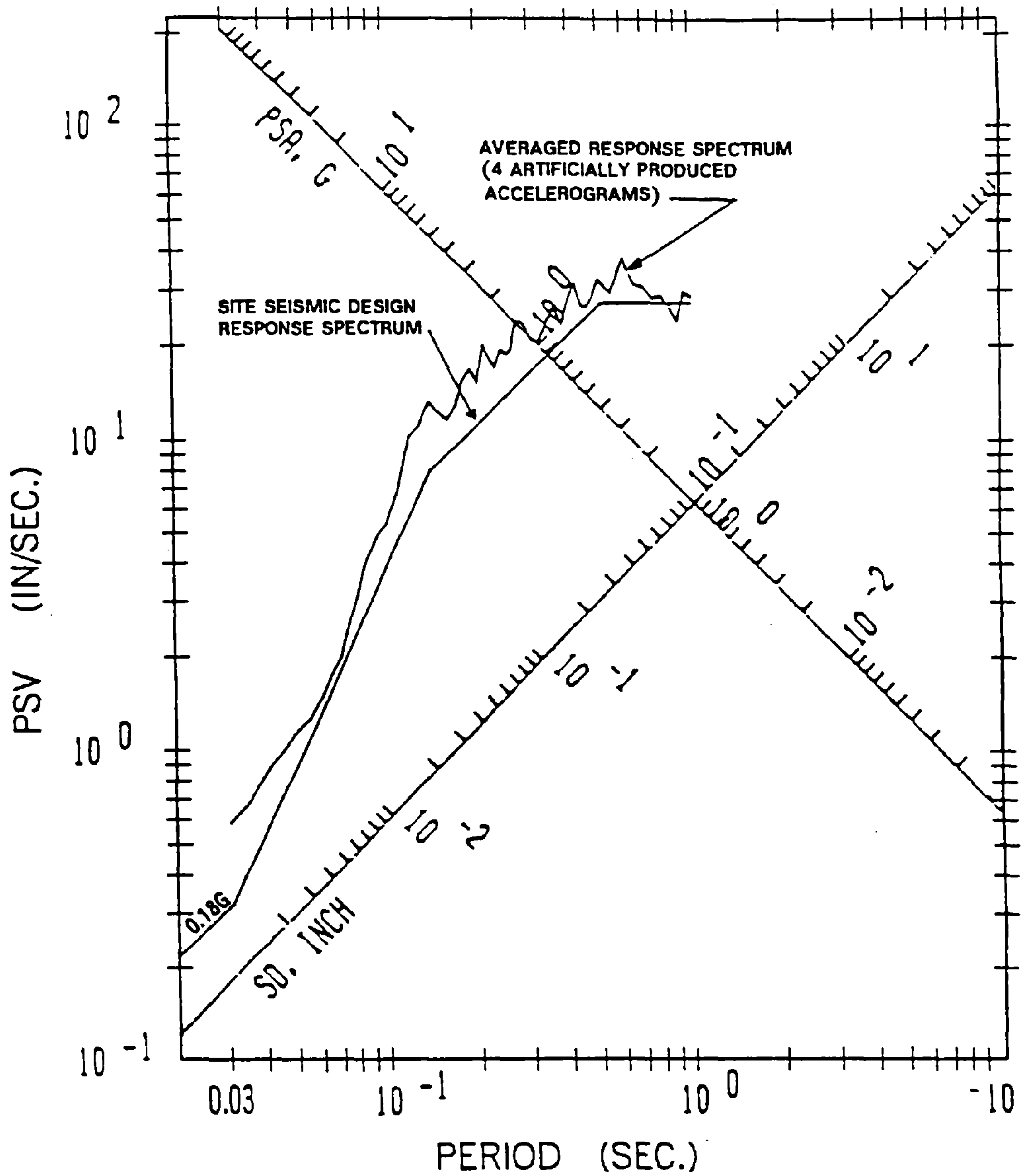
ALLOWABLE STRESSES FOR DUCT SUPPORTS

Elements	Load Combination[1]	Allowables
Component Standard Supports	(1) and (2)	Factor of Safety of 5 against ultimate strength
	(3), (4), and (5)	0.90 F_y (F_y = minimum specified yield stress) or a minimum factor of safety of 2.5 against ultimate strength
Steel Structural Members and Connecting Welds (Linear Supports)	(1)	AISC allowables
	(2)	1.5 x AISC allowables but less than 0.90 F_y[2]
	(3) and (4)	1.6 x AISC allowables but less than 0.90 F_y[2]
	(5)	1.7 x AISC allowables but less than 0.90 F_y[2]
Anchorage in Hardened Concrete		
Expansion Anchors	(1), (2), (3), (4), and (5)	Factor of Safety on minimum anchor ultimate tensile capacity
(a) Shell Types (SSD & SDI)		5
(b) Other Types (Wedge)		4

NOTES:

(1) In applying the above load combinations for design, dead load and thermal effects may be combined directly, accounting for their signs. Seismic loads are reversing and their effects must be combined without sign with the other loads. The latter is also true for DBA loads. (See Section 3.7.3.17.3 for definition of loads and their combinations.)

(2) But less than 0.52 F_y for shear stresses, and less than 0.90 F_{cr} for critical buckling stresses.

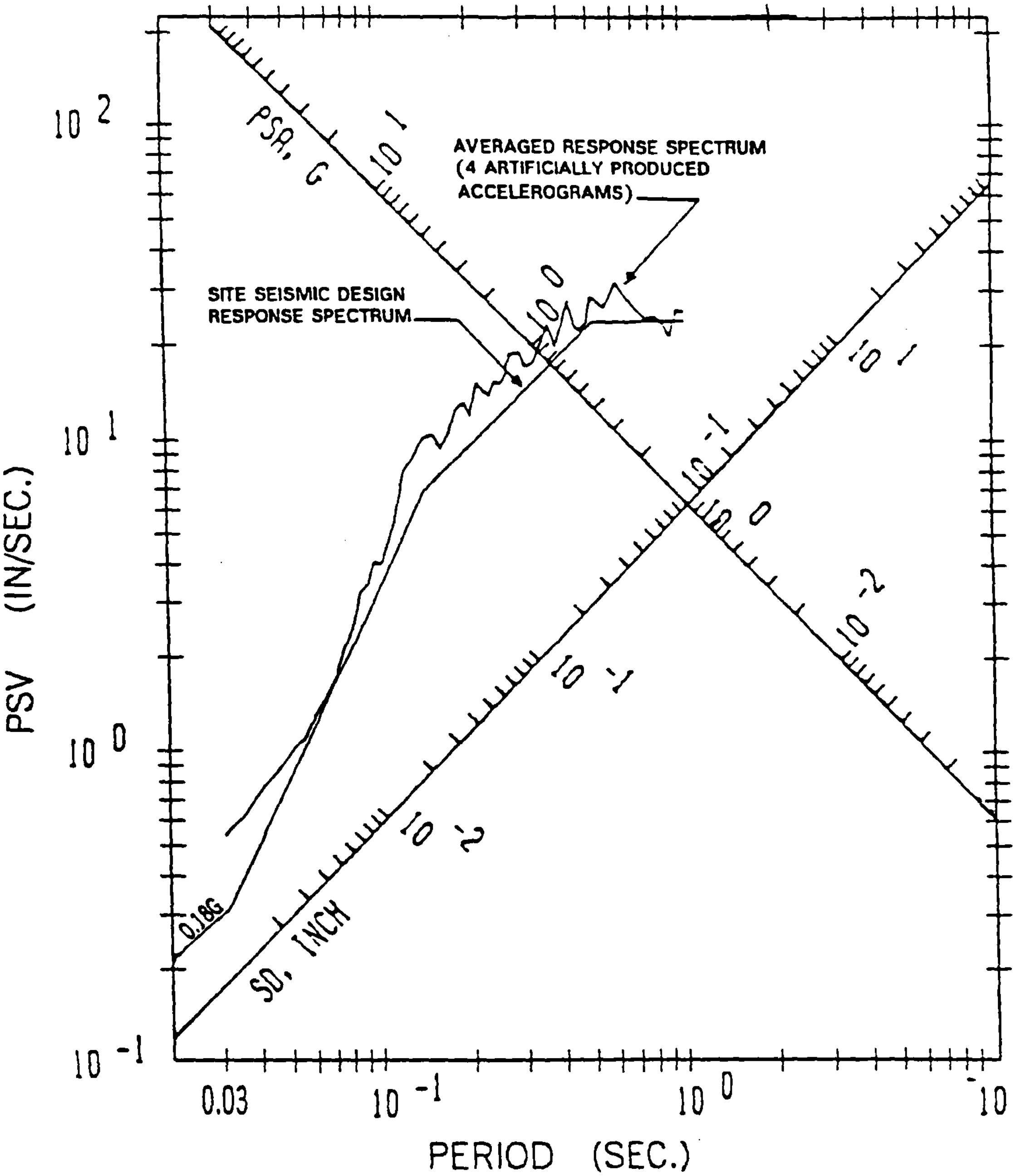


SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
1/2% DAMPING

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

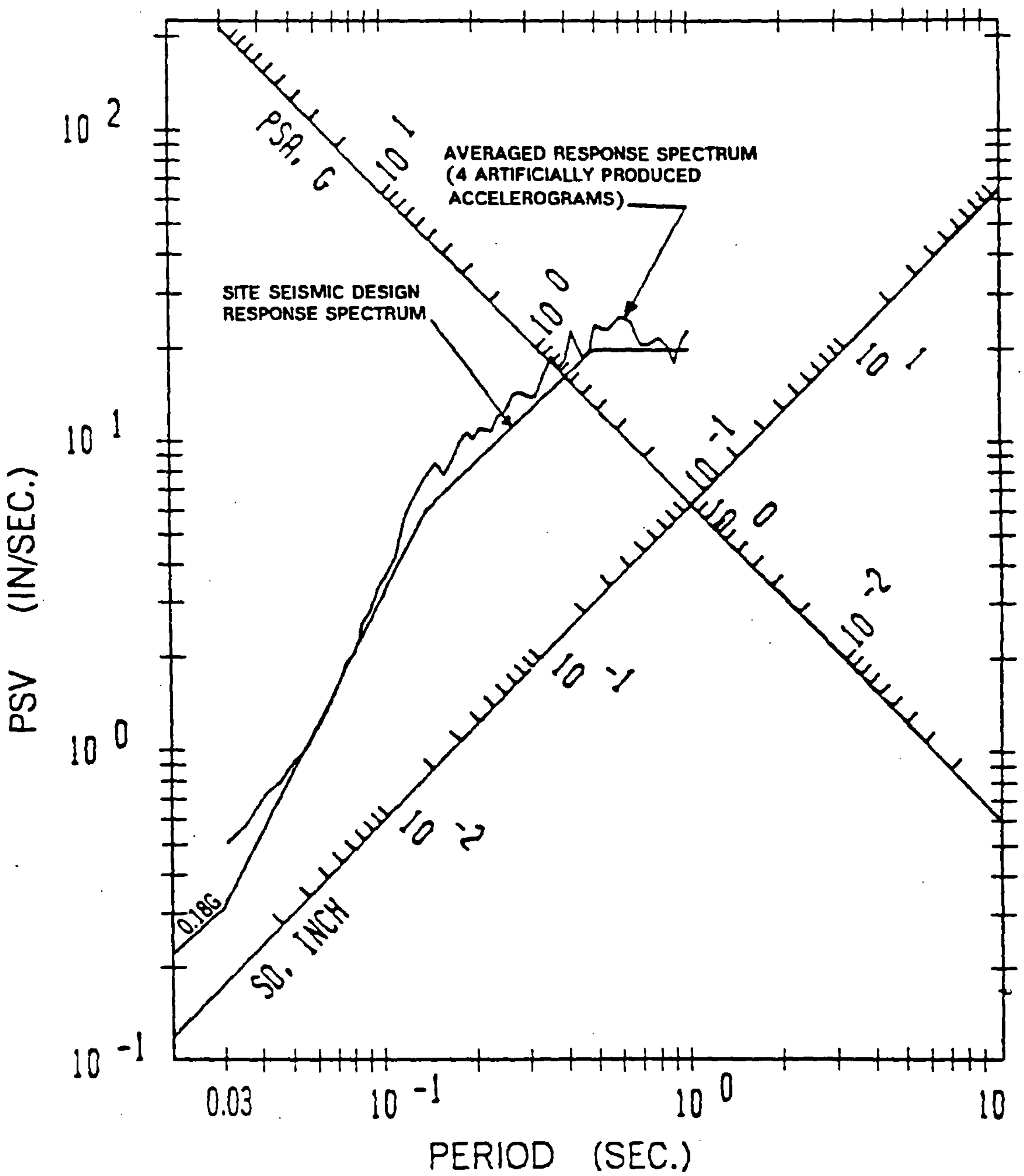
Response Spectra

FIGURE 3.7-1



SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
1% DAMPING

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Response Spectra FIGURE 3.7-2

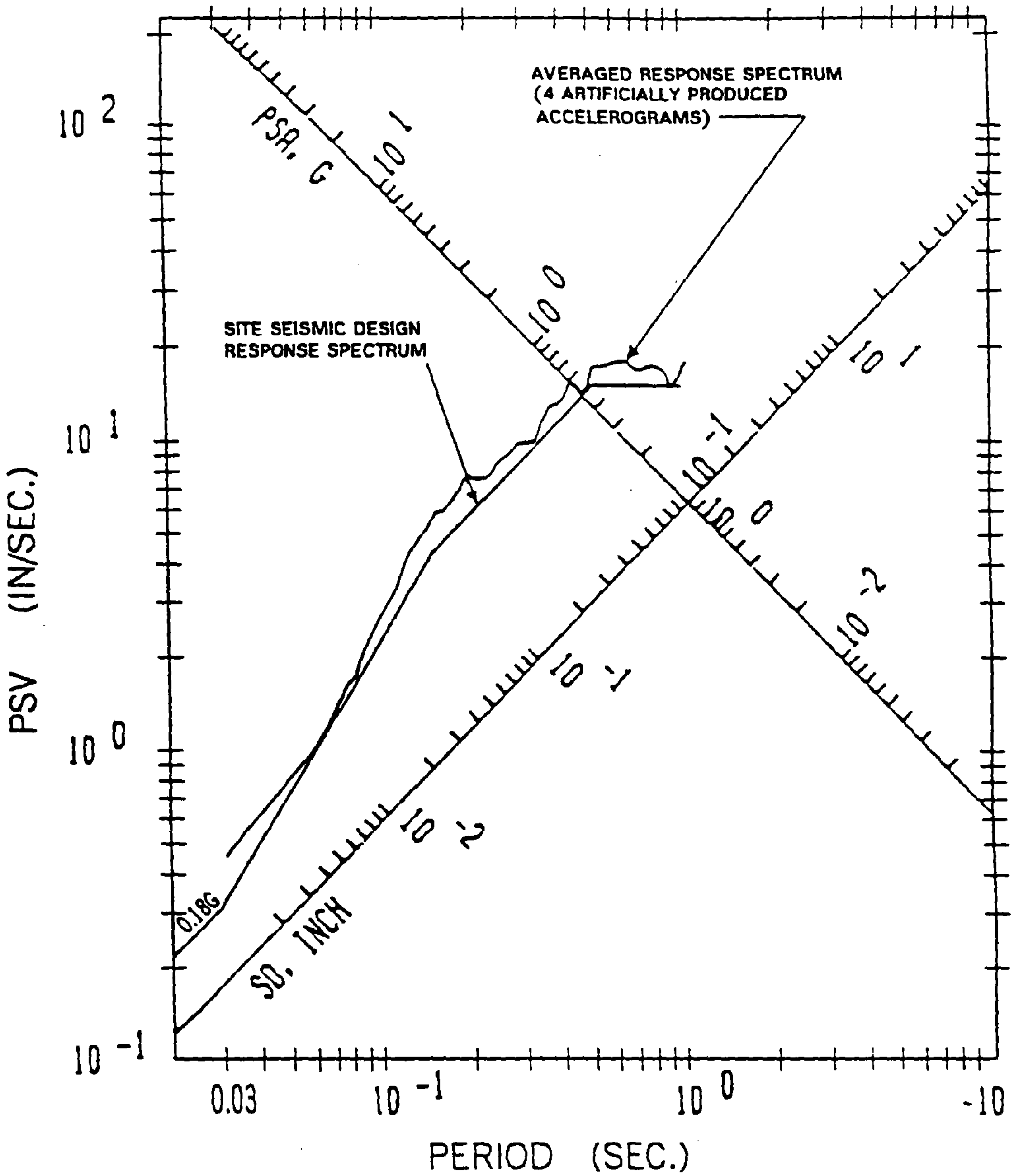


SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
2 % DAMPING

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Response Spectra

FIGURE 3.7-3

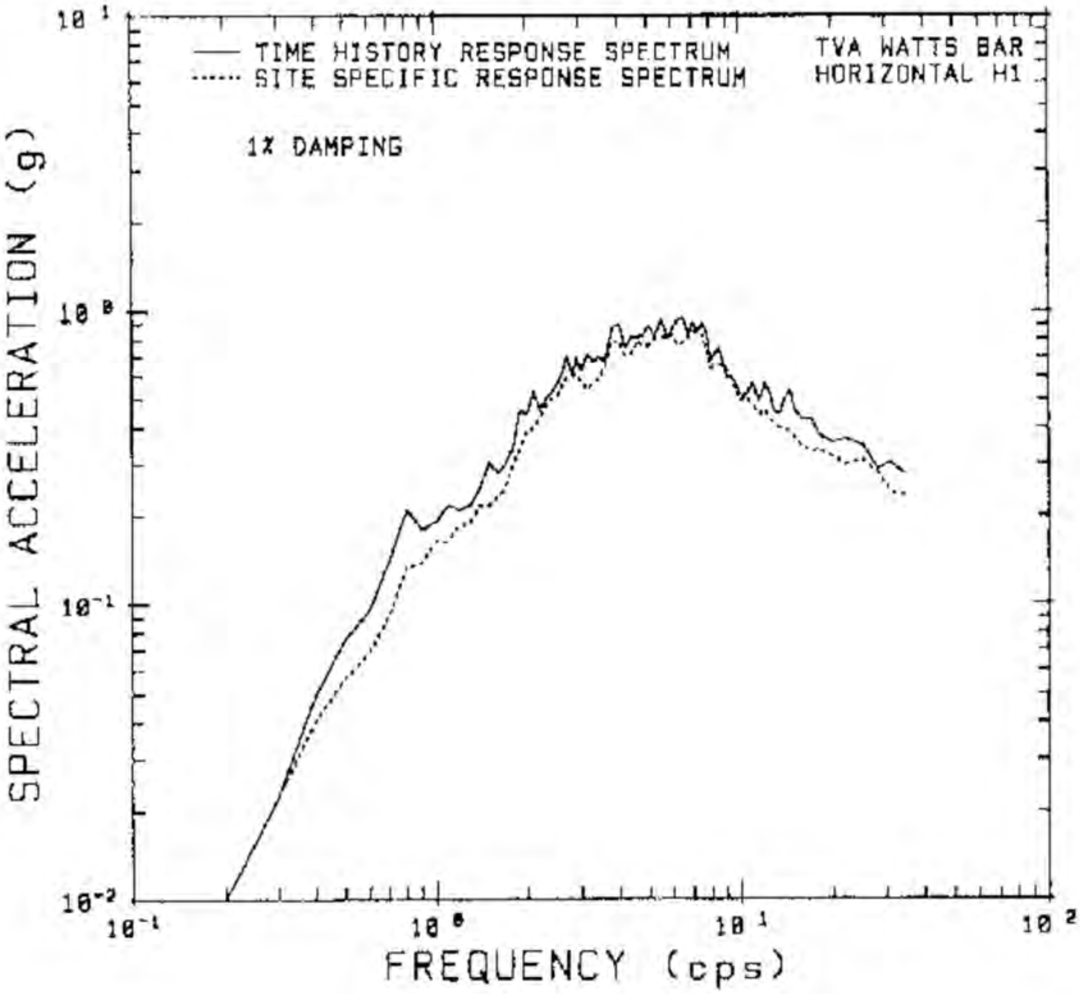


SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
5% DAMPING

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Response Spectra

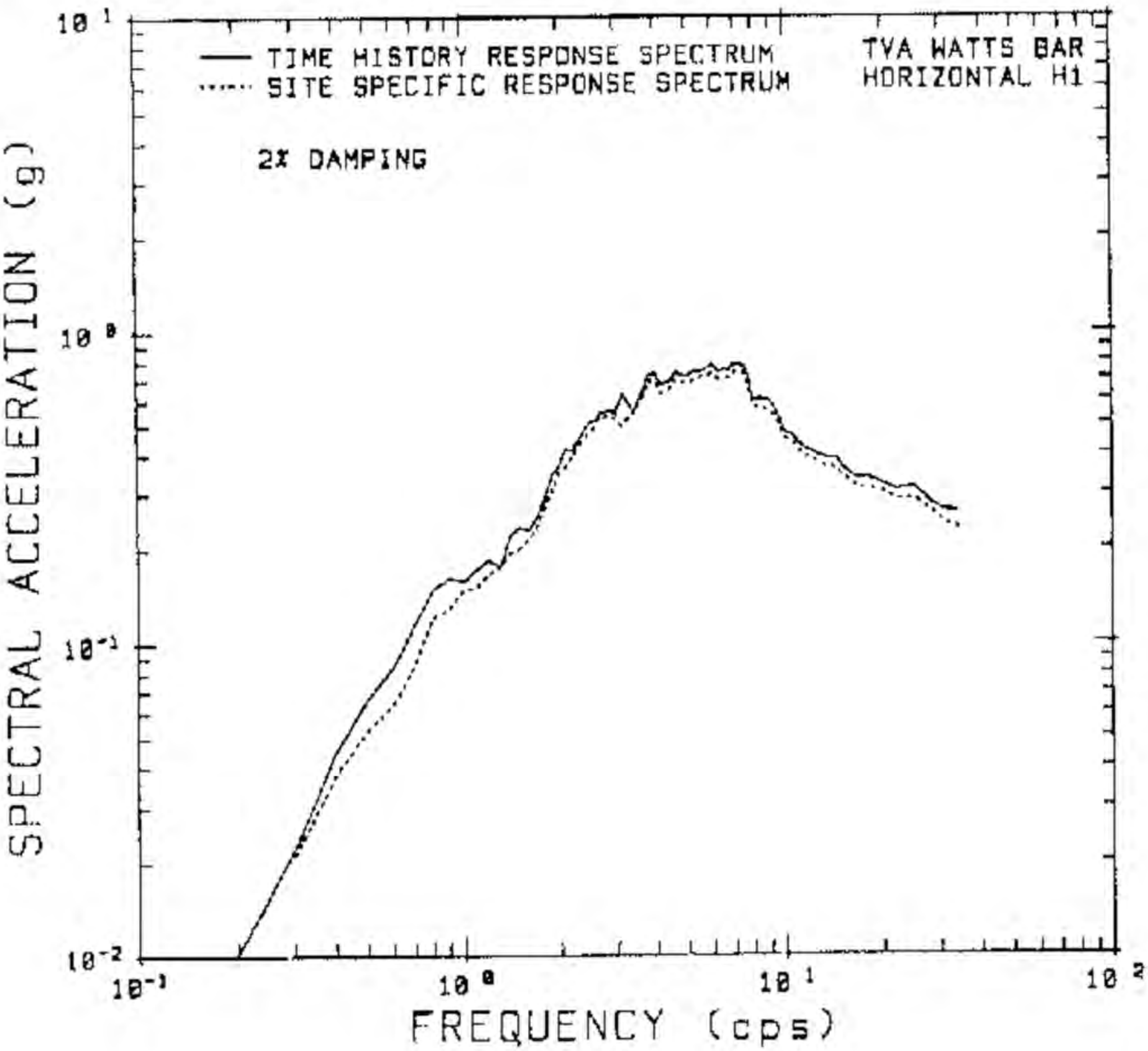
FIGURE 3.7-4



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
1% DAMPING

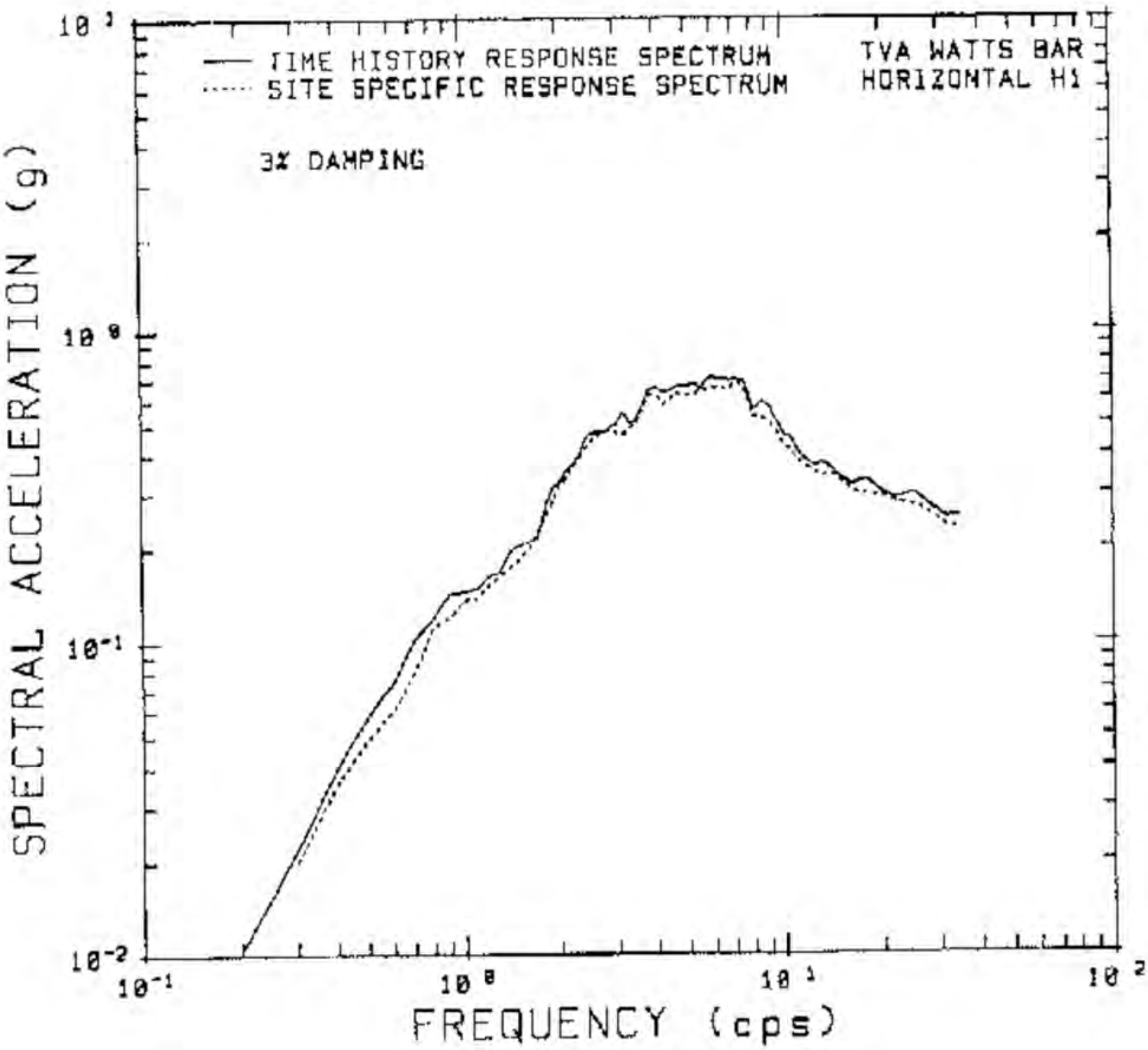
FIGURE 3.7-4a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
2% DAMPING

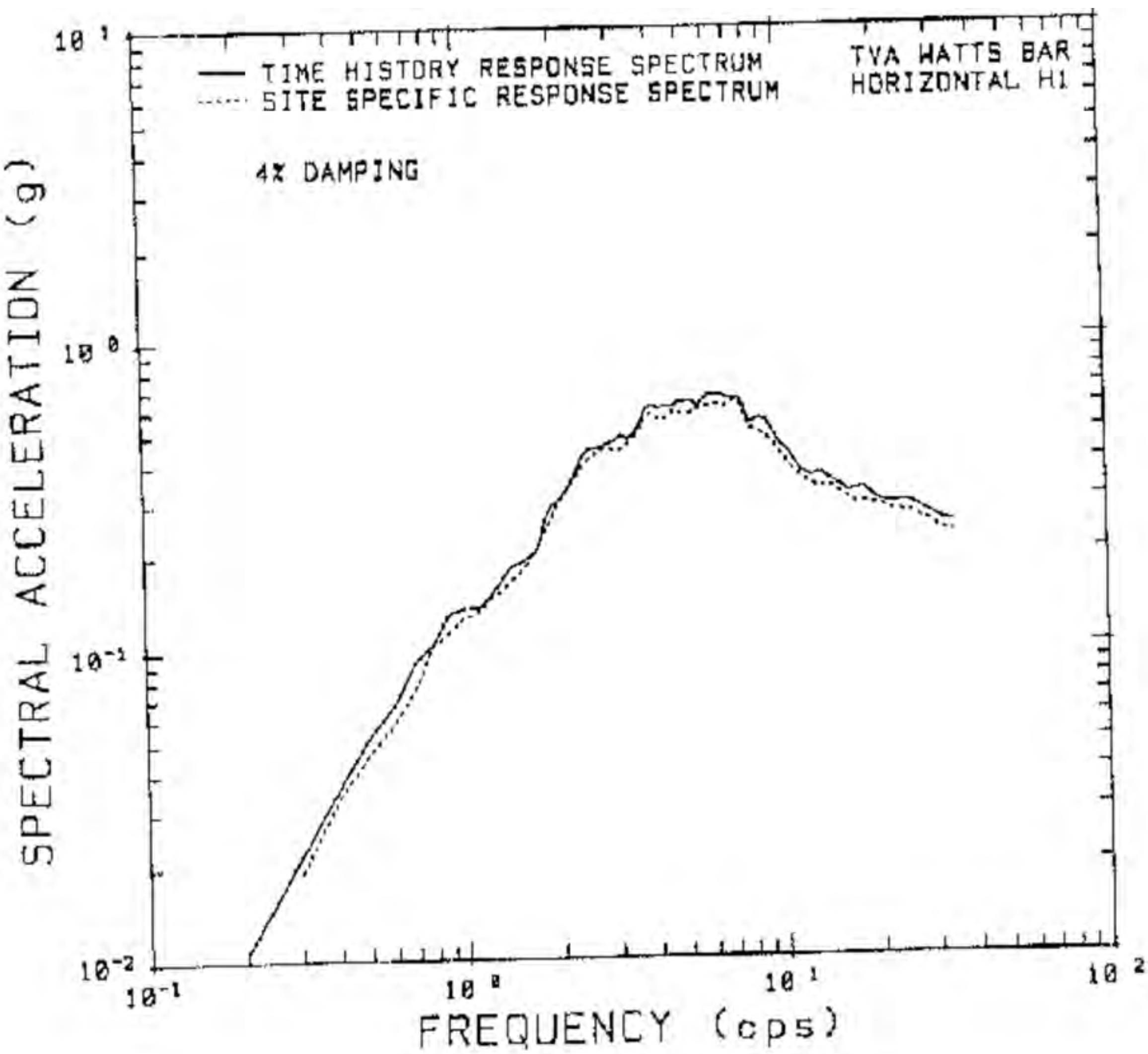
FIGURE 3.7-4b



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM SAFE SHUTDOWN EARTHQUAKE (N-S) ROCK SUPPORTED STRUCTURES 3% DAMPING

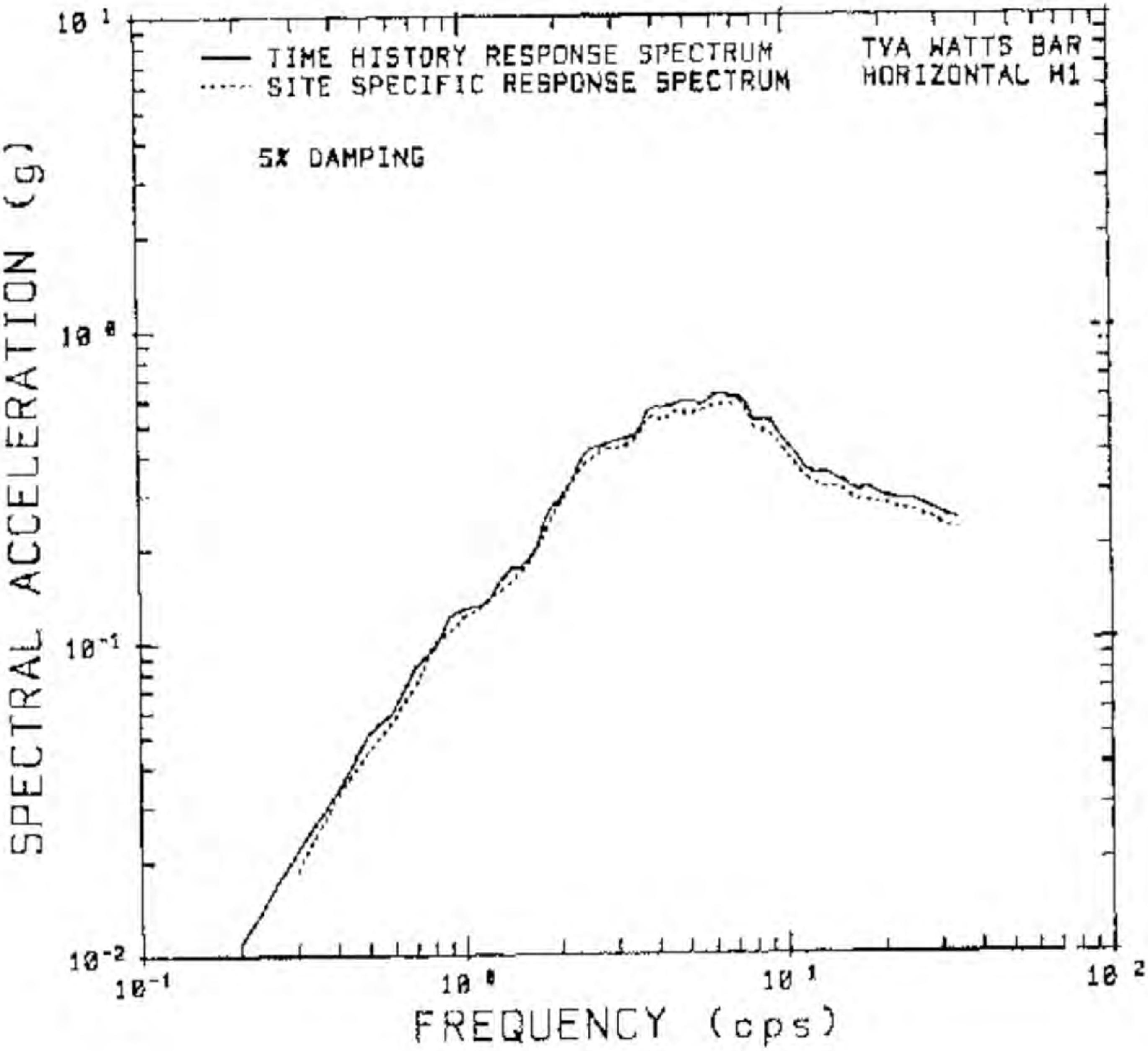
FIGURE 3.7-4c



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM SAFE SHUTDOWN EARTHQUAKE (N-S) ROCK SUPPORTED STRUCTURES 4% DAMPING

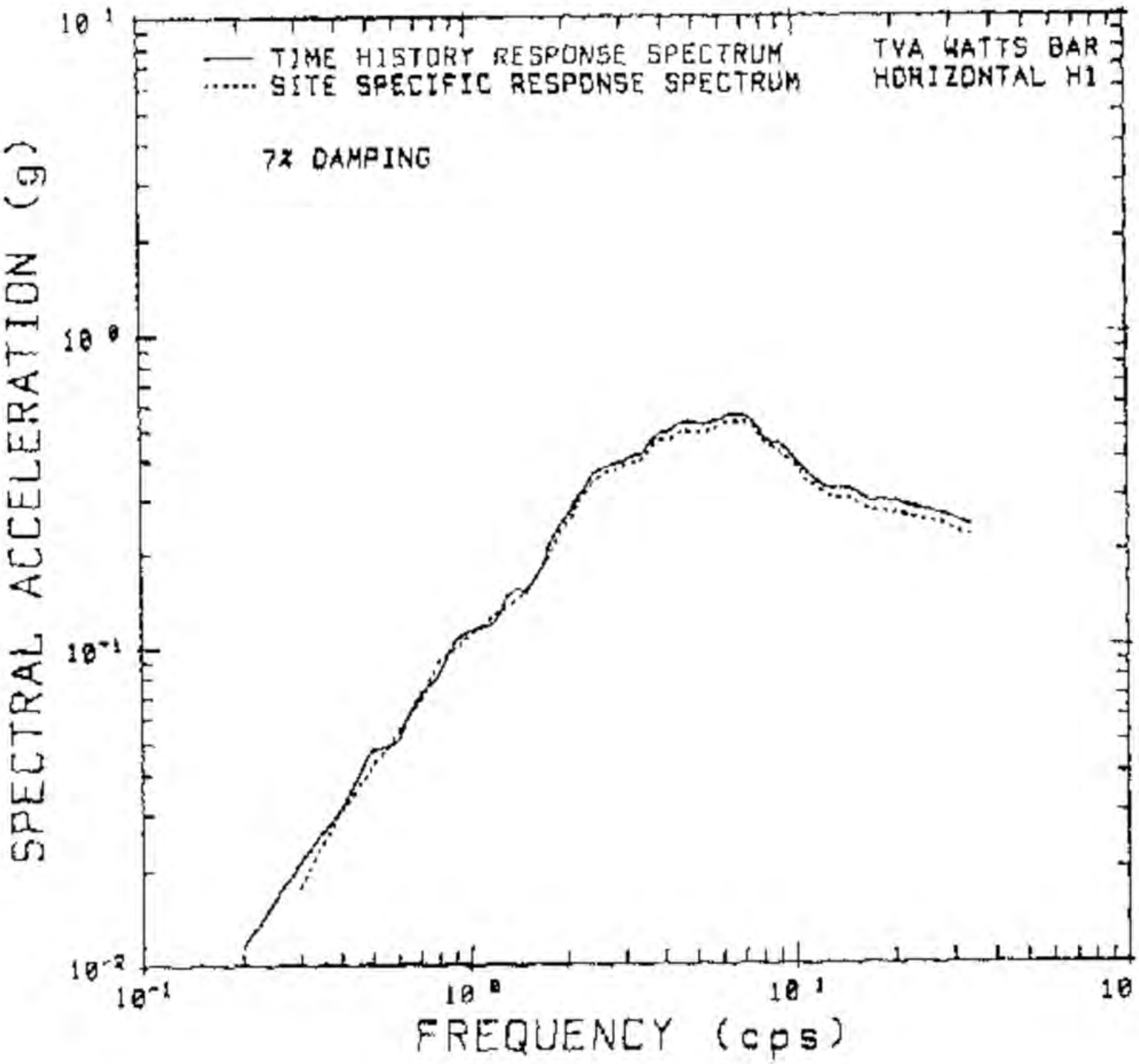
FIGURE 3.7-4d



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
5% DAMPING

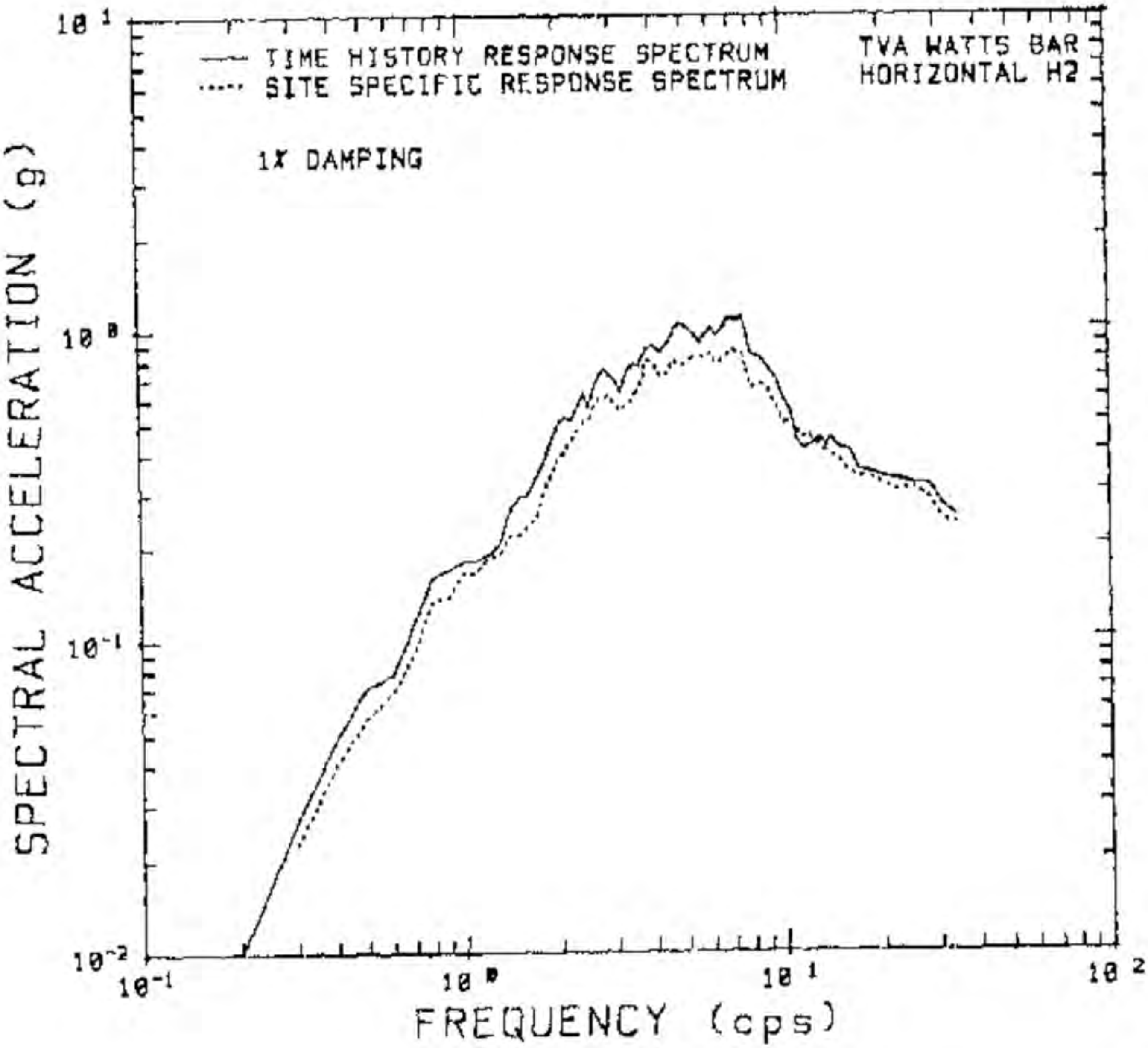
FIGURE 3.7-4e



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
7% DAMPING

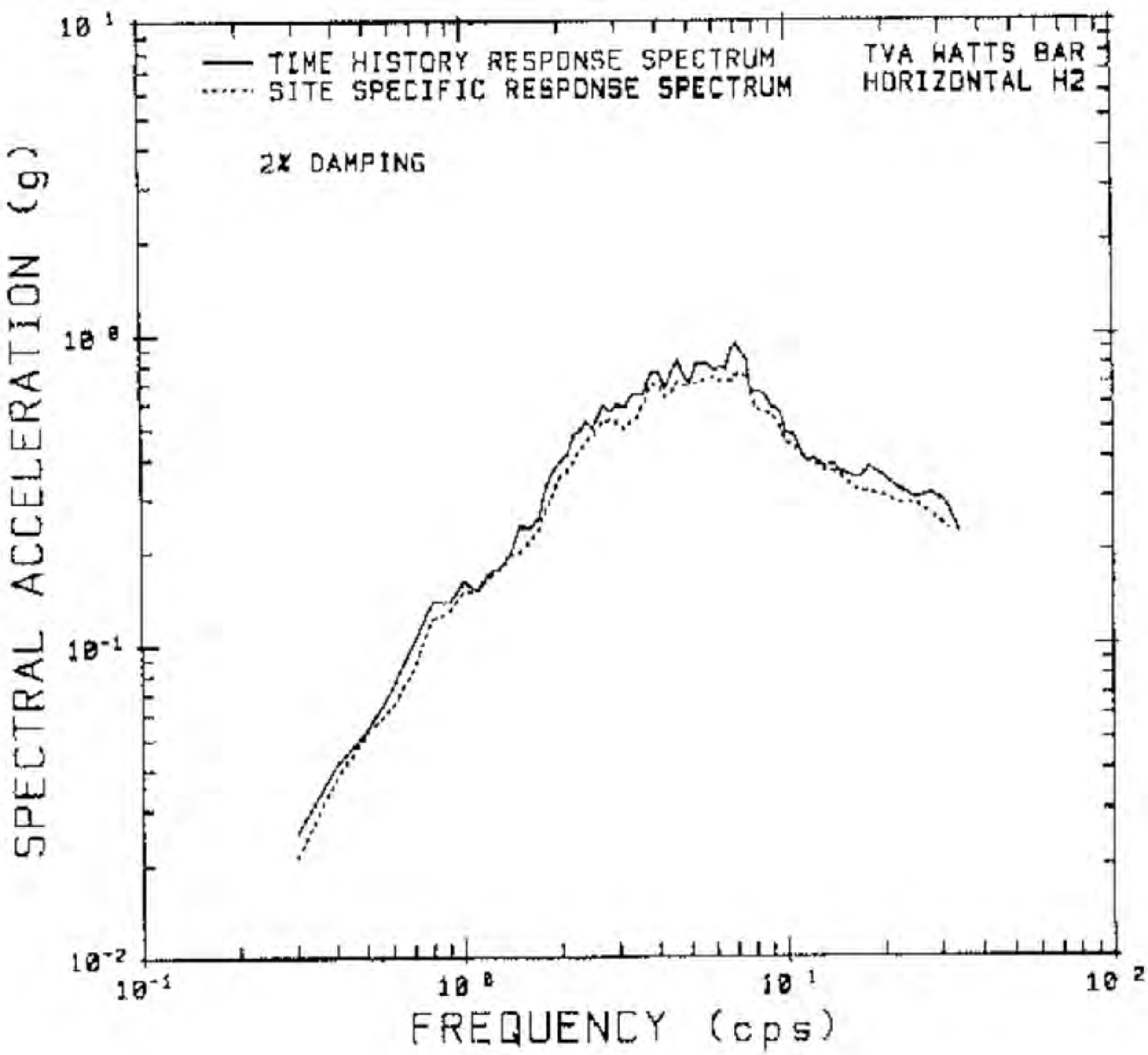
FIGURE 3.7-4f



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
1% DAMPING

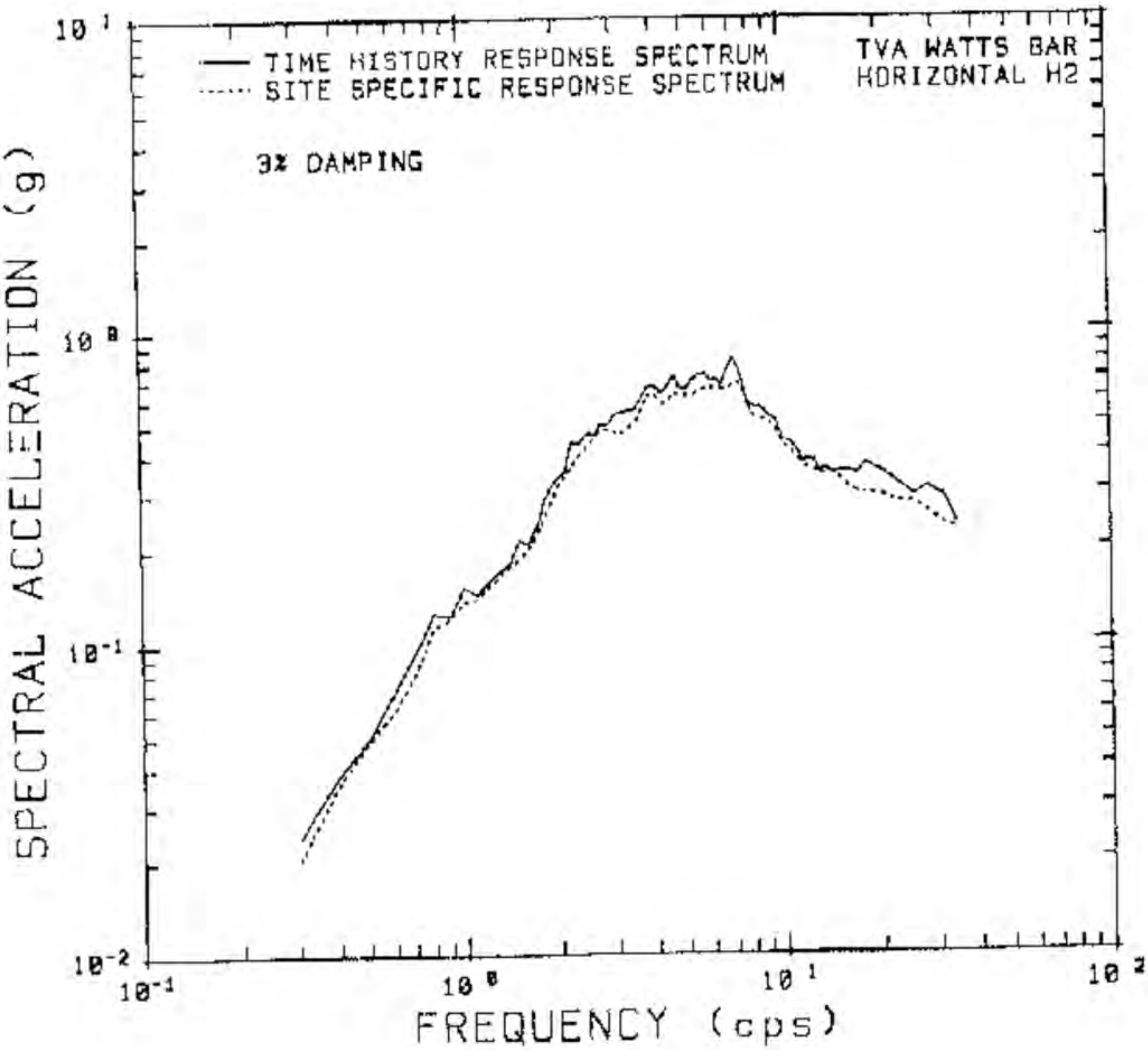
FIGURE 3.7-4g



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
2% DAMPING

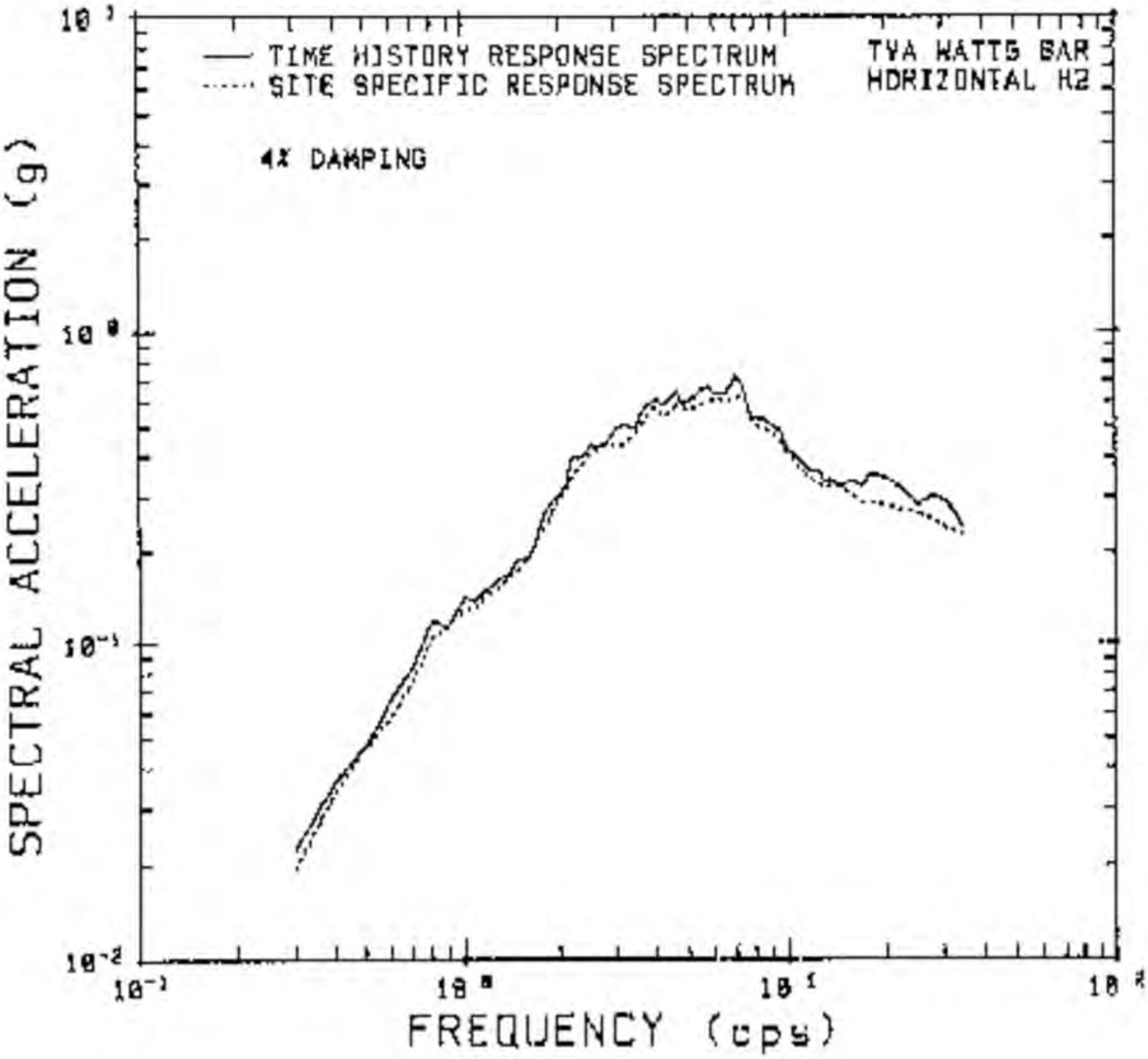
FIGURE 3.7-4h



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
3% DAMPING

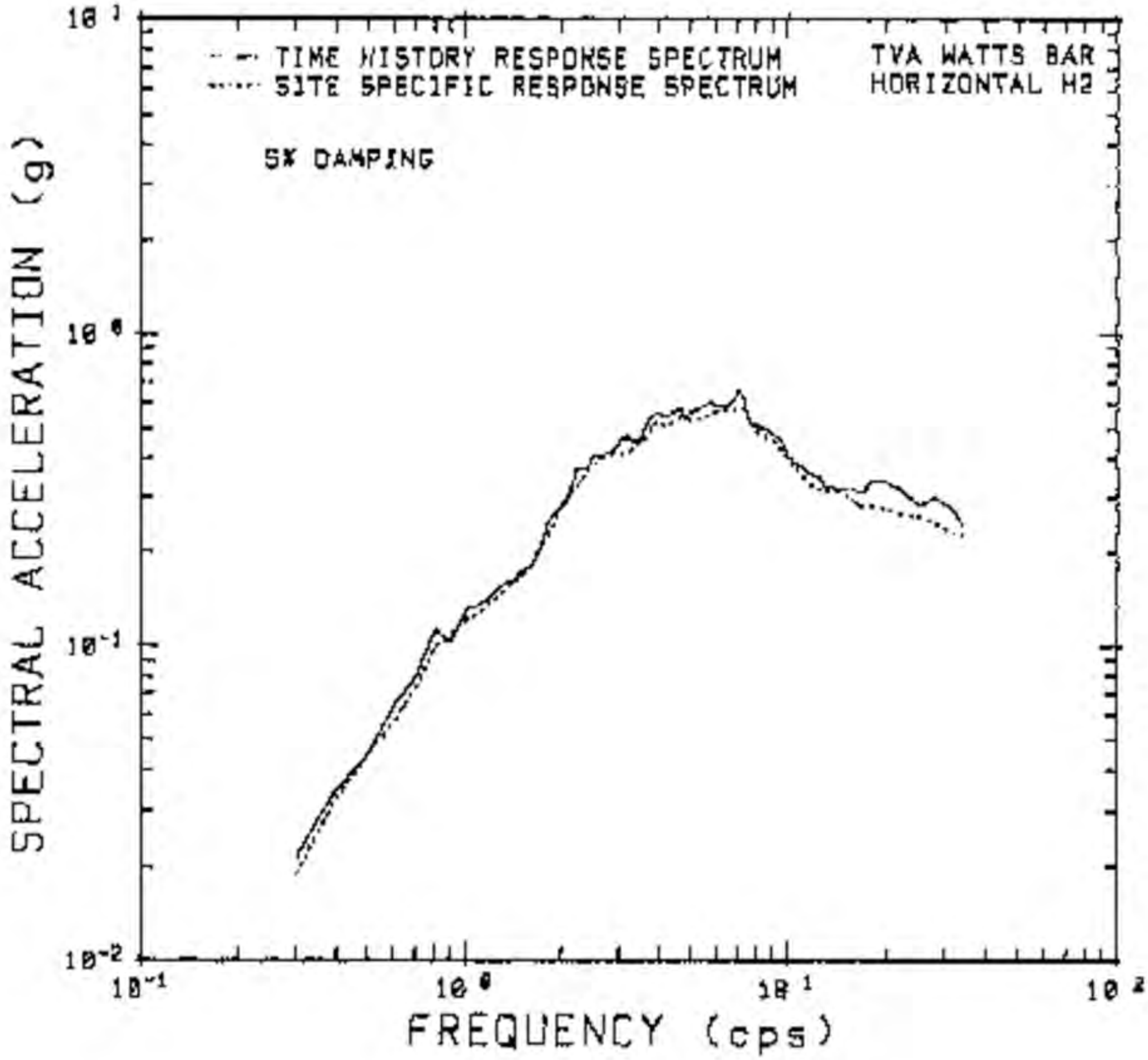
FIGURE 3.7-4i



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
4% DAMPING

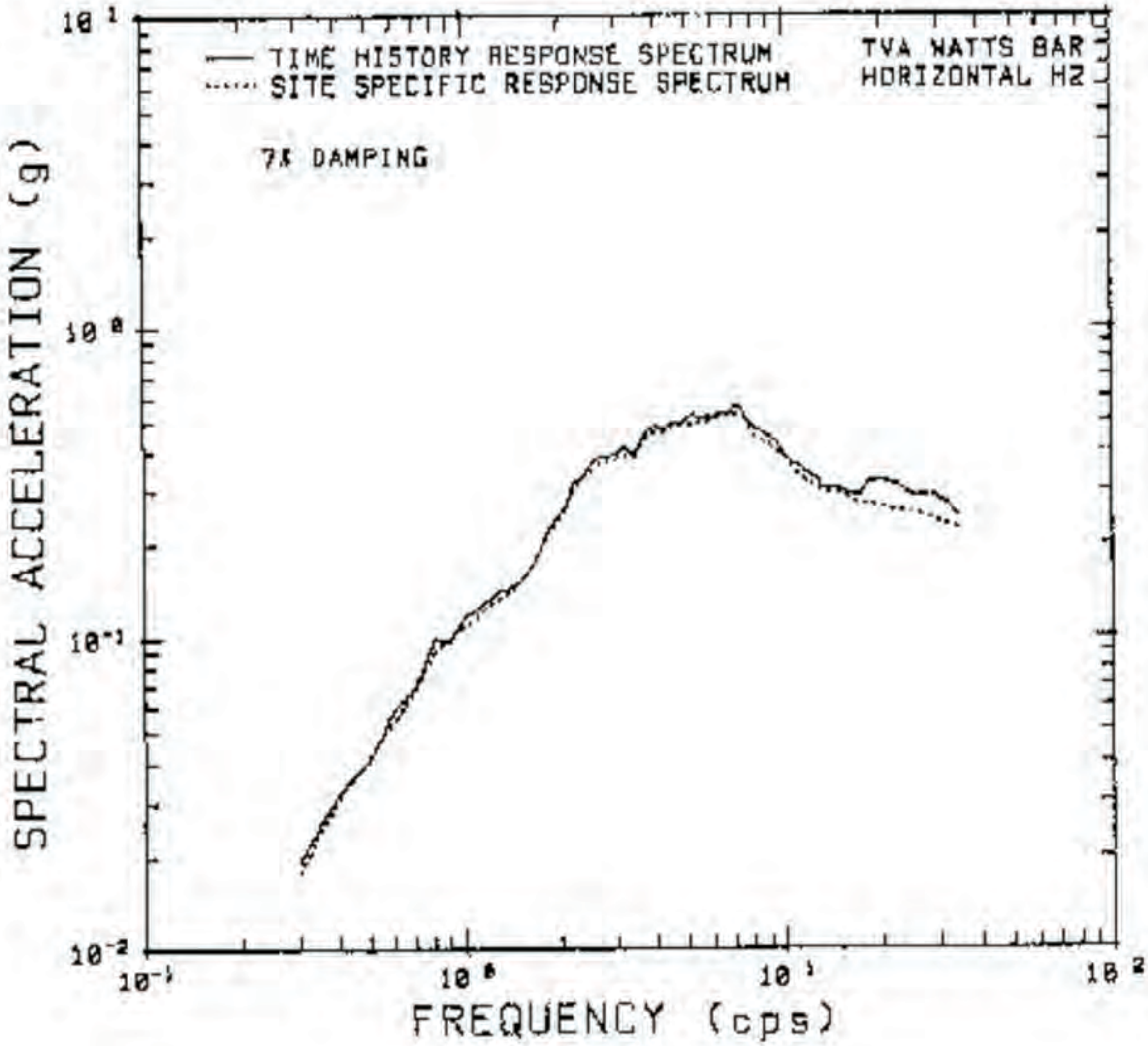
FIGURE 3.7-4j



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
5% DAMPING

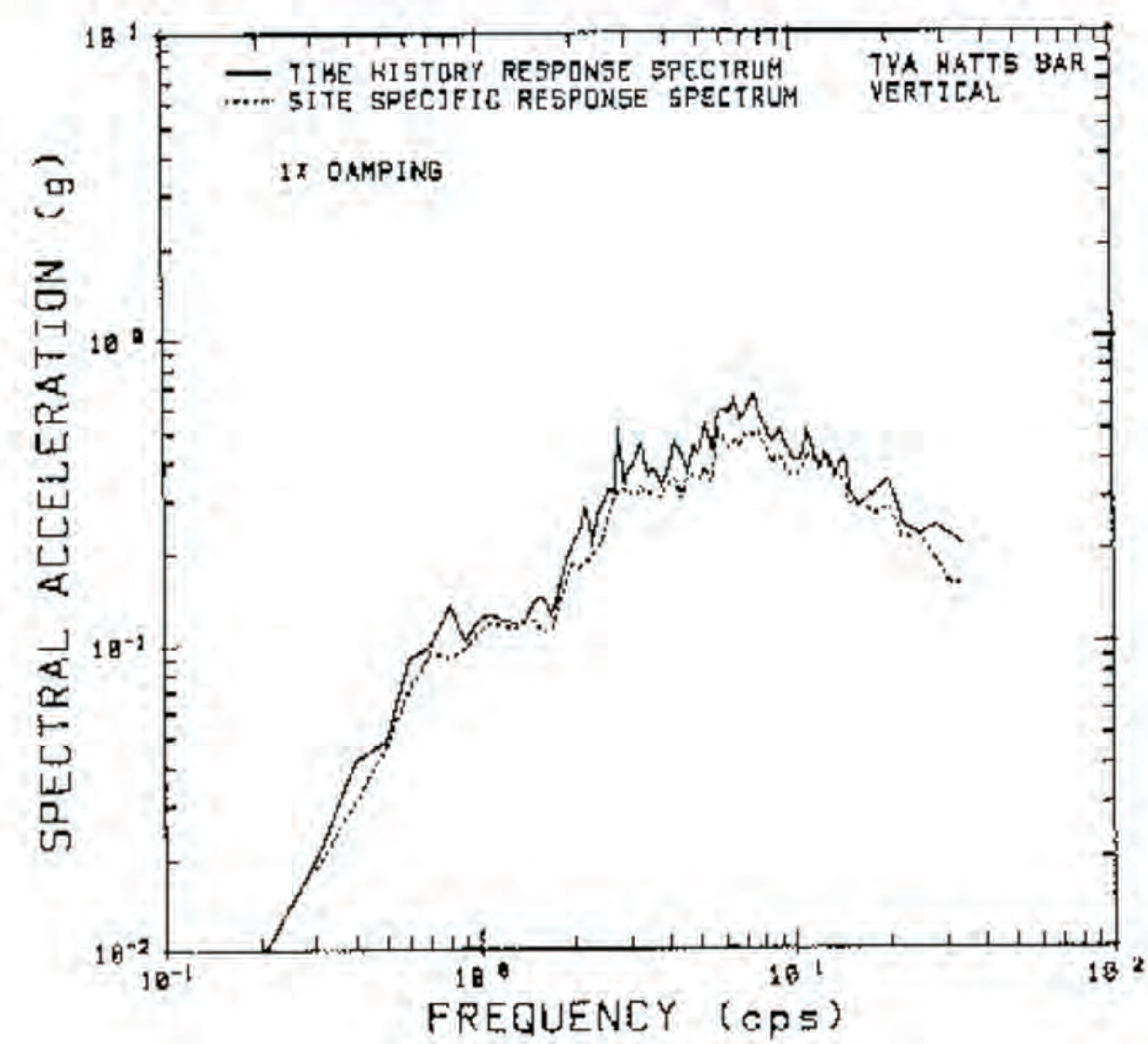
FIGURE 3.7-4k



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
7% DAMPING

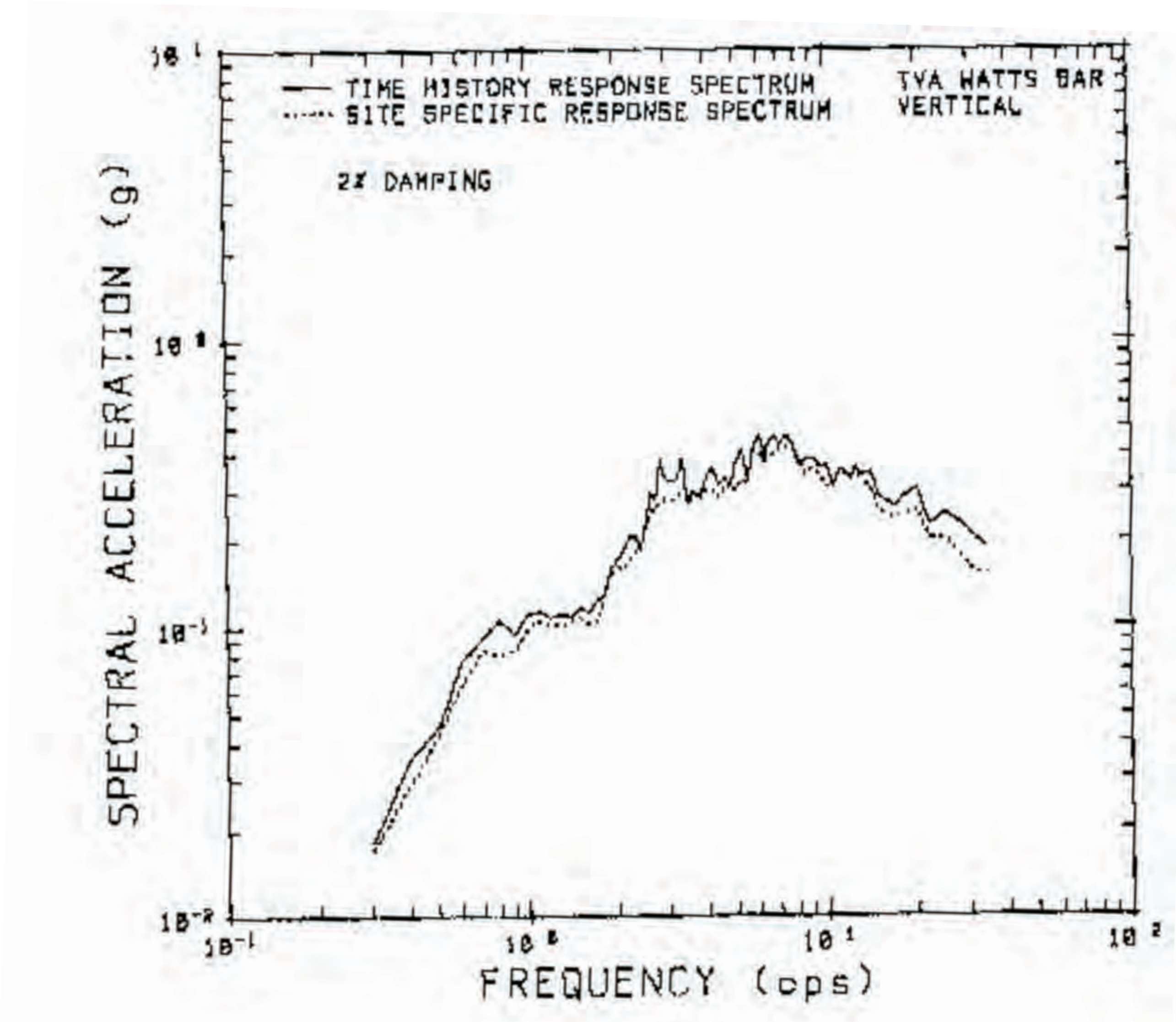
FIGURE 3.7-4l



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
1% DAMPING

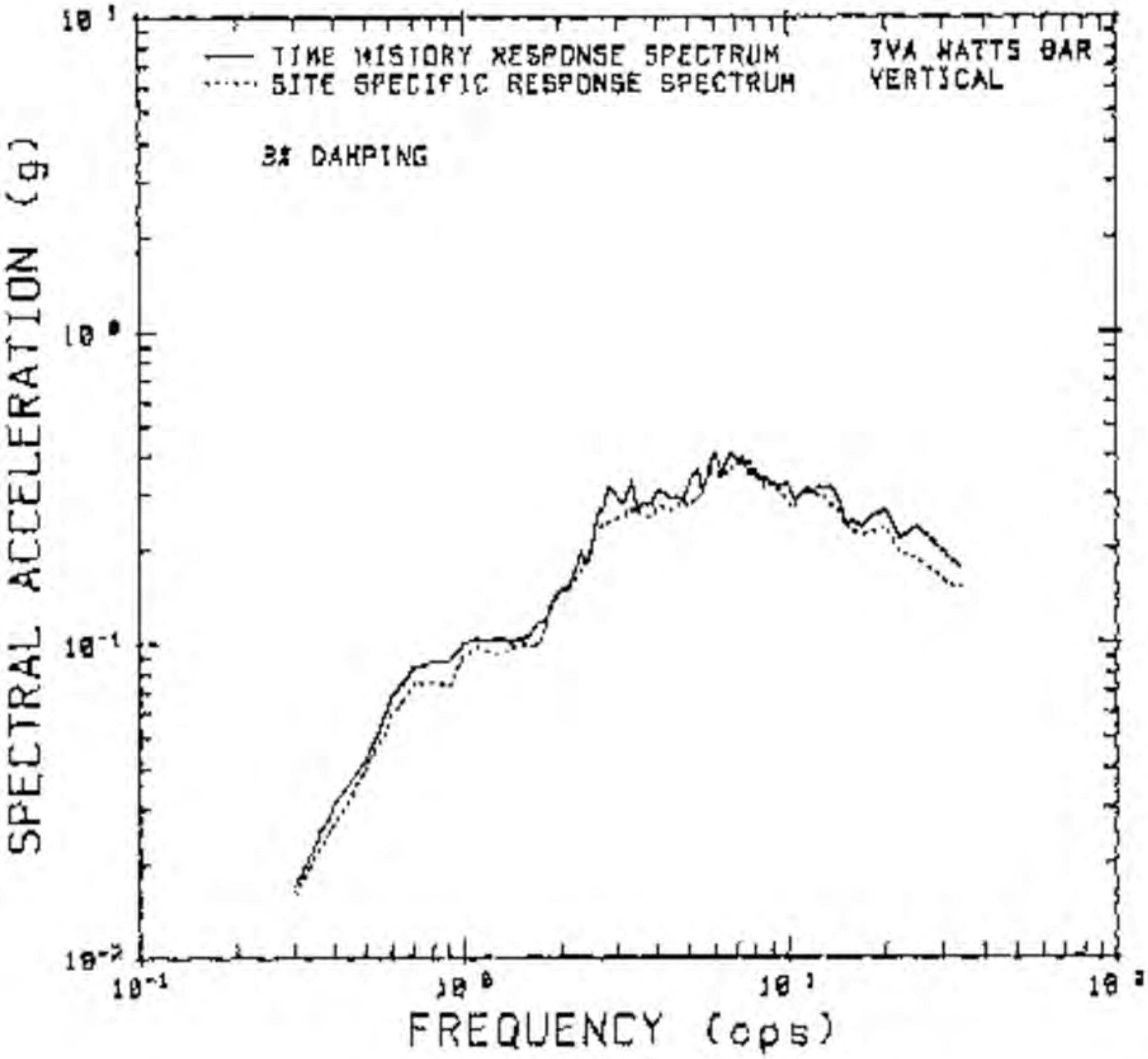
FIGURE 3.7-4m



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
2% DAMPING

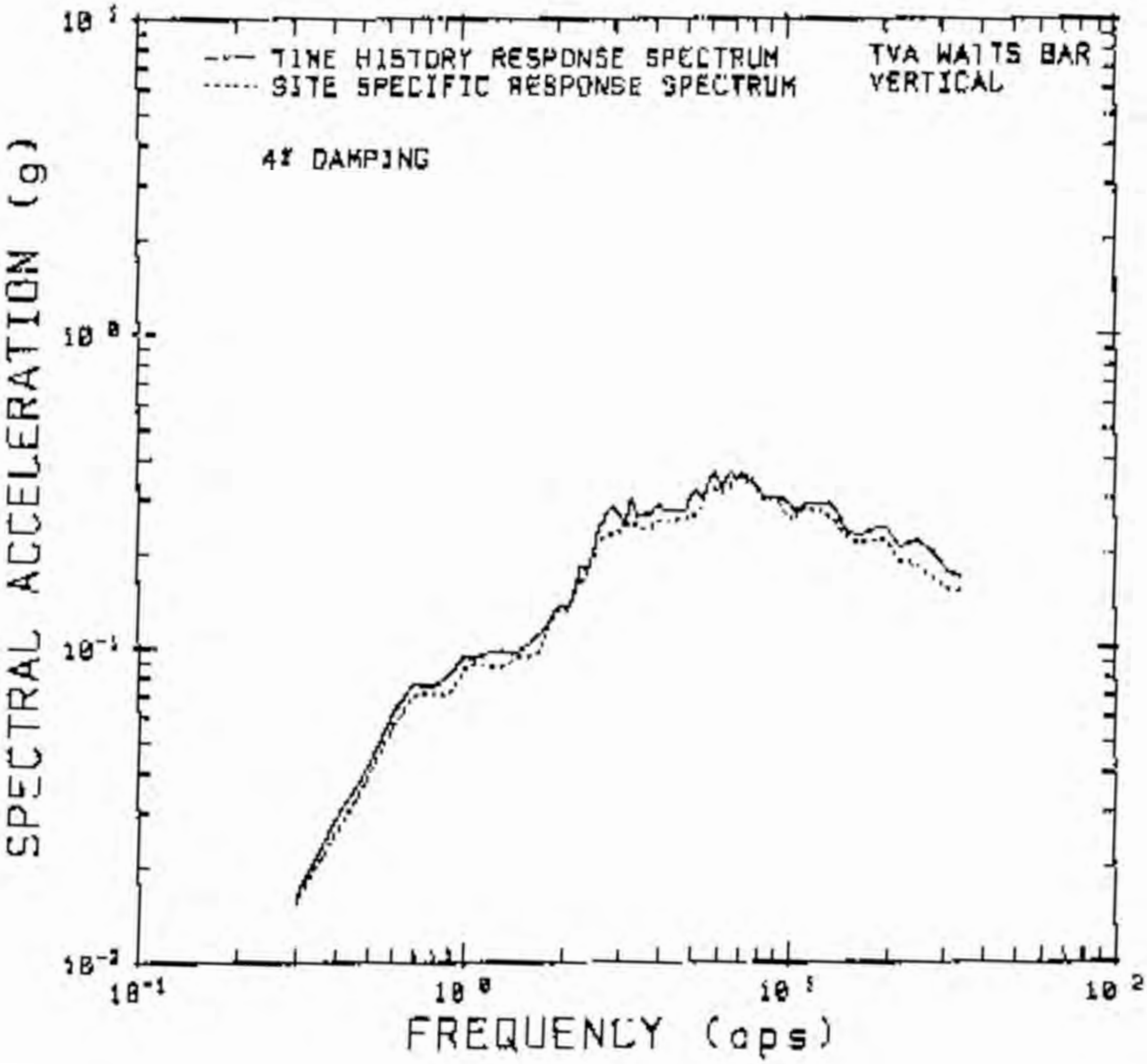
FIGURE 3.7-4n



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
3% DAMPING

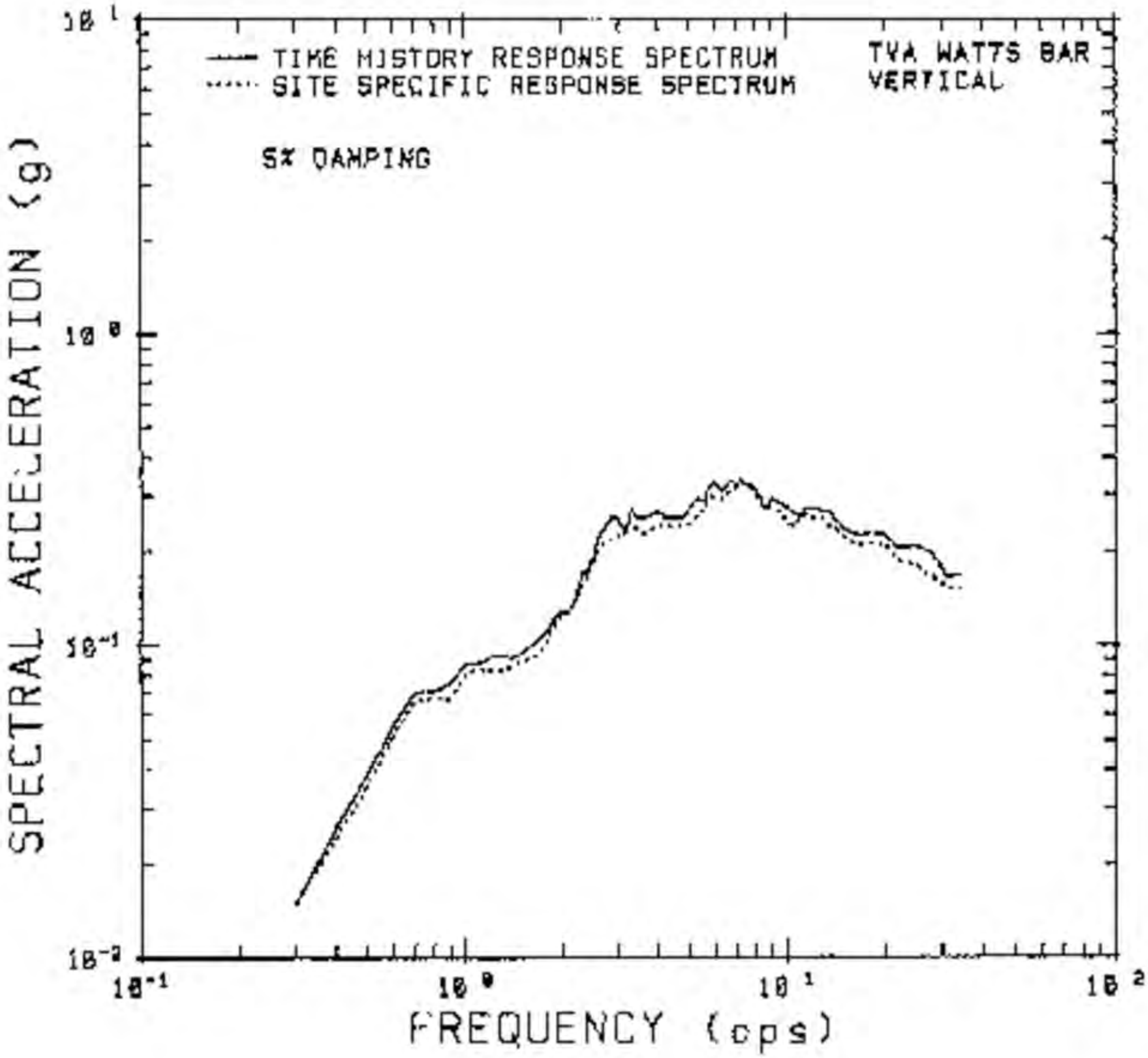
FIGURE 3.7-4o



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
4% DAMPING

FIGURE 3.7-4p



WATTS BAR NUCLEAR PLANT
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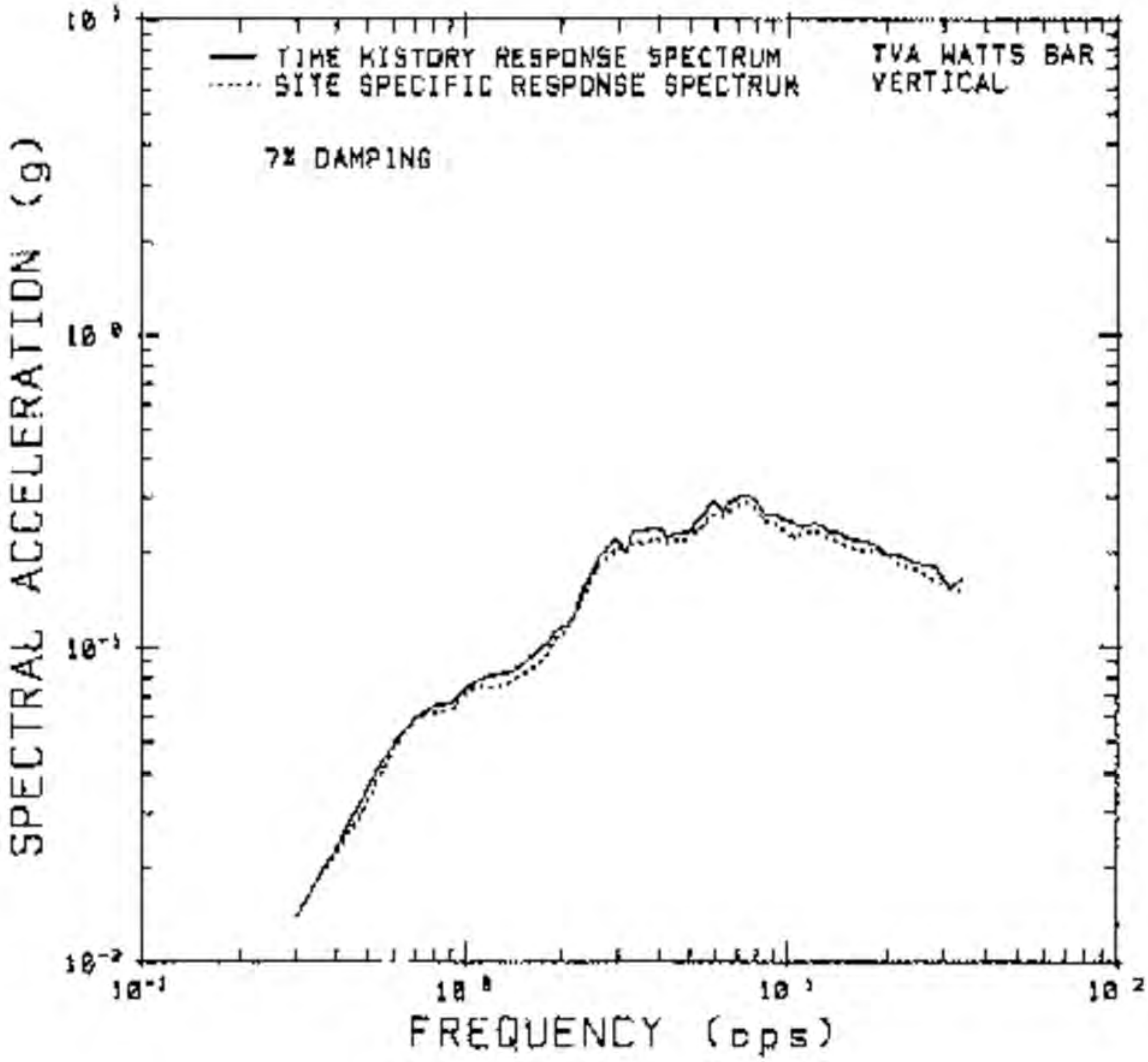
SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
5% DAMPING

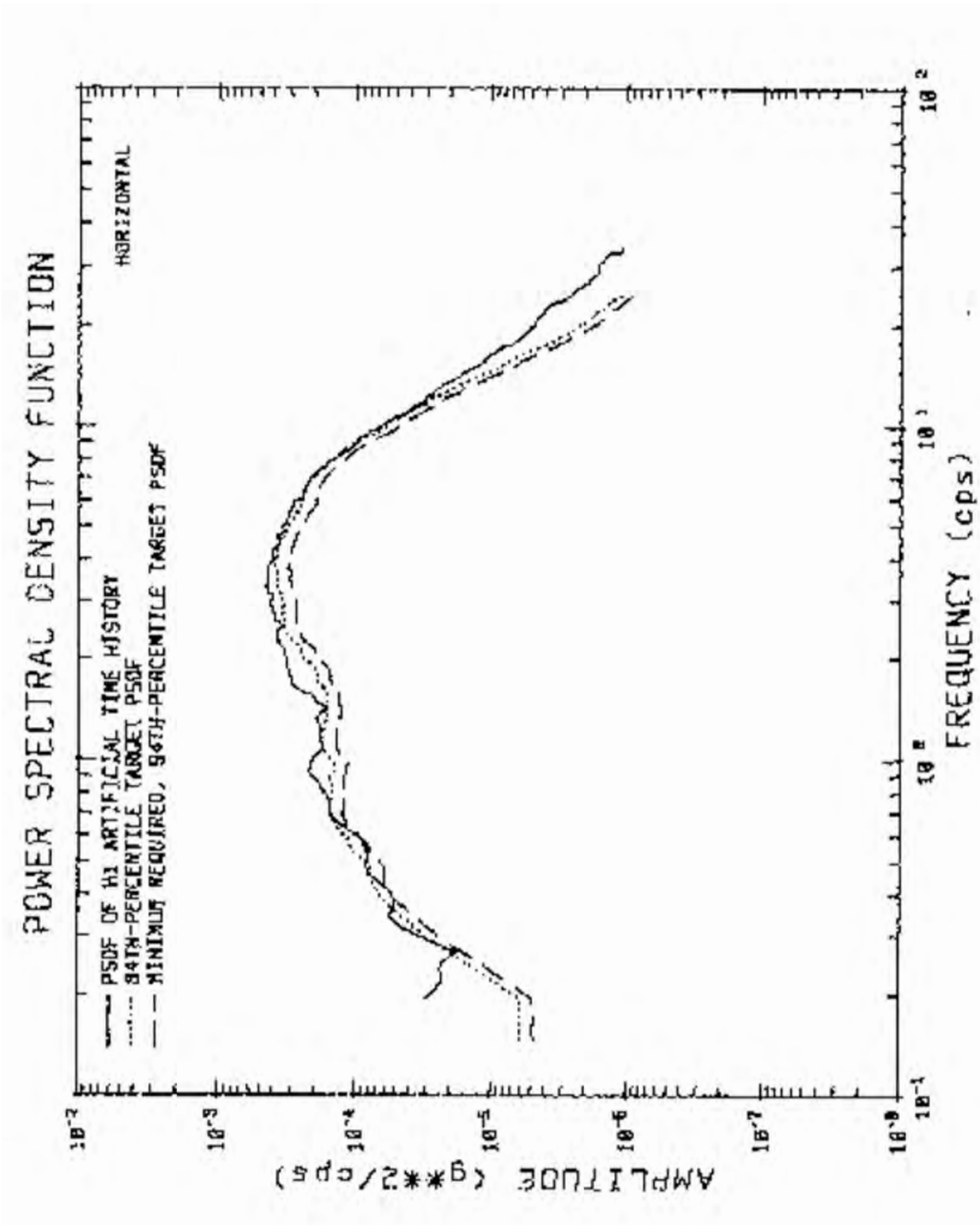
FIGURE 3.7-4q



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
7% DAMPING

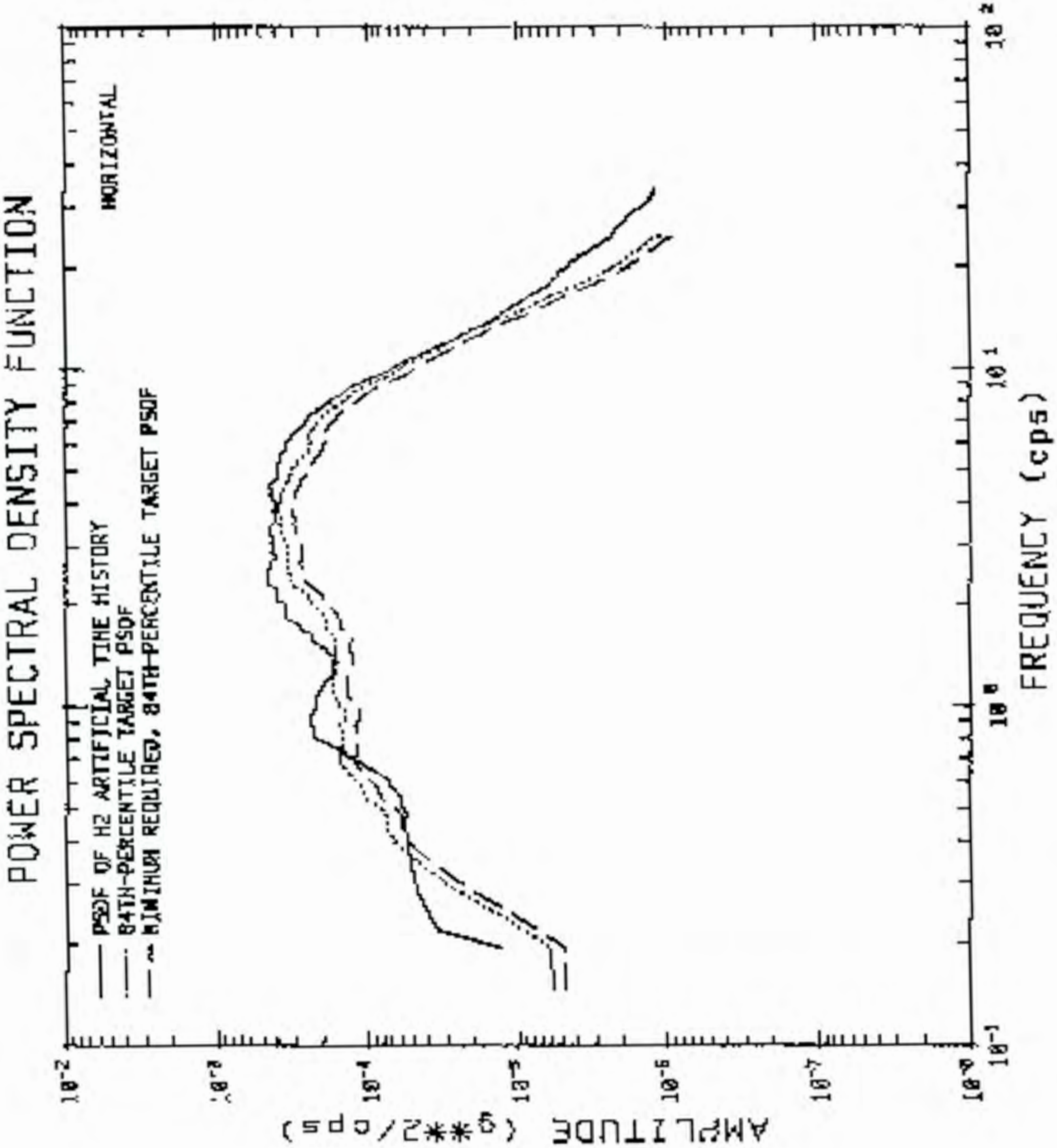
FIGURE 3.7-4r



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPARISONS OF H1 ARTIFICIAL TIME HISTORY PSDF WITH HORIZONTAL, 84TH-PERCENTILE, AND MINIMUM REQUIRED, 84TH-PERCENTILE TARGET PSDFs

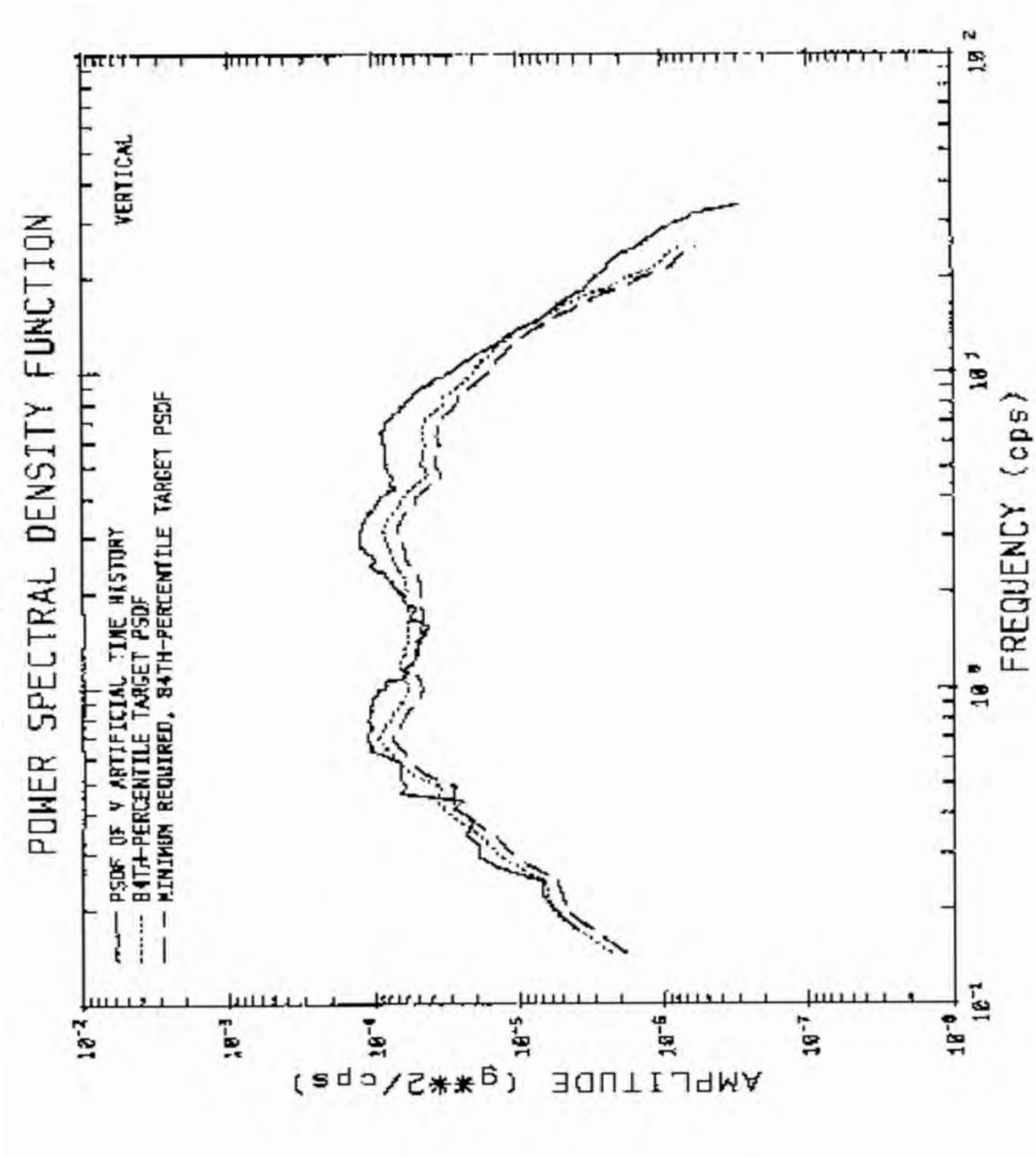
FIGURE 3.7-4s



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPARISONS OF H2 ARTIFICIAL TIME HISTORY PSDF WITH HORIZONTAL, 84TH-PERCENTILE, AND MINIMUM REQUIRED, 84TH-PERCENTILE TARGET PSDFs

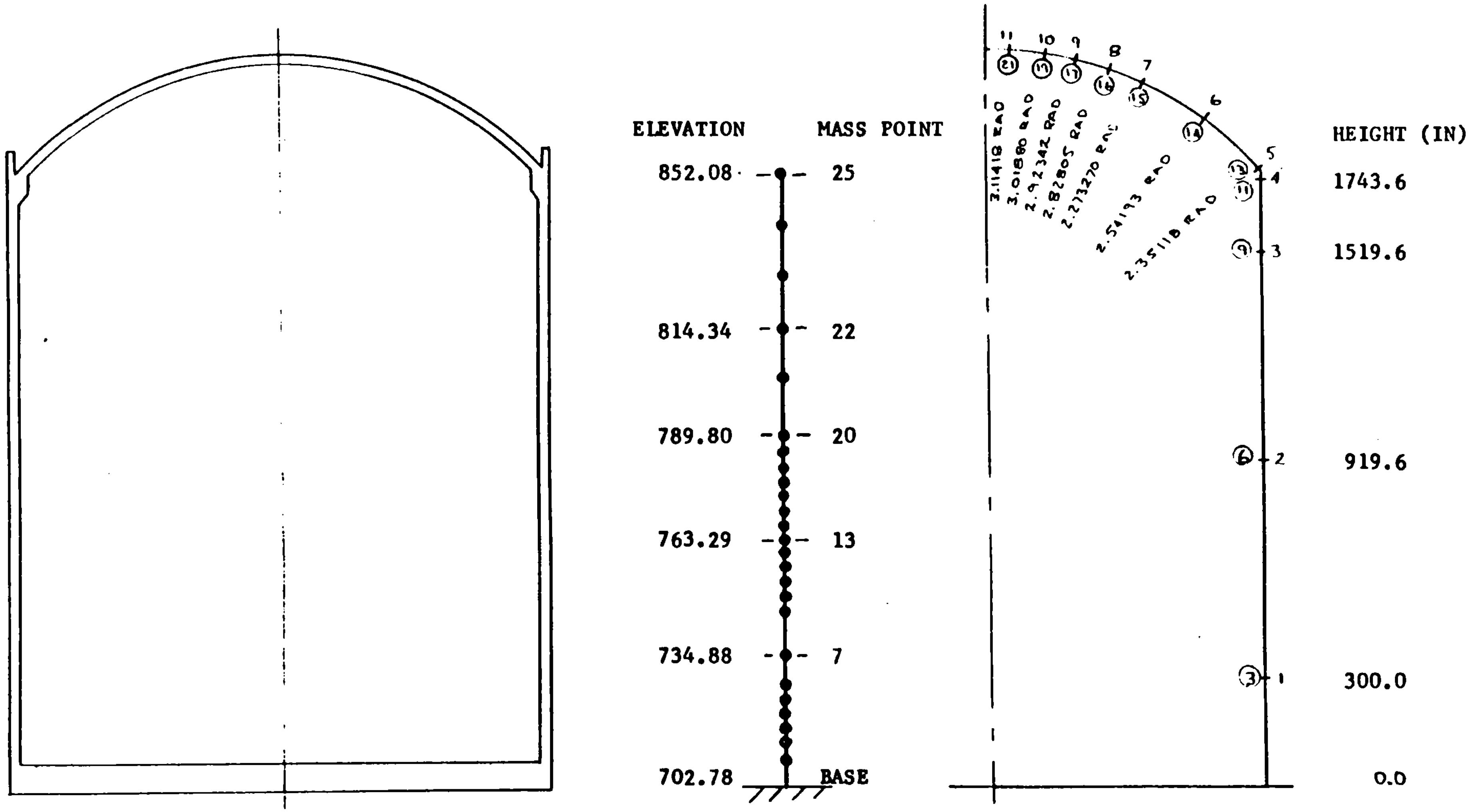
FIGURE 3.7-4t



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPARISONS OF V ARTIFICIAL TIME HISTORY PSDF WITH HORIZONTAL, 84TH-PERCENTILE, AND MINIMUM REQUIRED, 84TH-PERCENTILE TARGET PSDFs

FIGURE 3.7-4u



SECTION THRU
REACTOR SHIELD
BUILDING

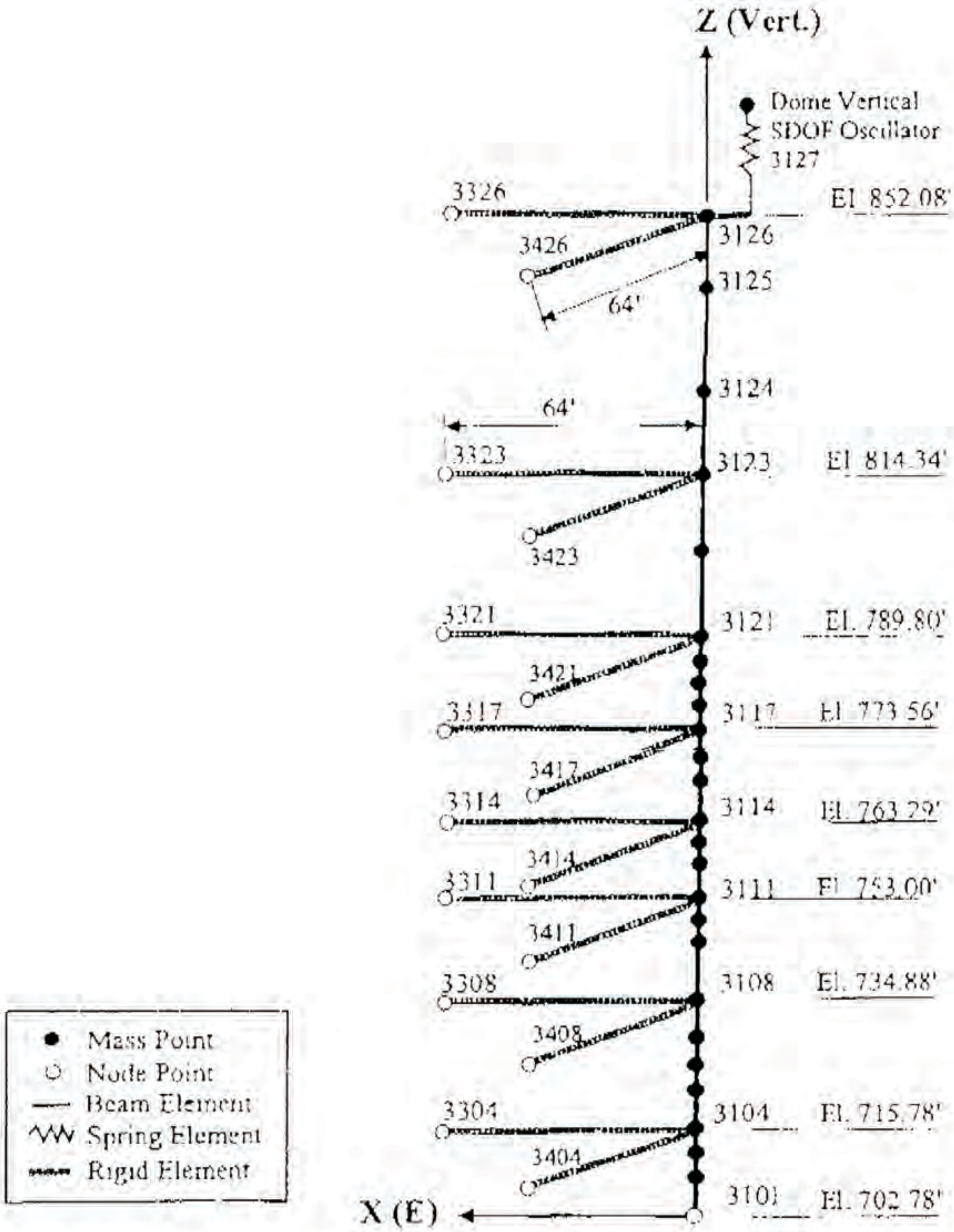
LUMPED MASS MODEL
FOR ANALYSIS OF
CYLINDIRCAL SHELL

LUMPED MASS MODEL
FOR ANALYSIS OF
DOME

× - MASS POINT NOS.
⊗ - NODE NOS. FROM GENSHL-2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

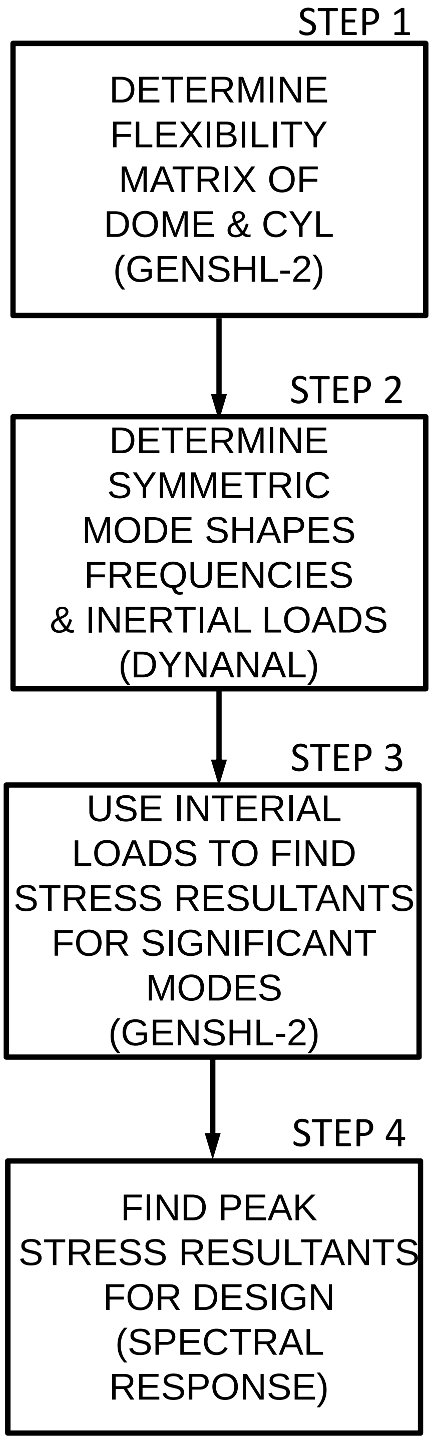
Lumped Mass Model
FIGURE 3.7-5



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Concentric Single-Stick Model Used for Unit-1 Shield Building

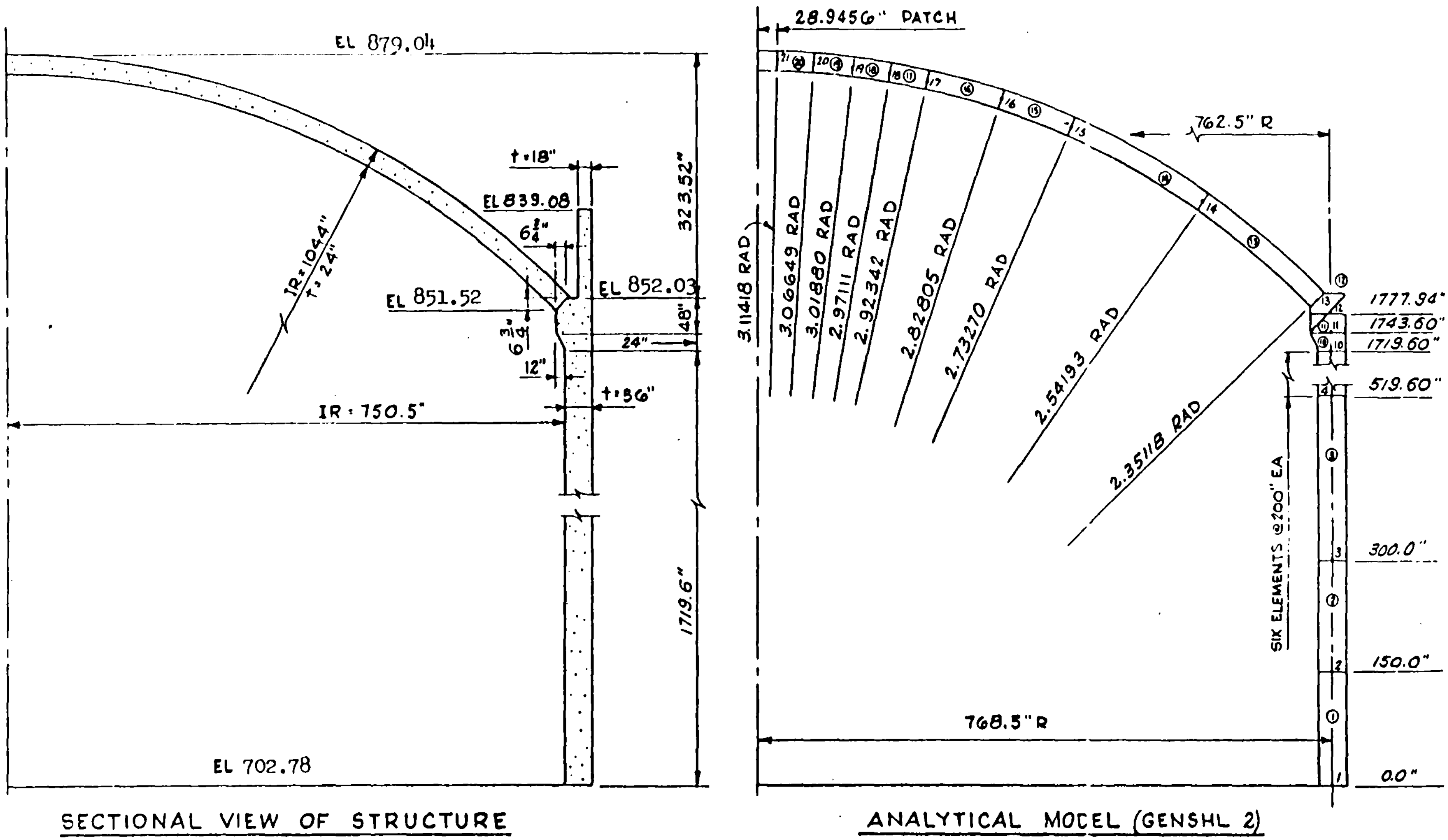
FIGURE 3.7-5a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Flow Chart of Operations
For Response of the Dome

FIGURE 3.7-6



SHELL MODEL FOR DOME ANALYSIS-SHIELD BUILDING

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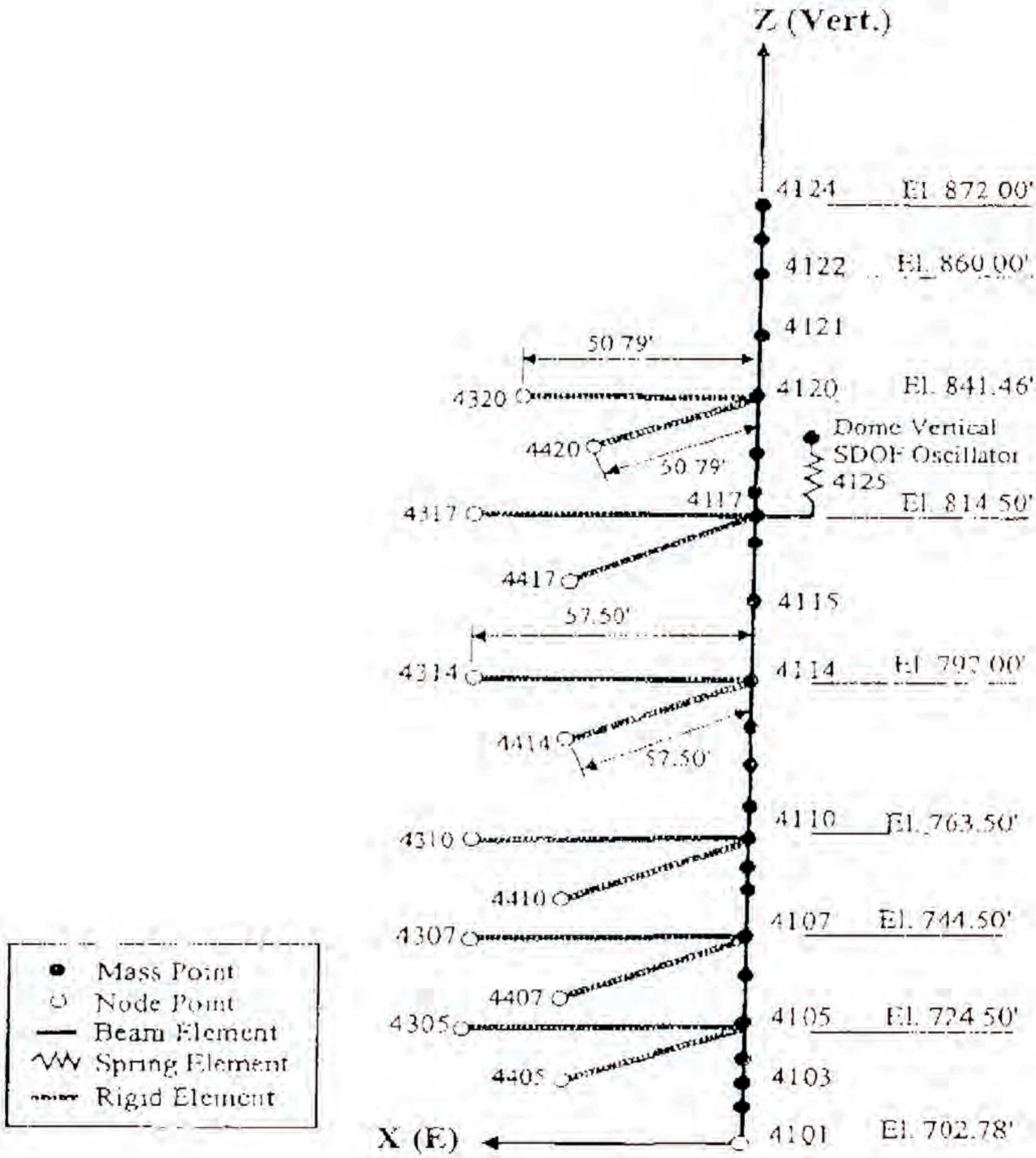
FINAL SAFETY
ANALYSIS REPORT

Shell Model for Dome Analysis

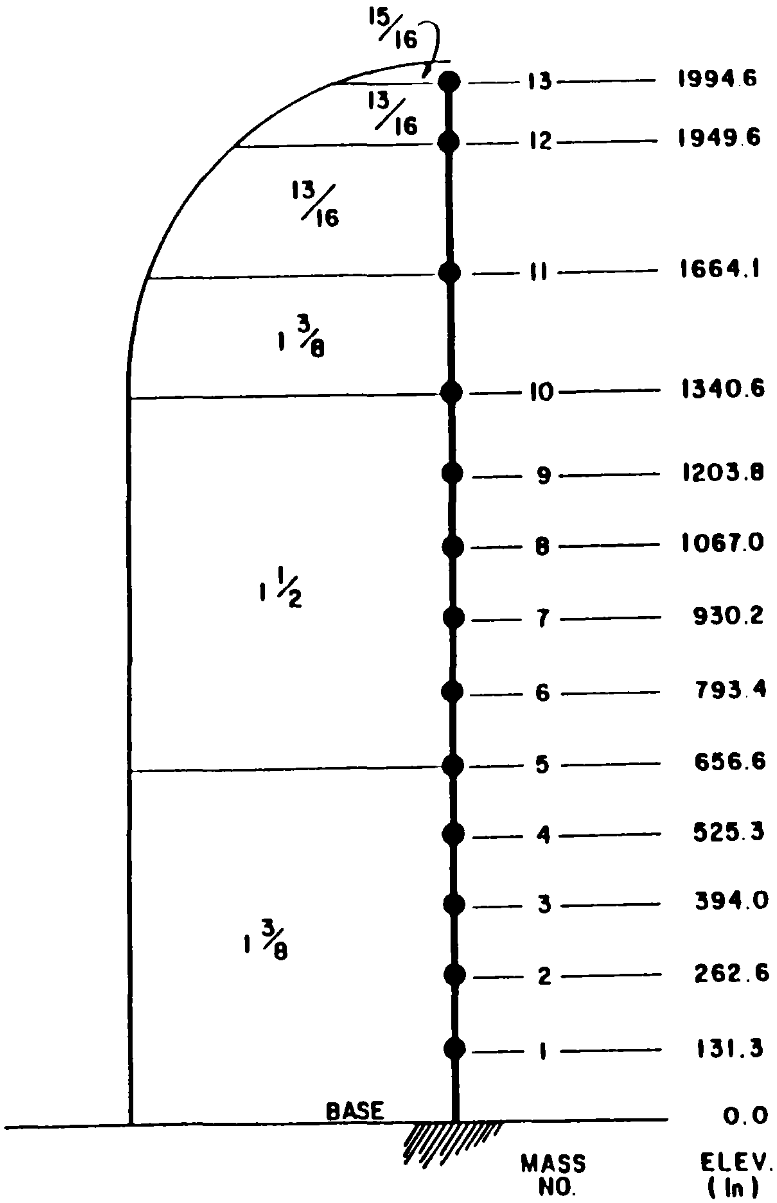
FIGURE 3.7-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Concentric Single-Stick Model Used for Unit-1 Steel Containment Vessel

FIGURE 3.7-7a



MASS NO.	I (In^4)	HORIZ A (In^2)	VERT A (In^2)	ELEV. (In)	WEIGHT (k)
1	$.1423 \times 10^{10}$	2981	5967	131.3	271.9
2	$.1423 \times 10^{10}$	2981	5967	262.6	509.5
3	$.1423 \times 10^{10}$	2981	5967	394.0	352.5
4	$.1423 \times 10^{10}$	2981	5967	525.3	425.7
5	$.1423 \times 10^{10}$	2981	5967	656.3	273.6
6	$.1553 \times 10^{10}$	3251	6510	793.4	501.2
7	$.1553 \times 10^{10}$	3251	6510	930.2	305.9
8	$.1553 \times 10^{10}$	3251	6510	1067.0	410.4
9	$.1553 \times 10^{10}$	3251	6510	1203.8	372.4
10	$.1423 \times 10^{10}$	2981	5967	1340.6	315.0
11	$.0578 \times 10^{10}$	1555	3111	1664.1	639.0
12	$.871 \times 10^{8}$	828	1656	1949.6	336.3
13	$.272 \times 10^{8}$	561	1123	1994.6	98.2

WATTS BAR NUCLEAR PLANT
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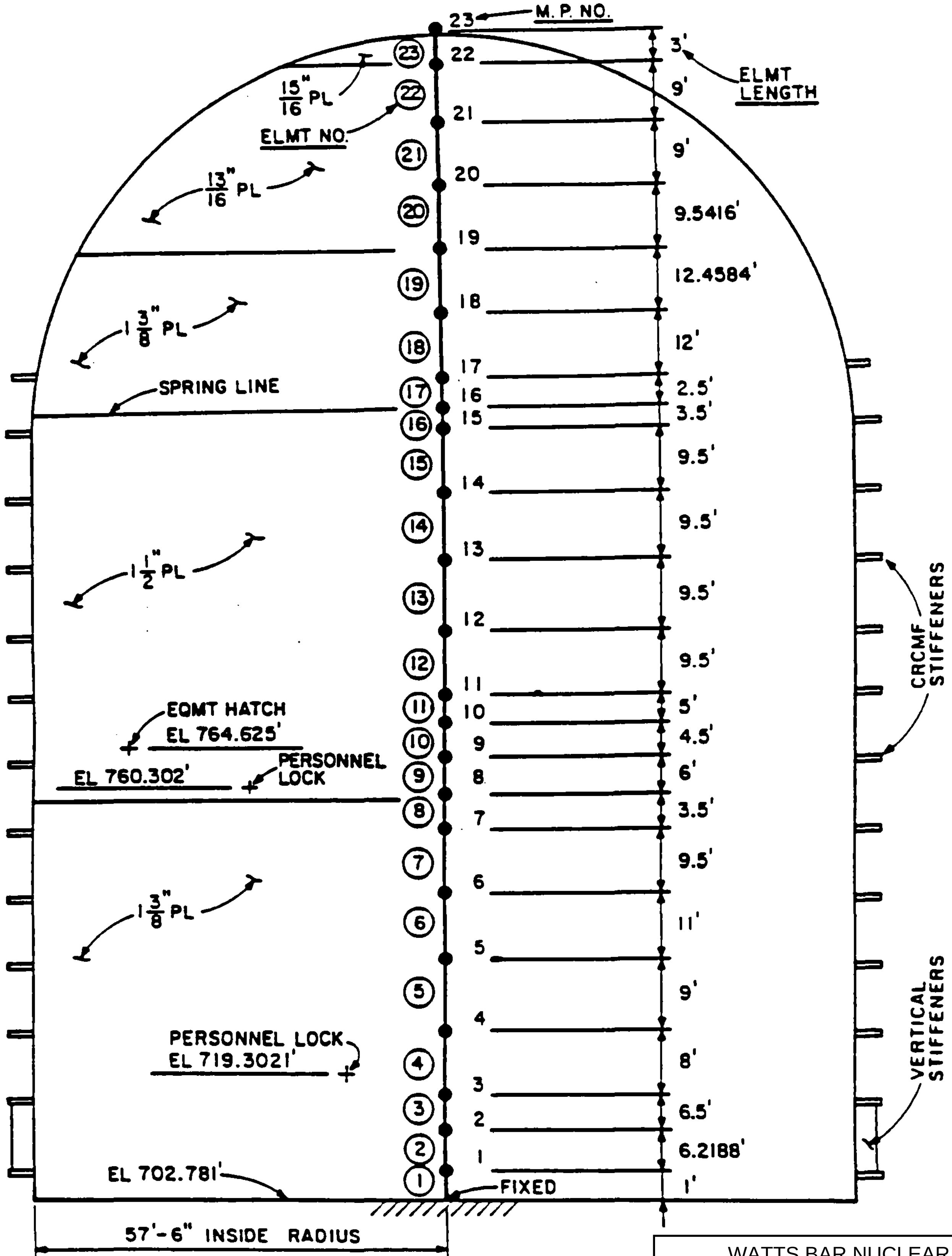
ANALYSIS REPORT

Containment Vessel Lumped Mass
Beam Model and Properties

FIGURE 3.7-7B

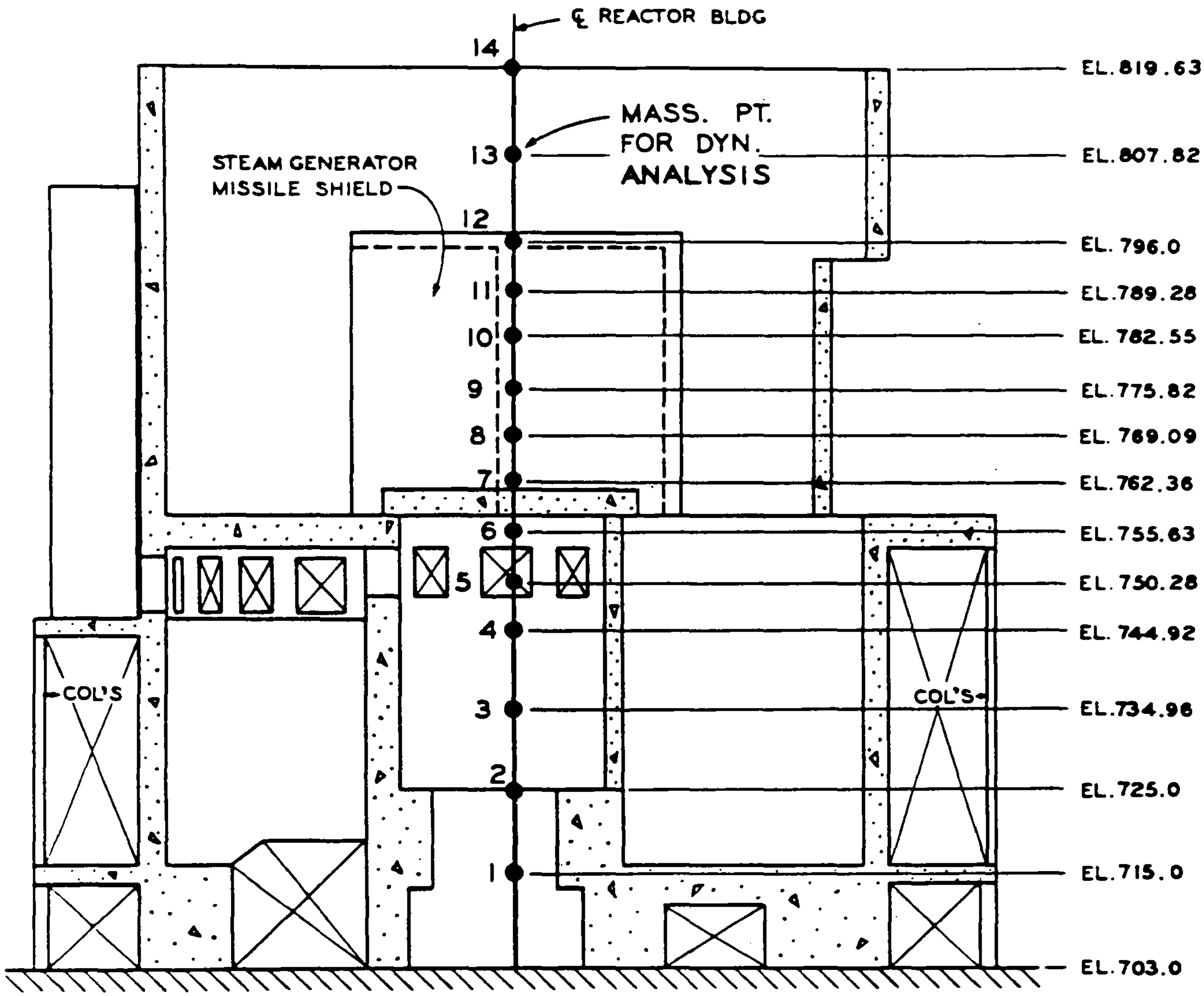


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of Steel
Containment Vessel
and Lumped Mass Model
for Seismic Analysis

FIGURE 3.7-7C

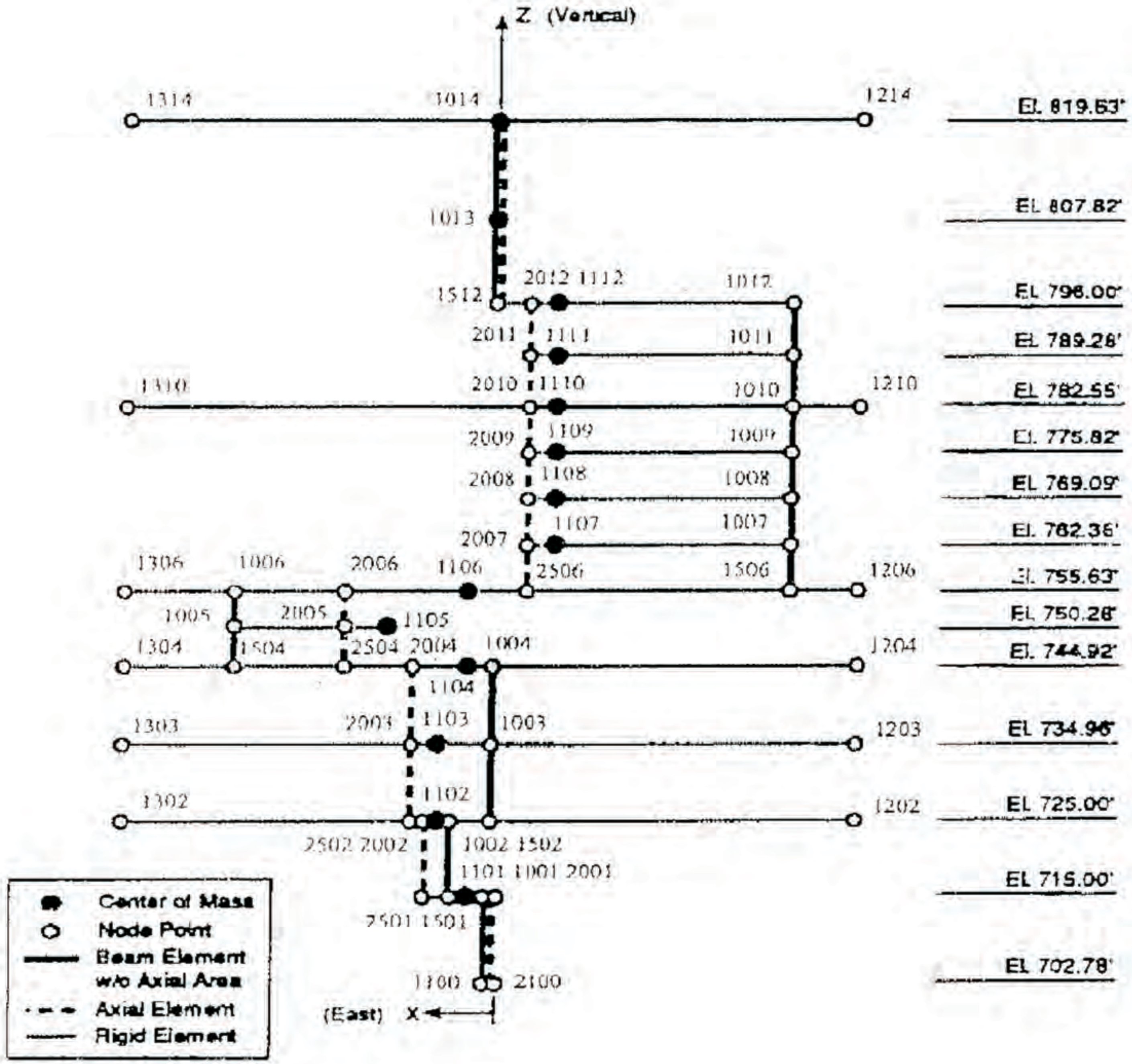
TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
REACTOR BUILDING - INTERIOR CONCRETE STRUCTURE



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation
Looking North
Lumped Mass Model
for Dynamic Analysis

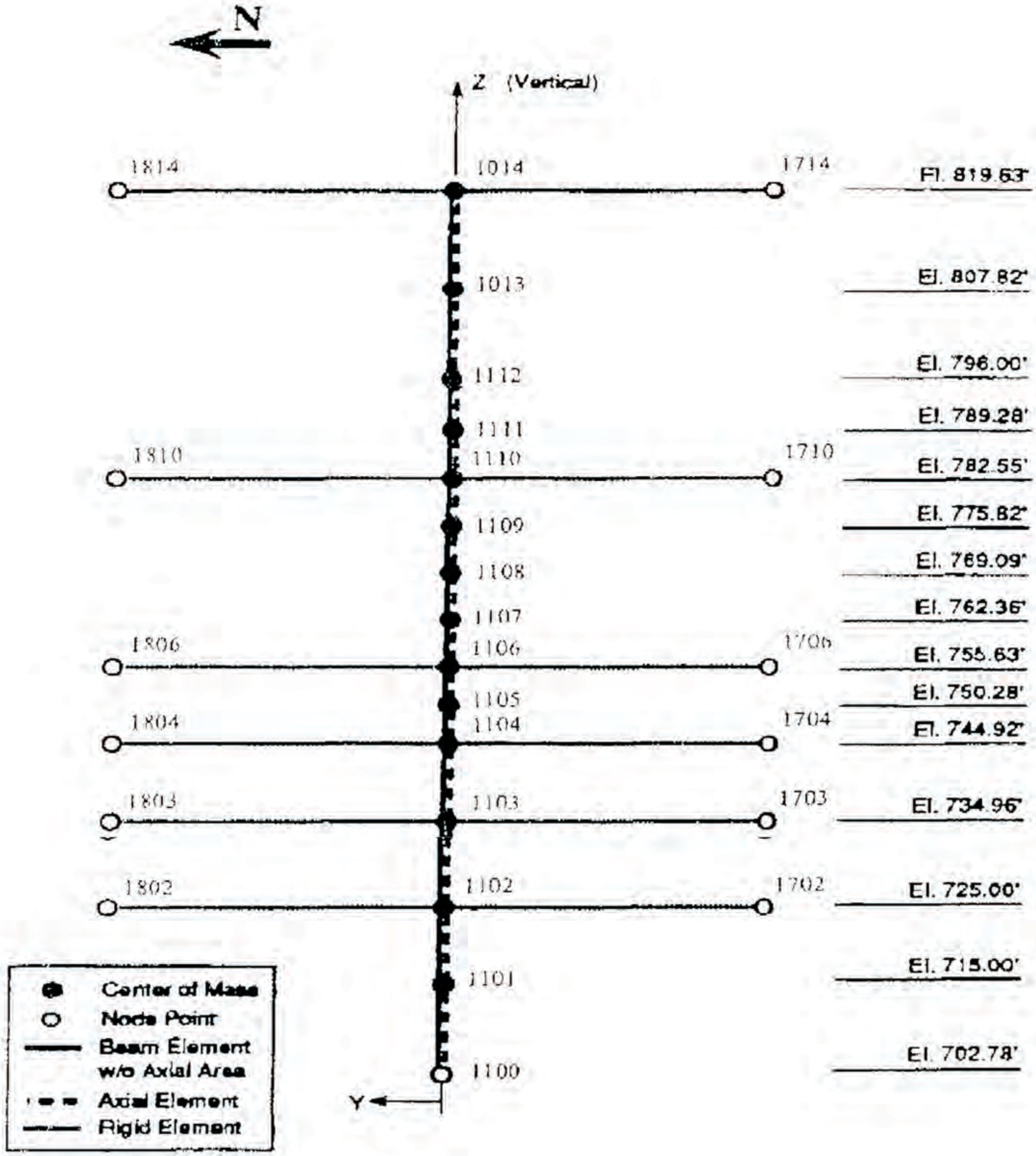
FIGURE 3.7-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Three-dimensional Multiple-Stick Seismic Model Used for Unit-1 Interior Concrete Structures (ICS) -XZ (EW-Vertical) Plane

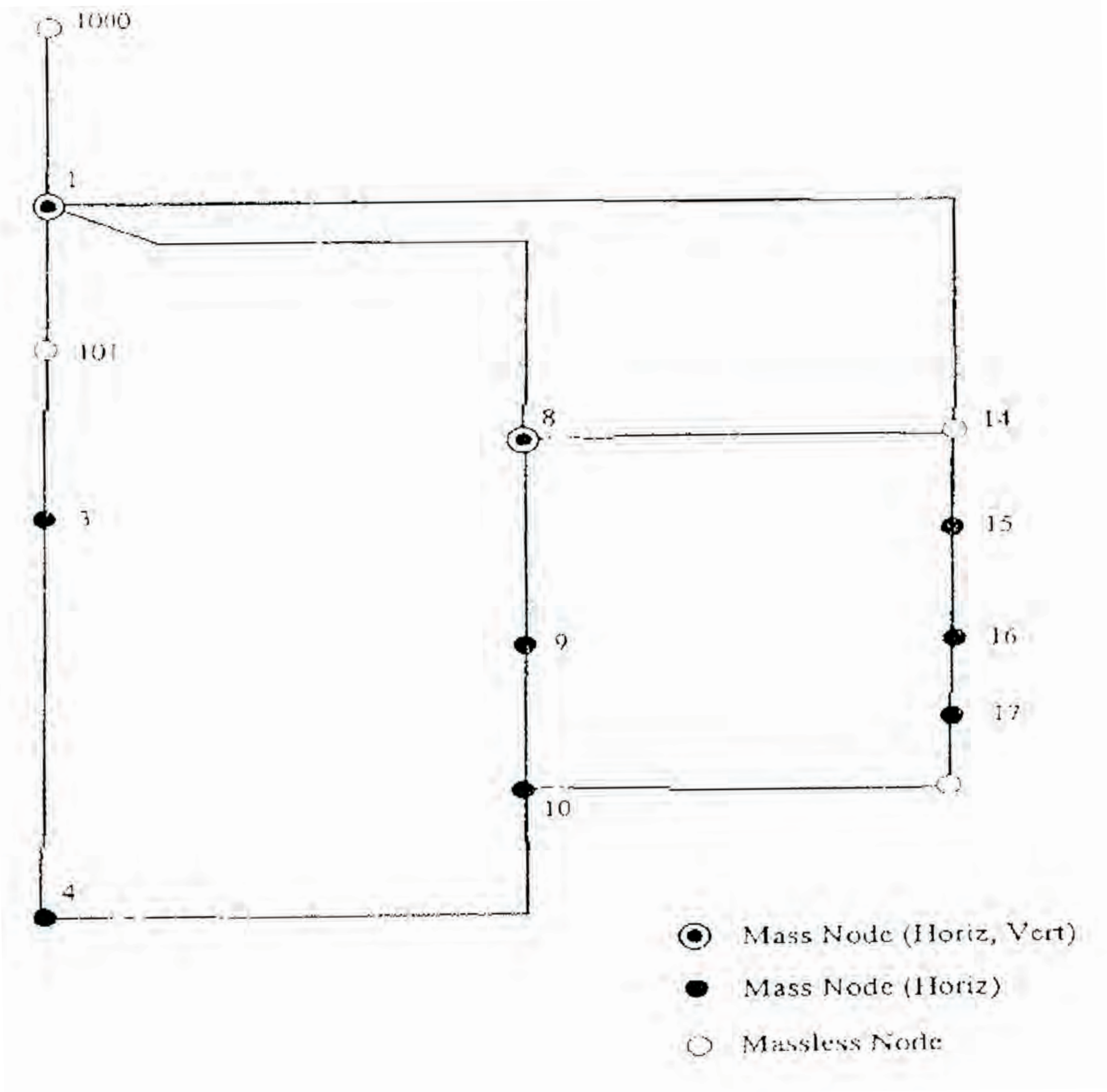
FIGURE 3.7-8a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Three-dimensional Multiple-Stick Seismic Model Used for Unit-1 Interior Concrete Structures (ICS) -YZ (NS-Vertical) Plane

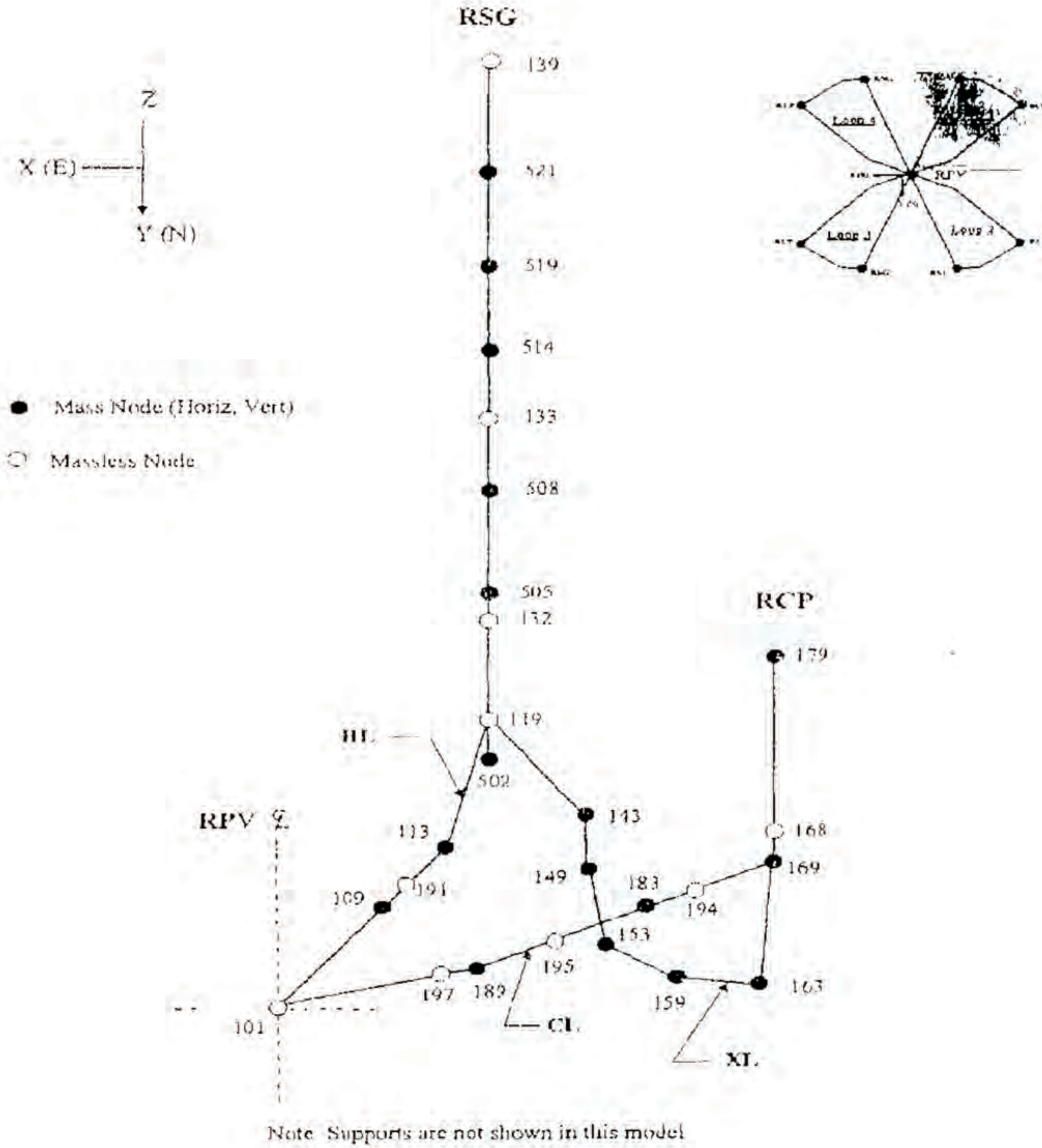
FIGURE 3.7-8b



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Configuration of the WBN Unit-1 Reactor Pressure Vessel (RPV) Seismic Model

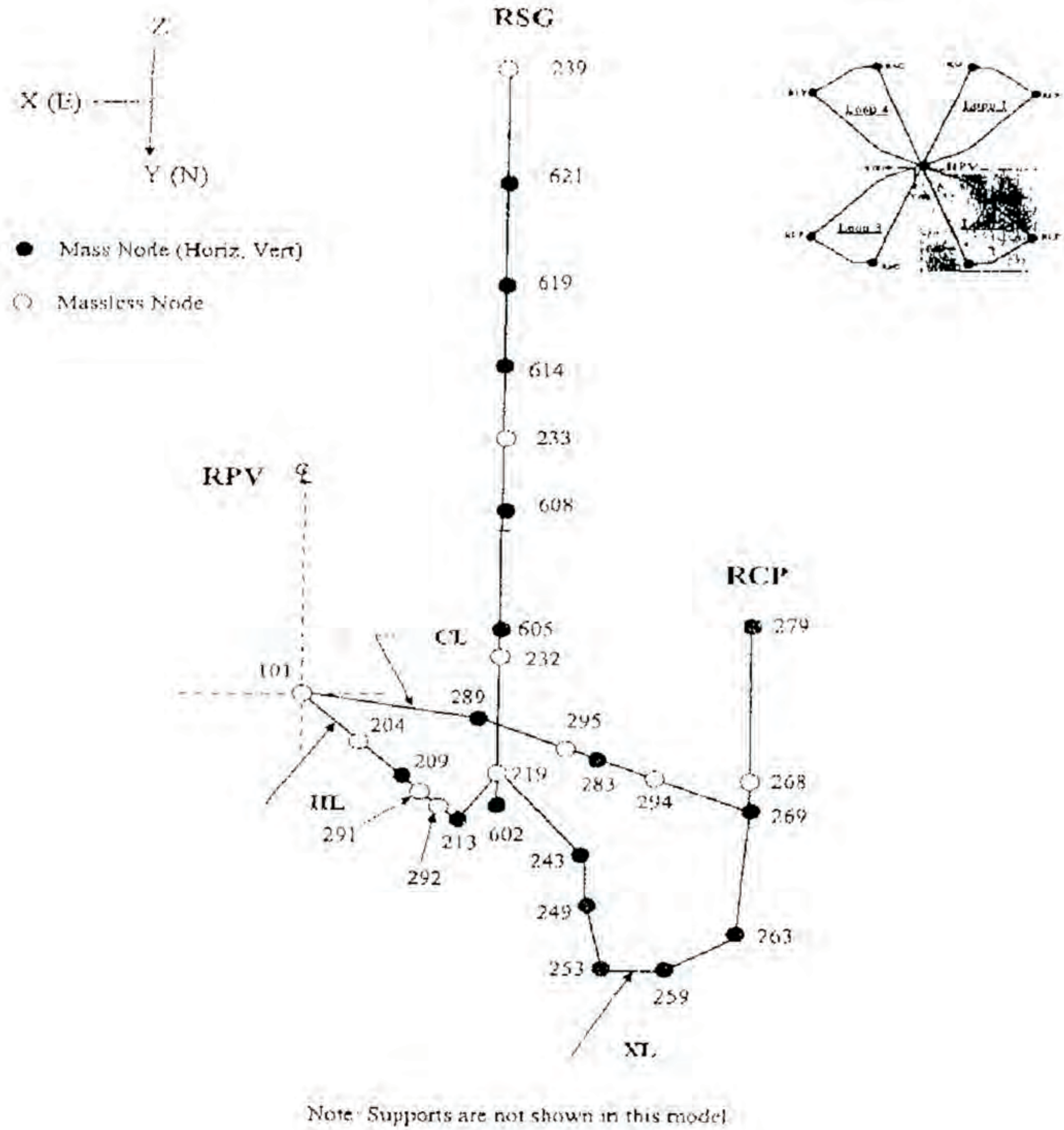
FIGURE 3.7-8c



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 1

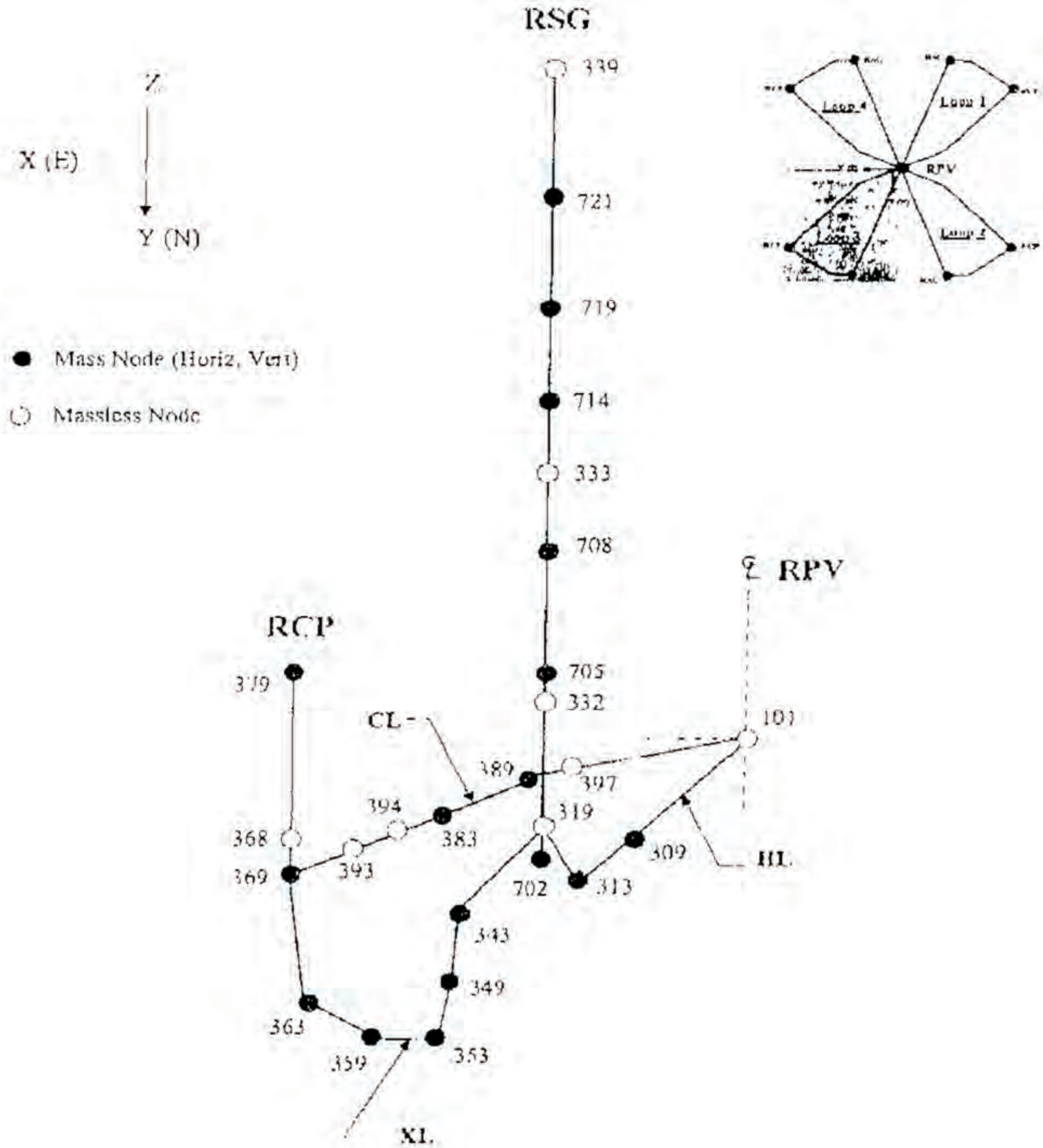
FIGURE 3.7-8d



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 2

FIGURE 3.7-8e

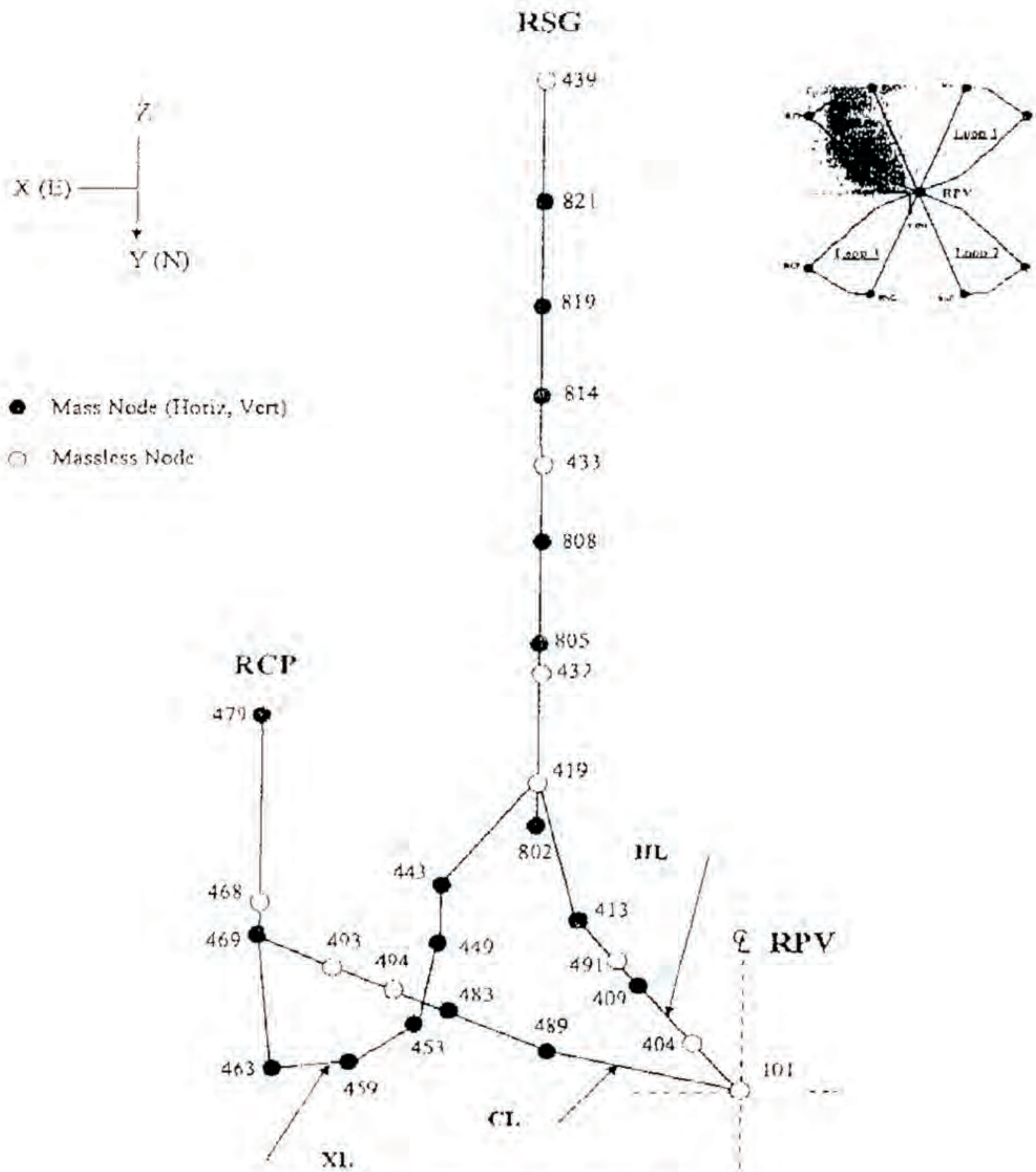


Note: Supports are not shown in this model

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 3

FIGURE 3.7-8f

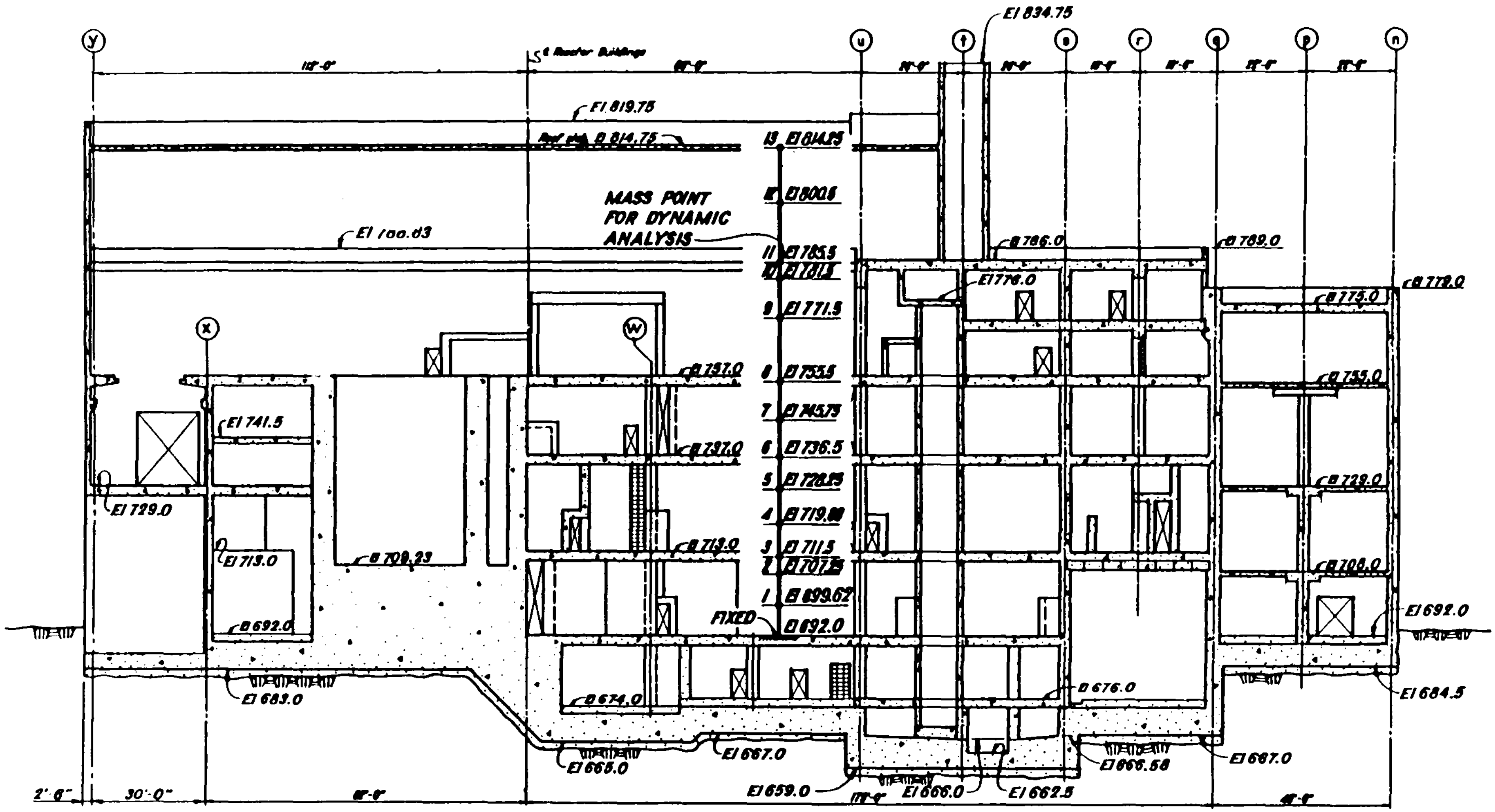


Note: Supports are not shown in this model.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 4

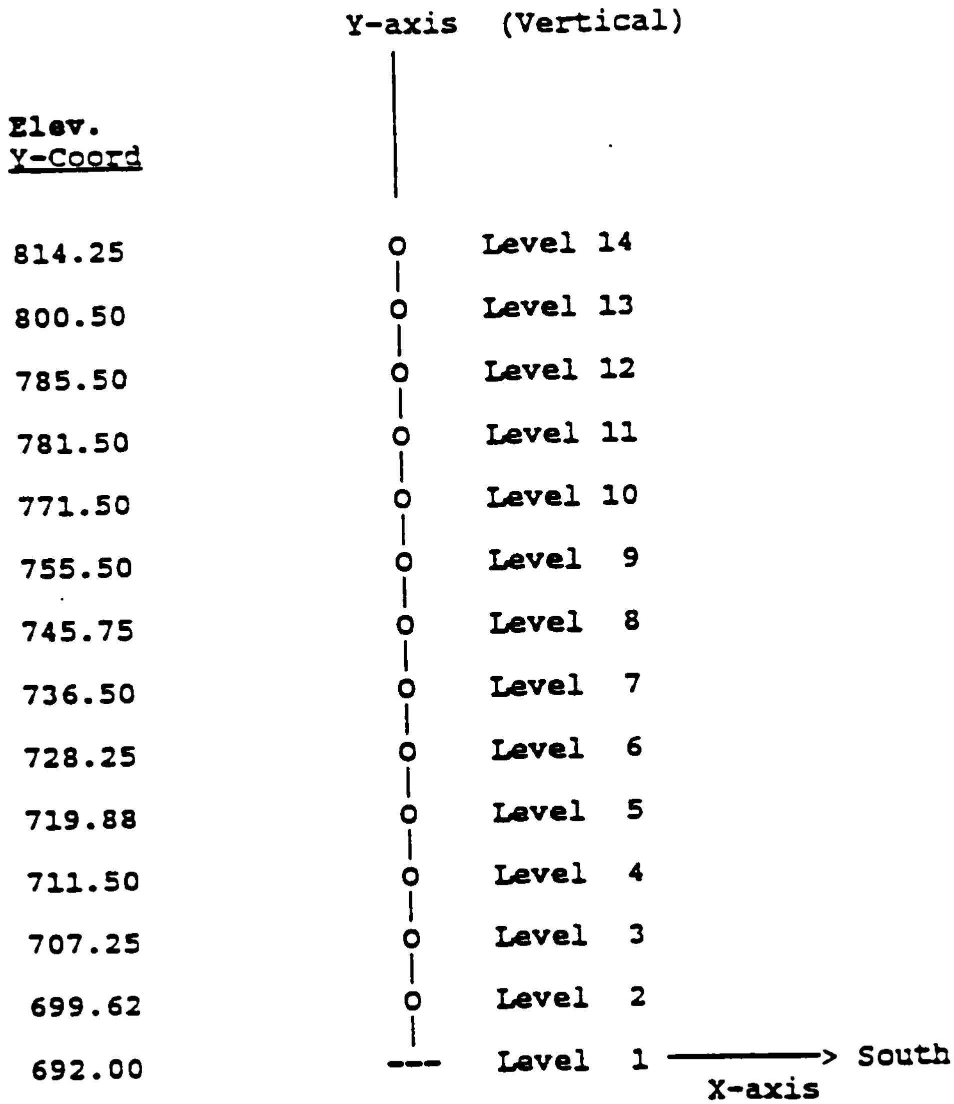
FIGURE 3.7-8g



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of Auxiliary-Control Building - Lumped Mass Model for Dynamic Analysis

FIGURE 3.7-9

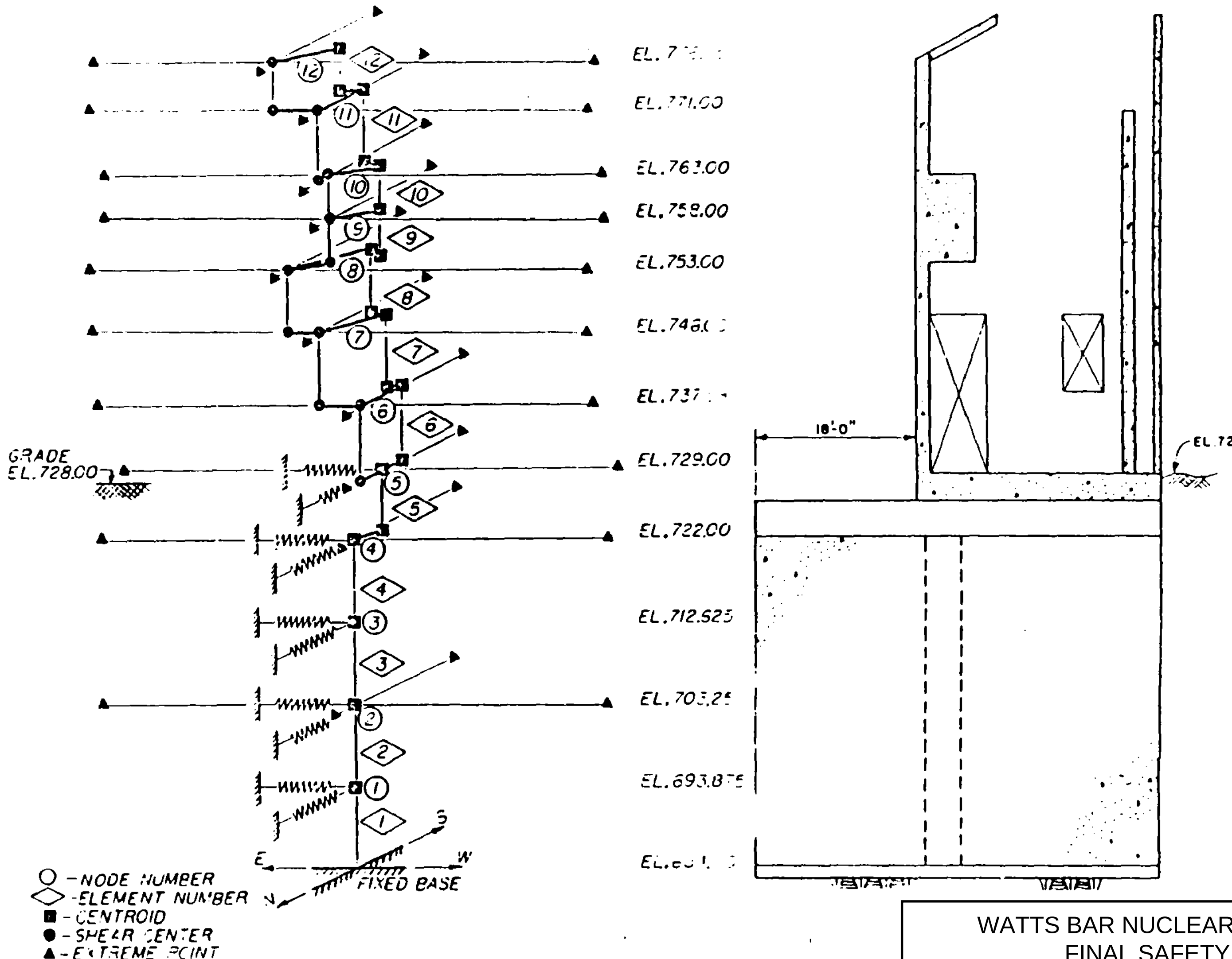


The coordinates of the Set B and Set C centers of mass and centers of rigidity are provided in Table 3.7-9A.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

ACB Seismic Model
(Set B and Set C)

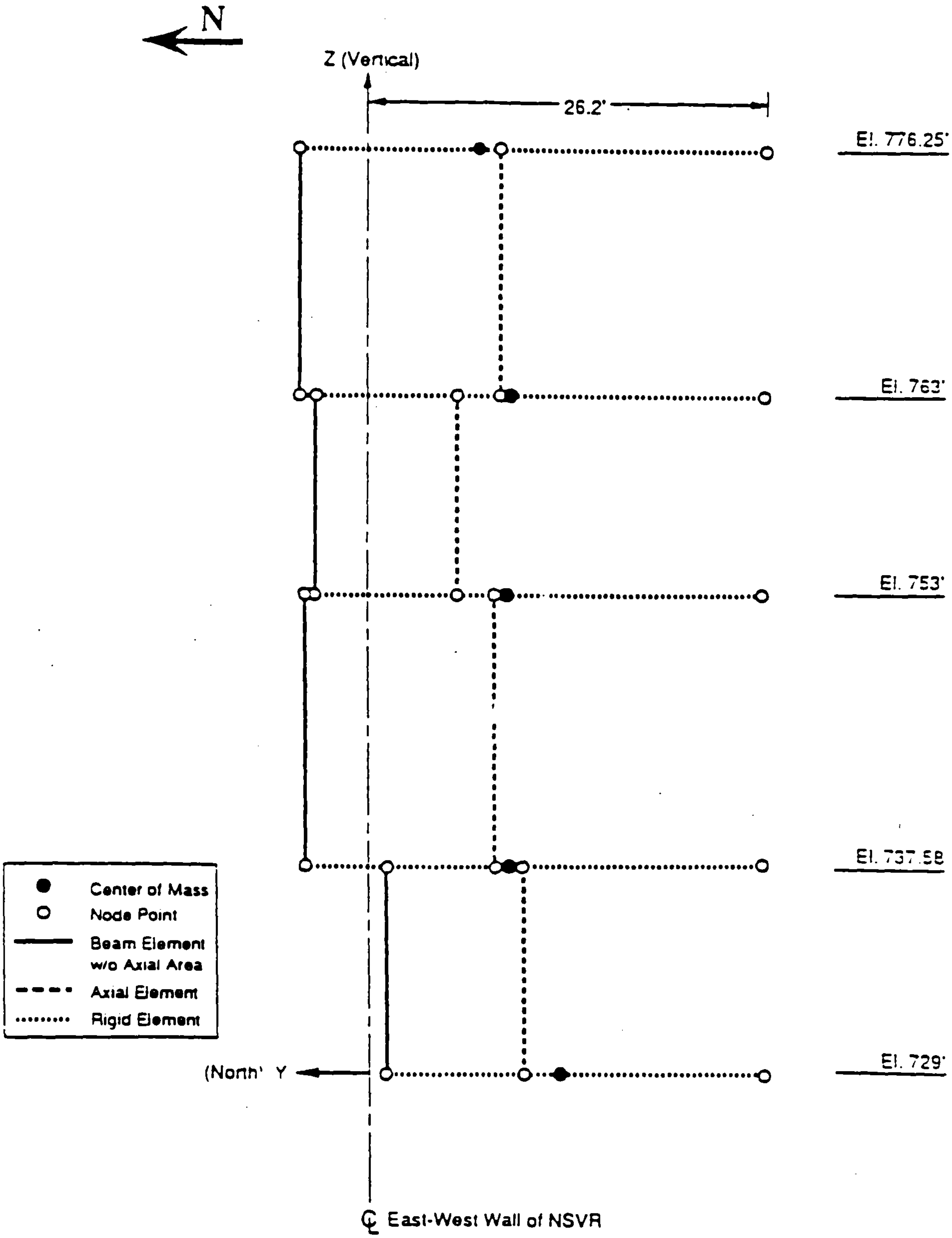
FIGURE 3.7-9A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of North Steam
Valve Room and Lumped Mass Model
for Seismic Analysis

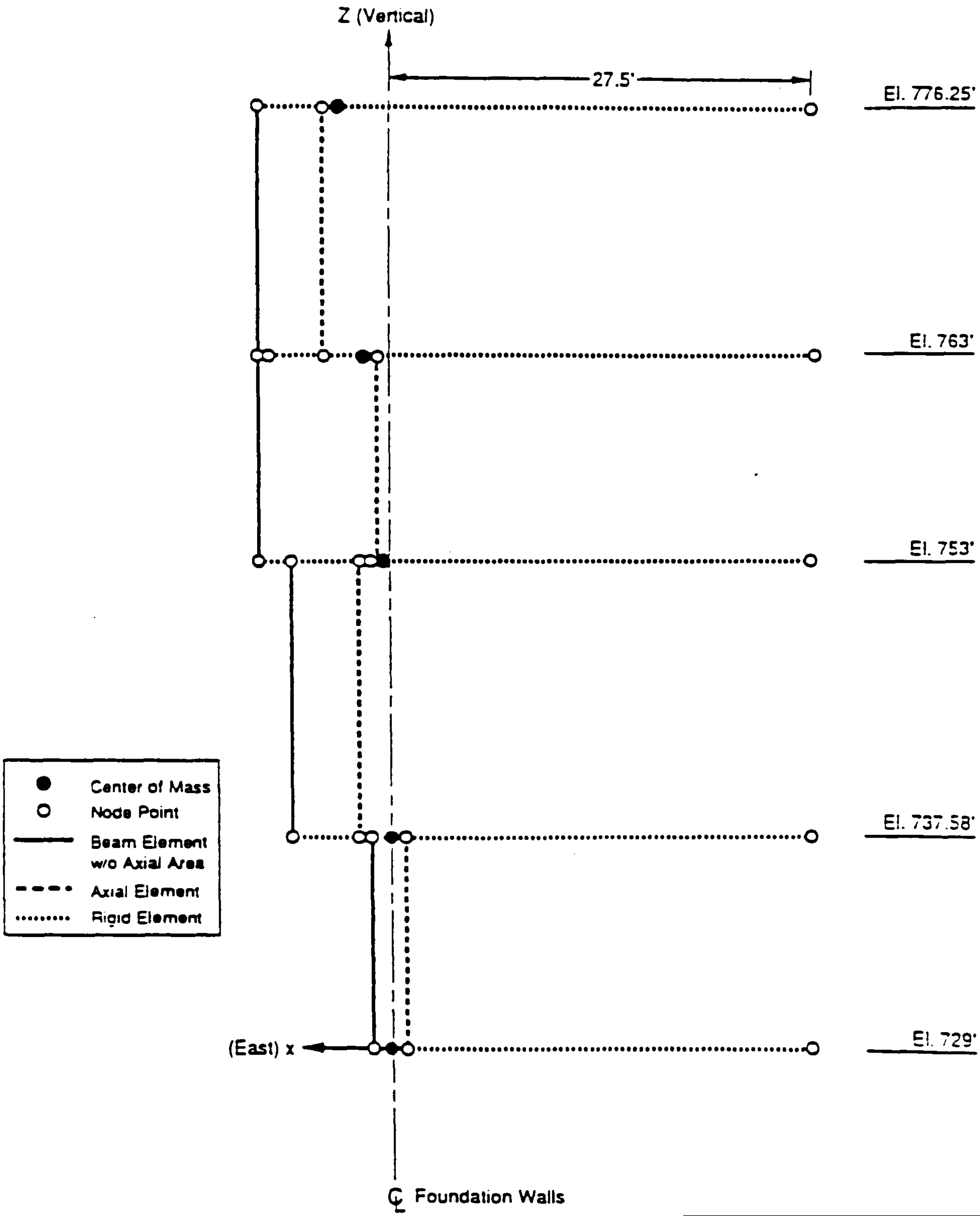
FIGURE 3.7-10



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lumped-Mass Stick Model for the NSVR Superstructure - XZ Plane

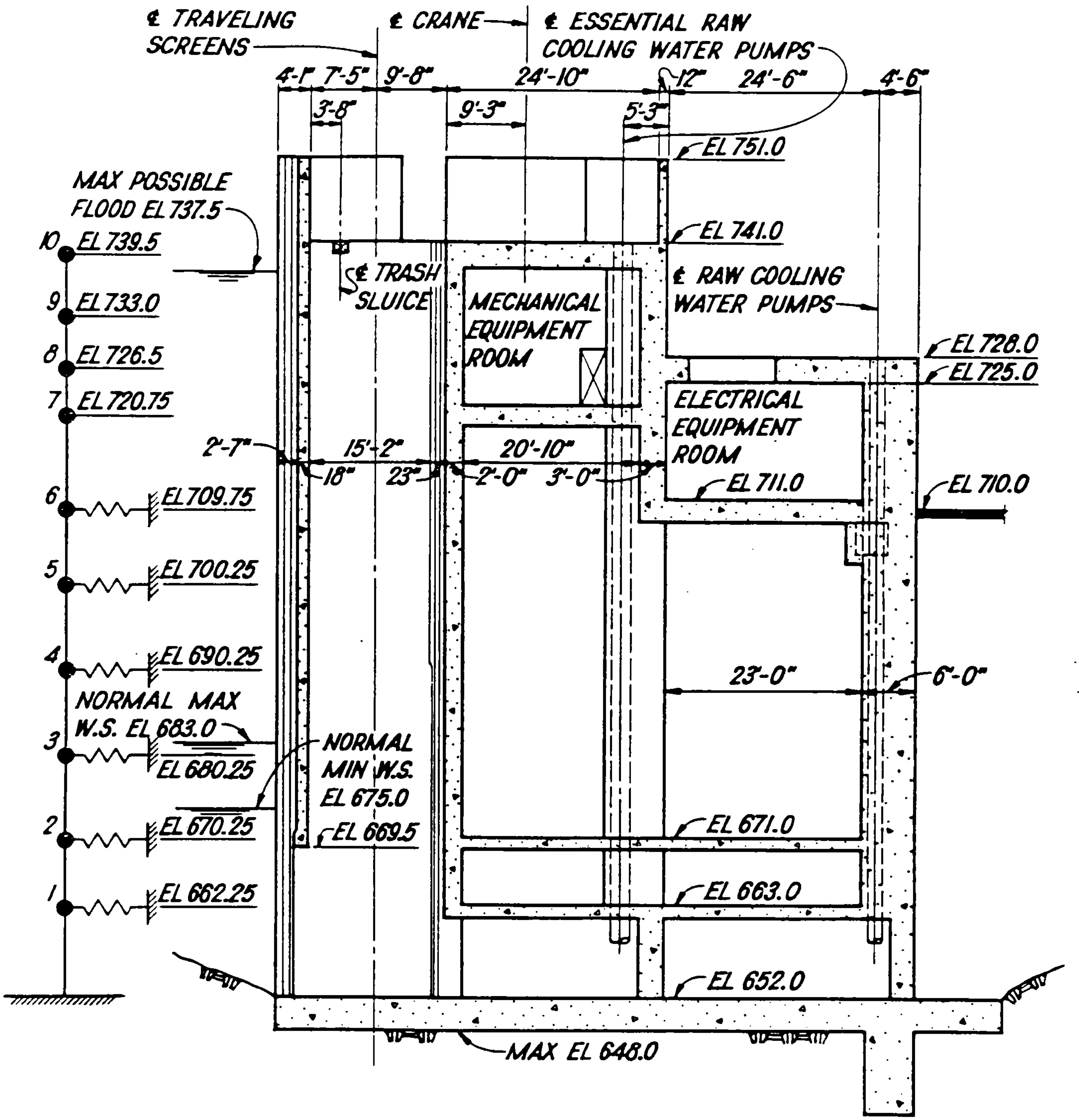
FIGURE 3.7-10A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lumped-Mass Stick Model for the NSVR Superstructure - XZ Plane

FIGURE 3.7-10B



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of
Intake Pumping Station
Lumped Mass Model for
Dynamic Analysis

FIGURE 3.7-11

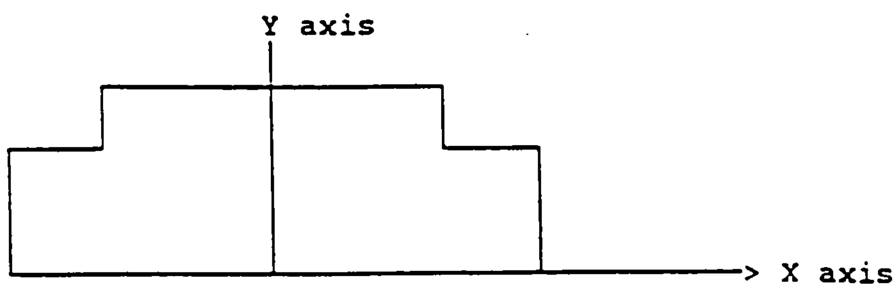
Elev.
Z-Coord

Z-axis (Vertical)

754.00
739.50
733.00
726.50
720.75
709.75
700.25
690.25
680.25
670.25
662.25
652.00
648.00 <== Fixed Base

The coordinates of centers of mass and centers of rigidity are presented in Table 3.7-16B.

Coordinate System:

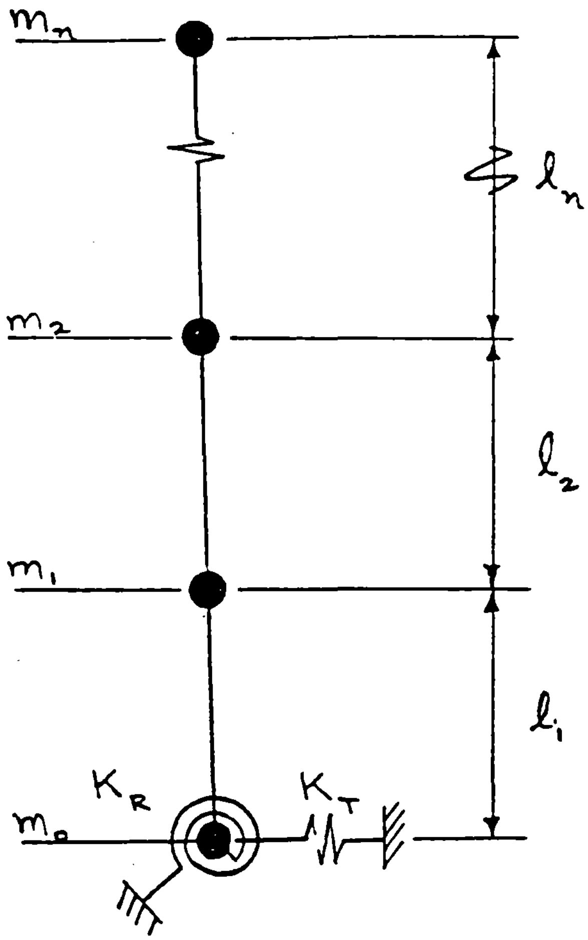


X Axis - Longitudinal direction
Y axis - Transverse direction
Z axis - Vertical direction

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IPS Seismic Model

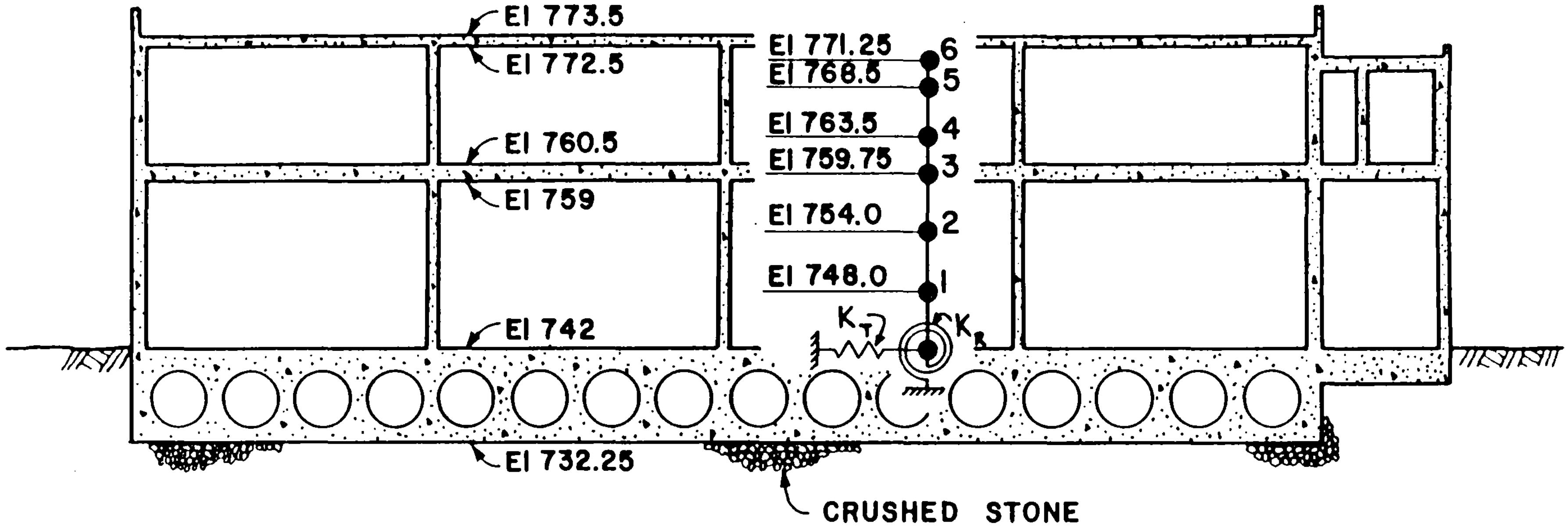
FIGURE 3.7-11A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Mathematical Model
for Soil Structure Interaction

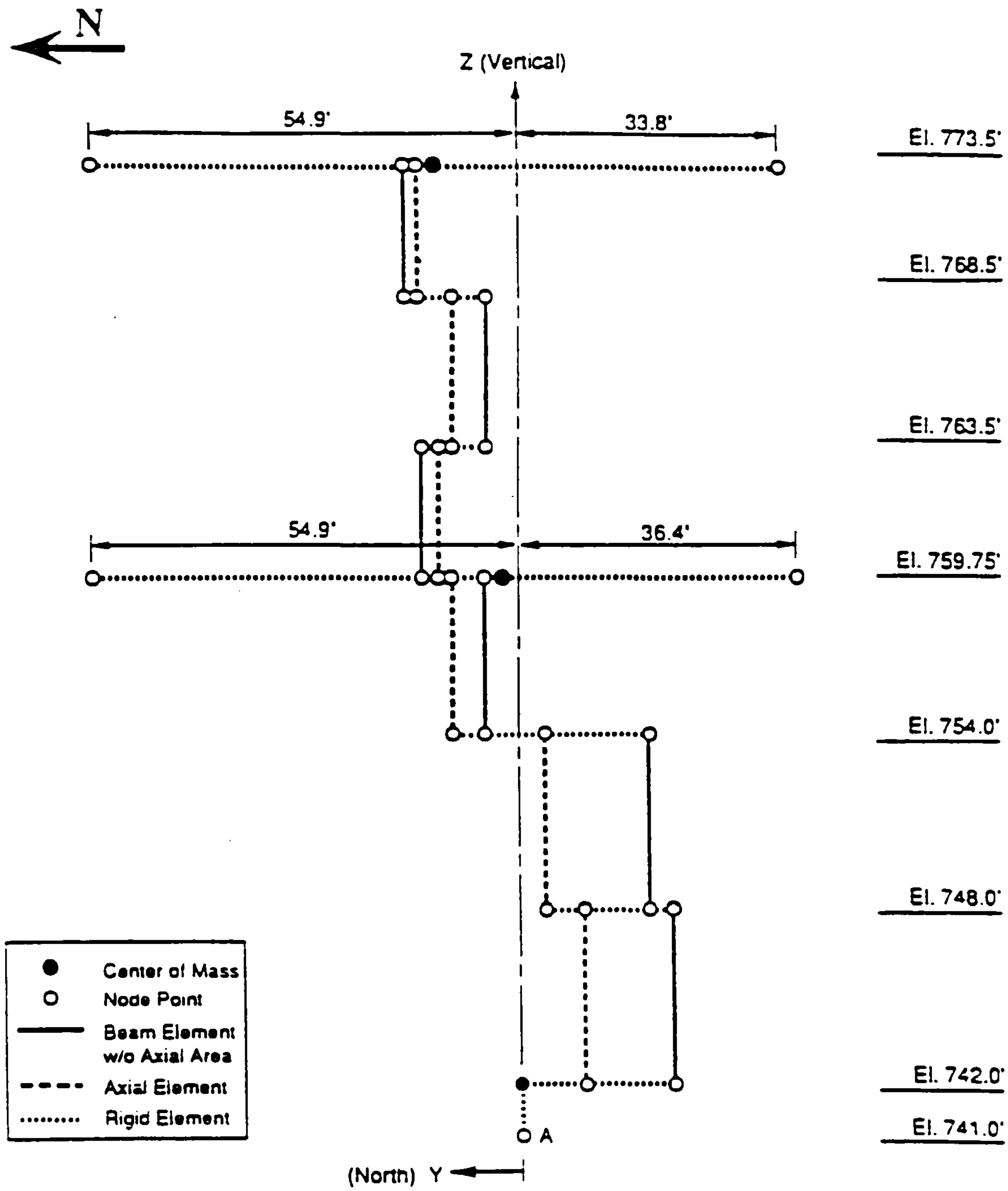
FIGURE 3.7-12



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

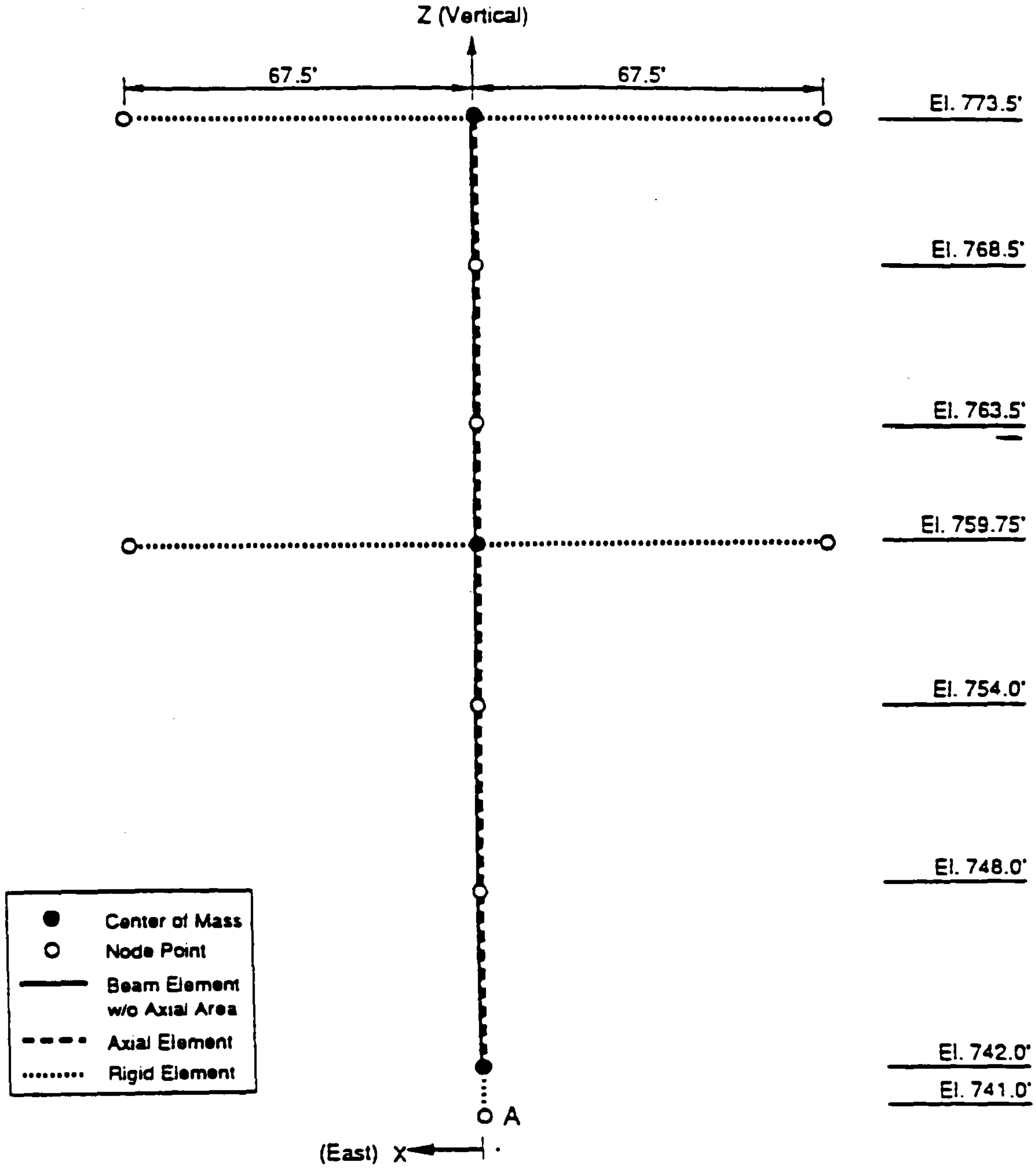
Sectional Elevation of
Diesel Generator Building
Lumped Mass Model for Dynamic Analysis

FIGURE 3.7-13



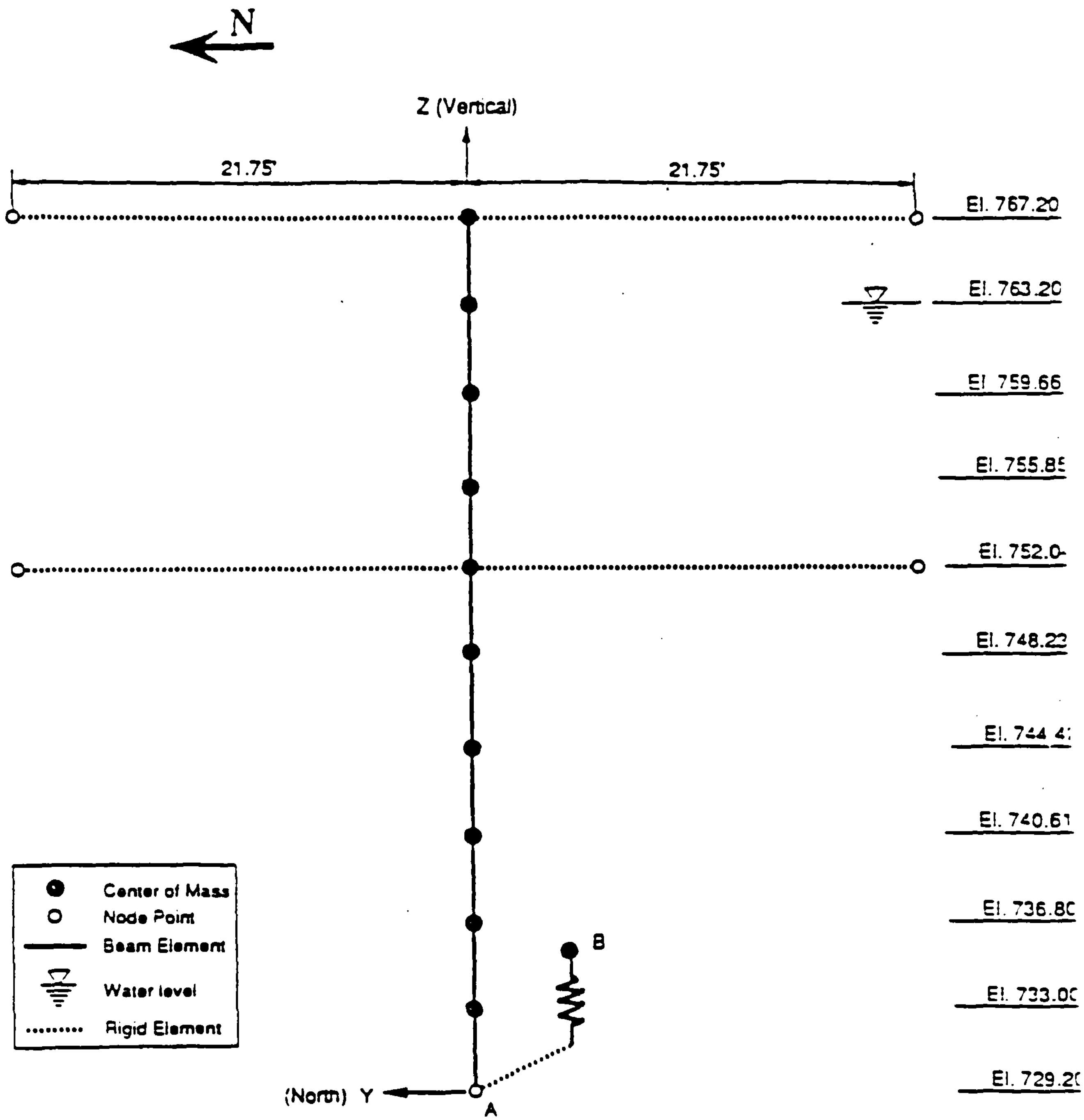
A: Geometric Center of Basemat

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Seismic Analysis Model for Diesel Generator Building XZ Plane FIGURE 3.7-13A



A: Geometric Center of Basemat

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Seismic Analysis Model for Diesel Generator Building XZ Plane FIGURE 3.7-13B

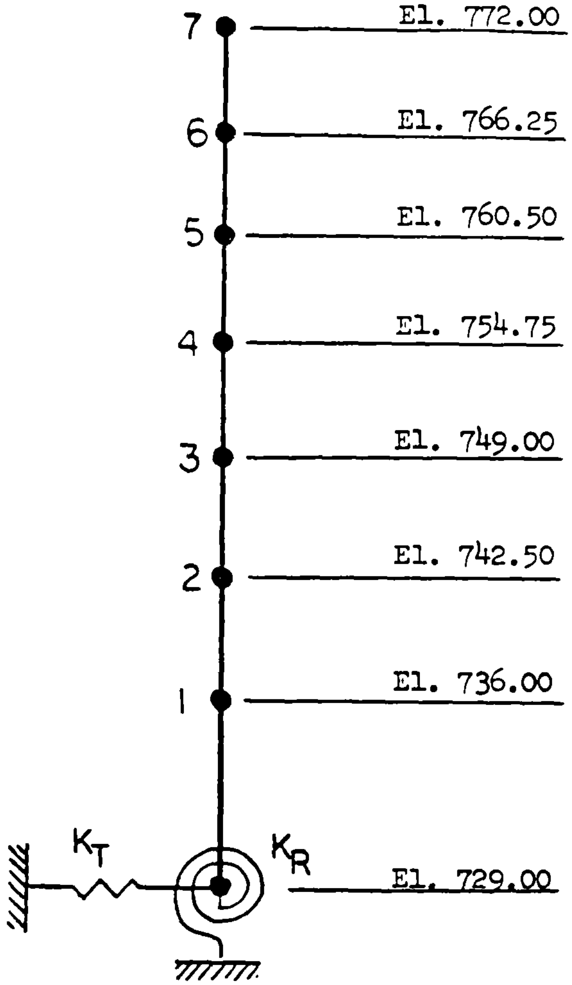


A: Geometric Center of Basemat
B: Hydrodynamic Vertical Oscillator

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lumped-Mass Stick Model
for Refueling Water Storage Tank

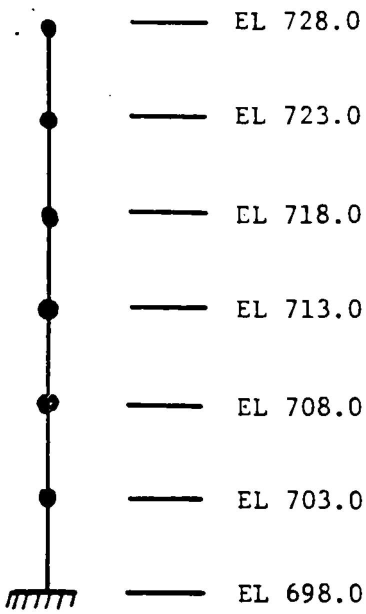
FIGURE 3.7-13C



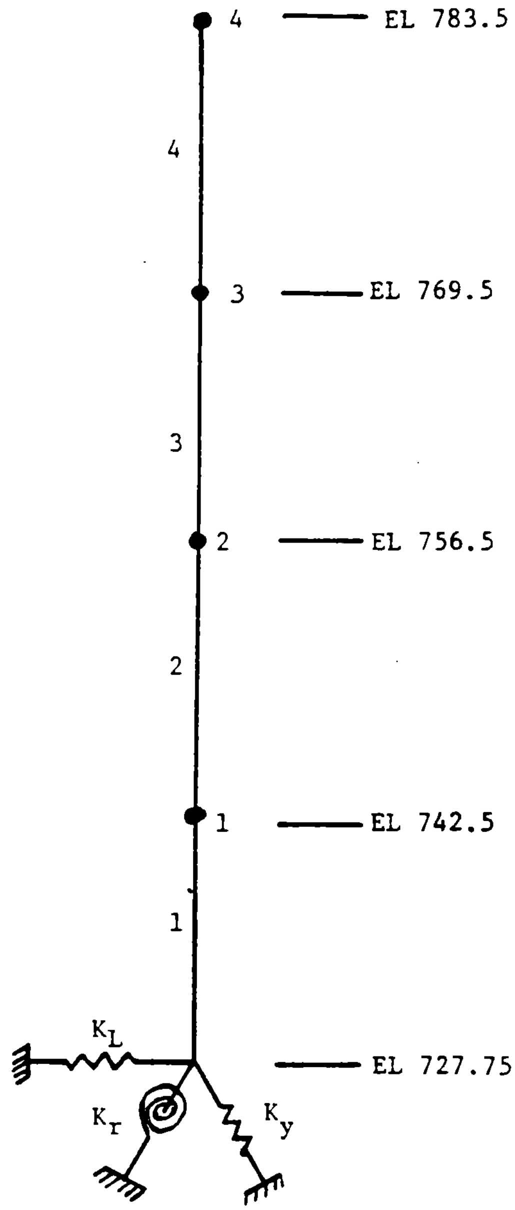
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Mathematical Model
for Dynamic Analysis of the
Waste Packaging Area

FIGURE 3.7-14

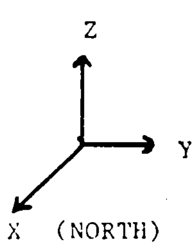


SOIL DEPOSIT



CDWE BUILDING

NODE NUMBERS ON RIGHT
ELEMENT NUMBERS ON LEFT



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Condensate Demineralizer Waste Evaporator Building - Lumped Models for Normal Mode Analysis

FIGURE 3.7-15A

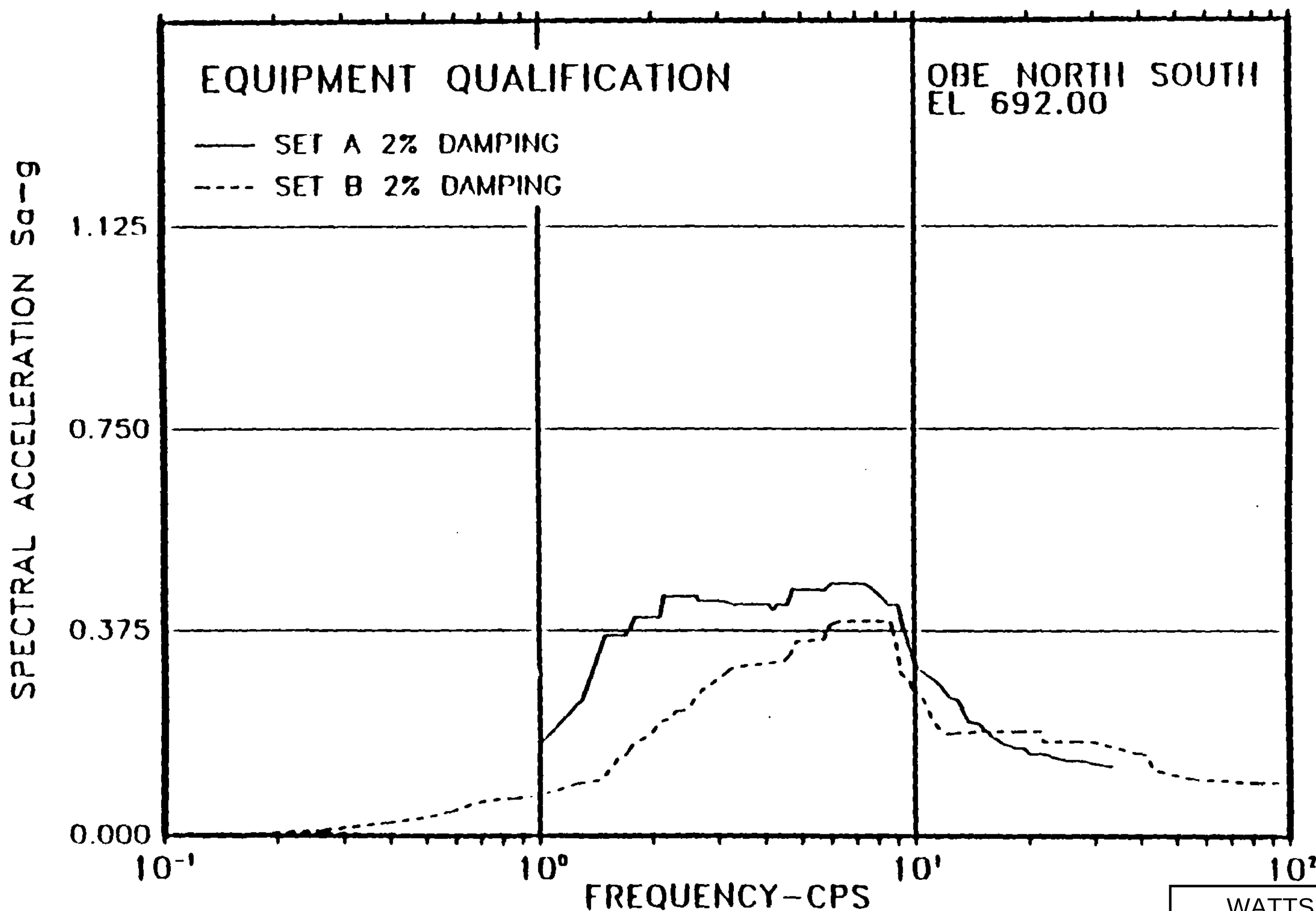
Figure 3.7-15B

Deleted

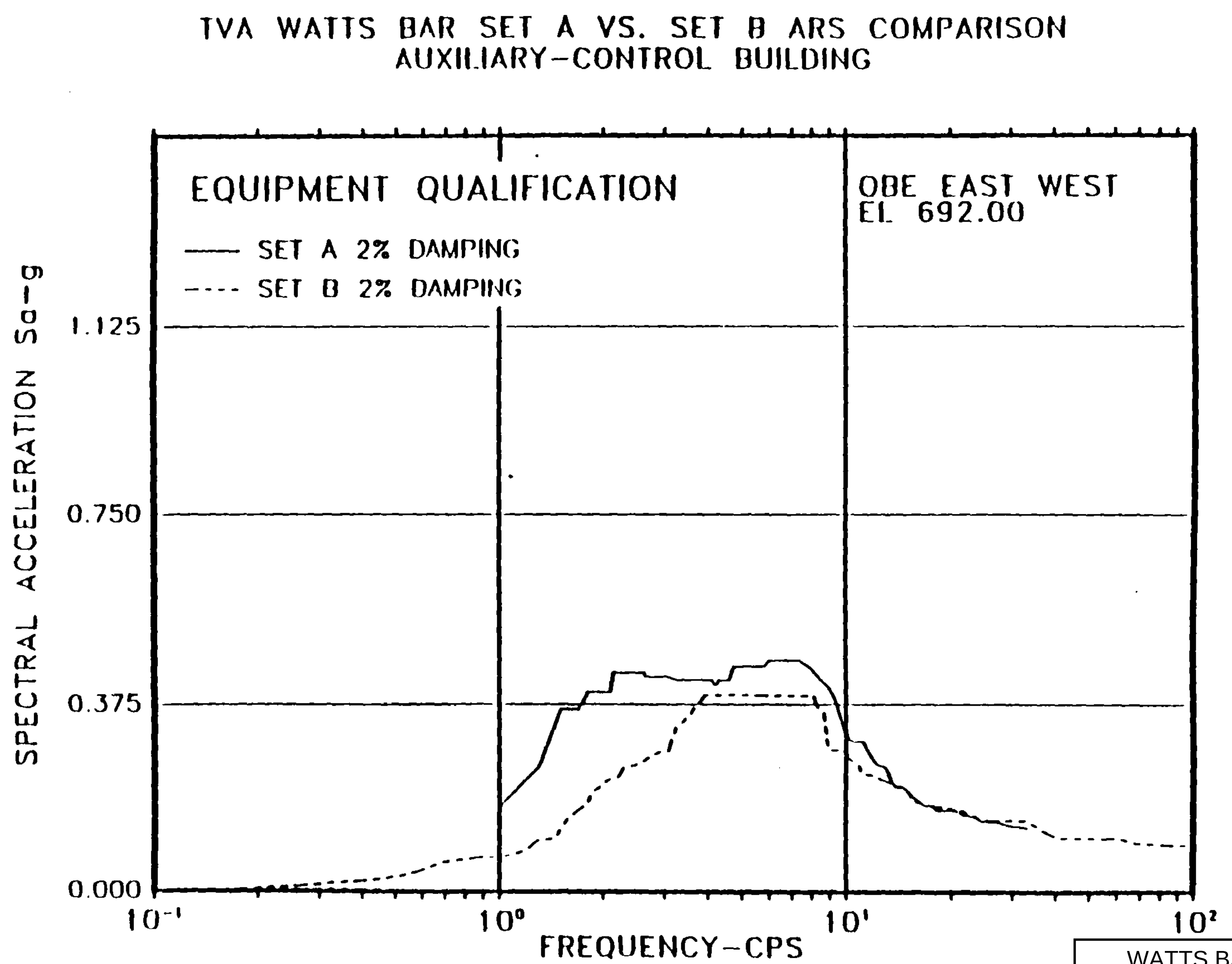
Figure 3.7-15C

Deleted

TVA WATTS BAR SET A VS. SET B ARS COMPARISON
AUXILIARY-CONTROL BUILDING



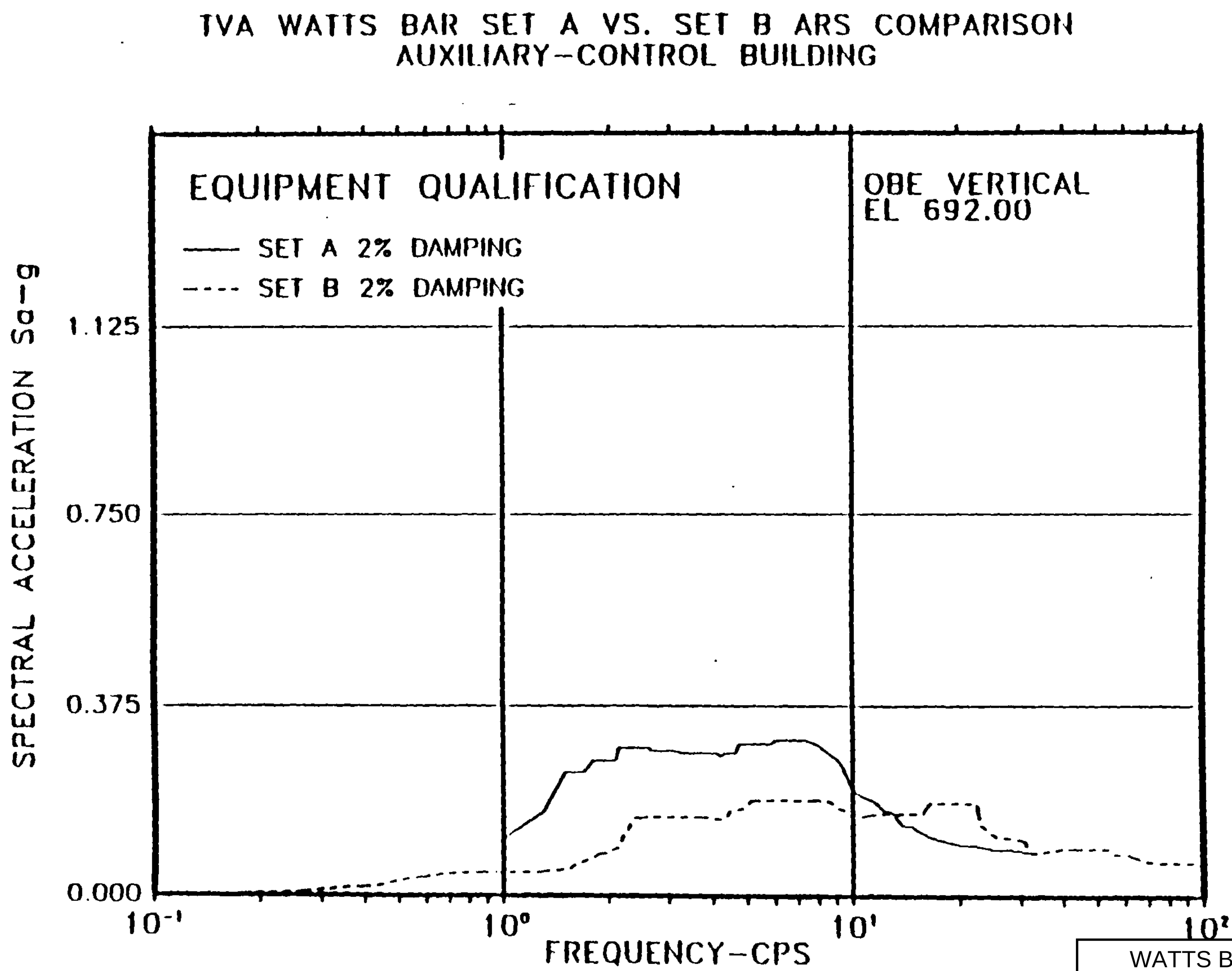
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Set A VS. Set B FIGURE 3.7-15D



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

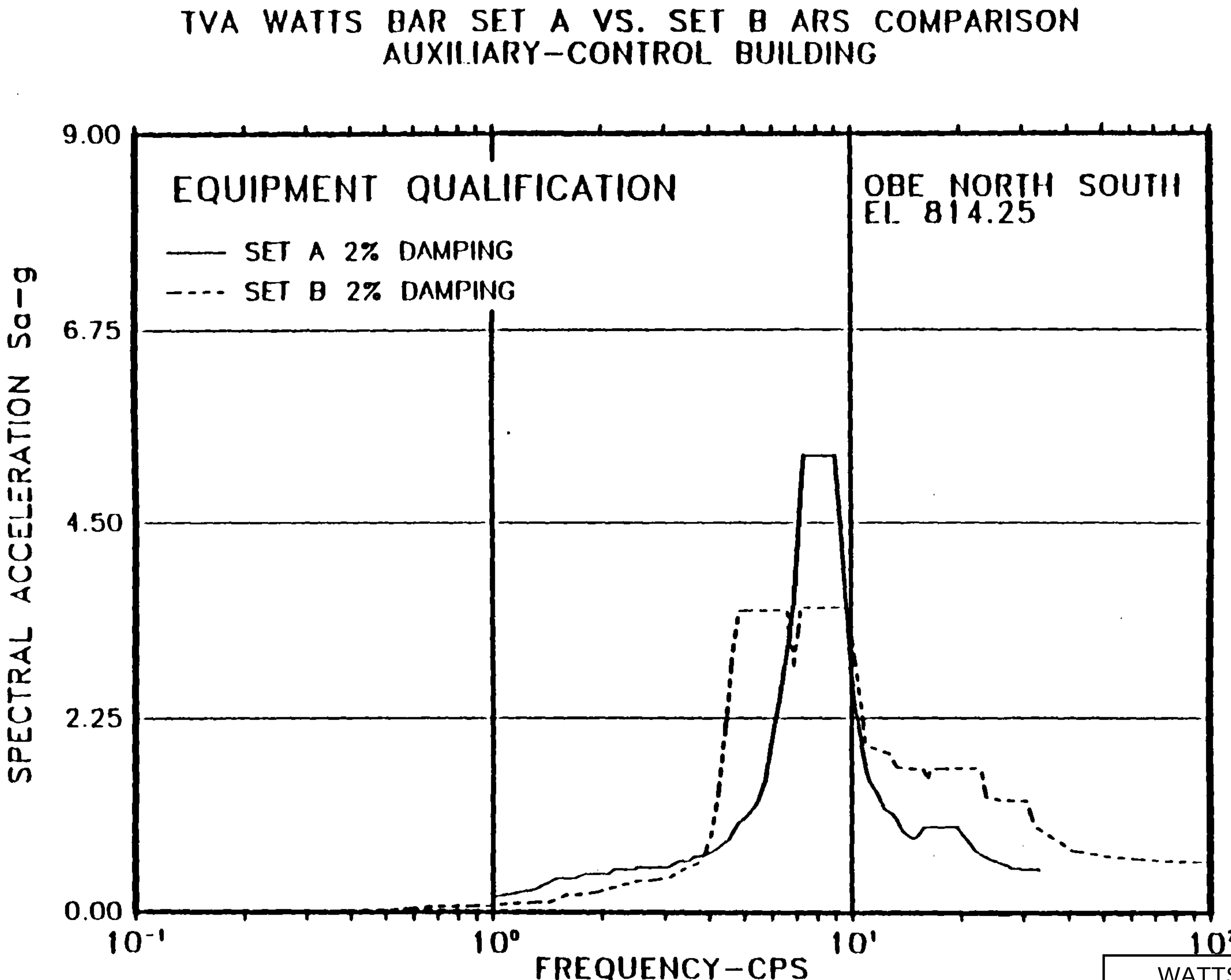
FIGURE 3.7-15E



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

FIGURE 3.7-15F

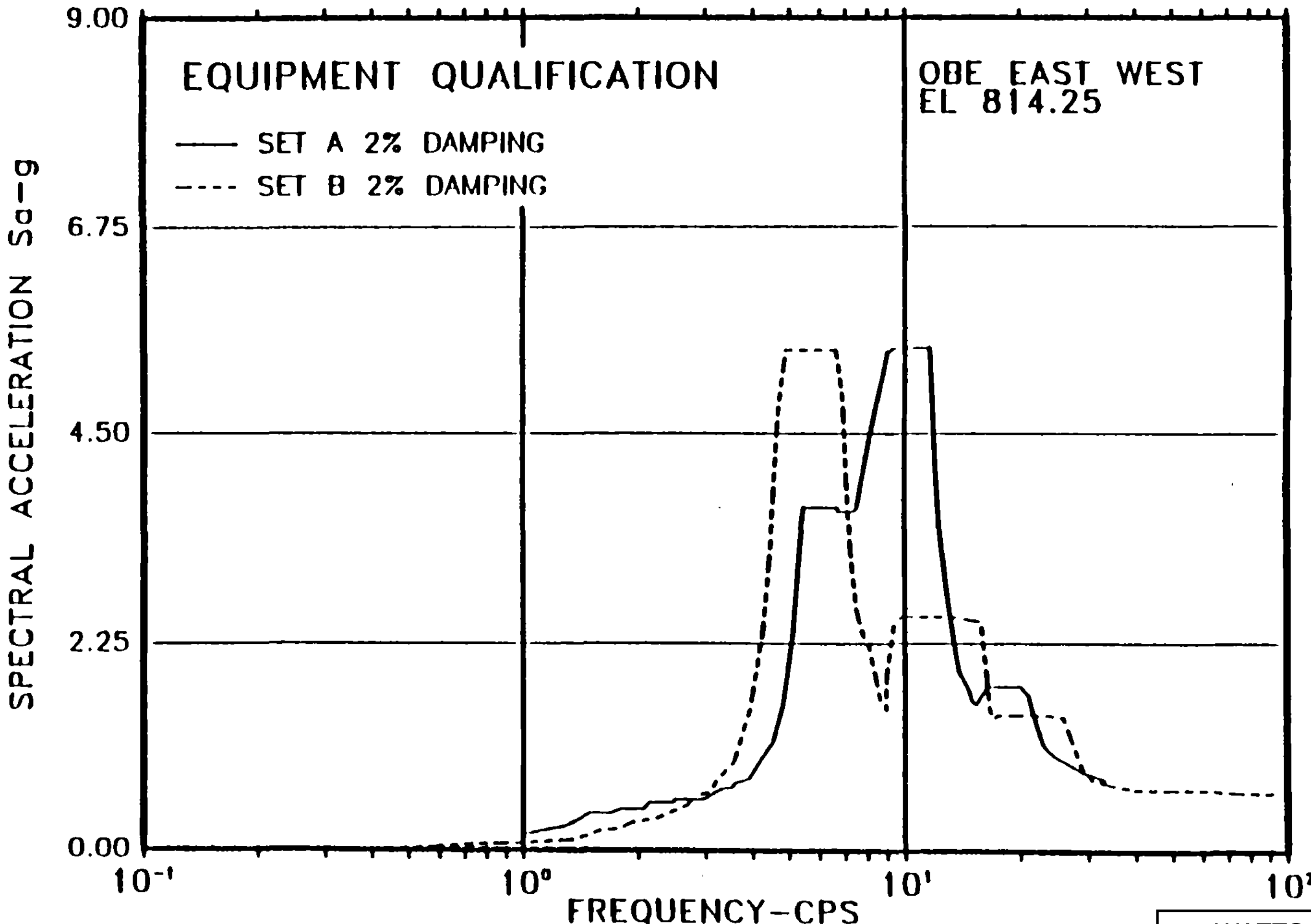


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

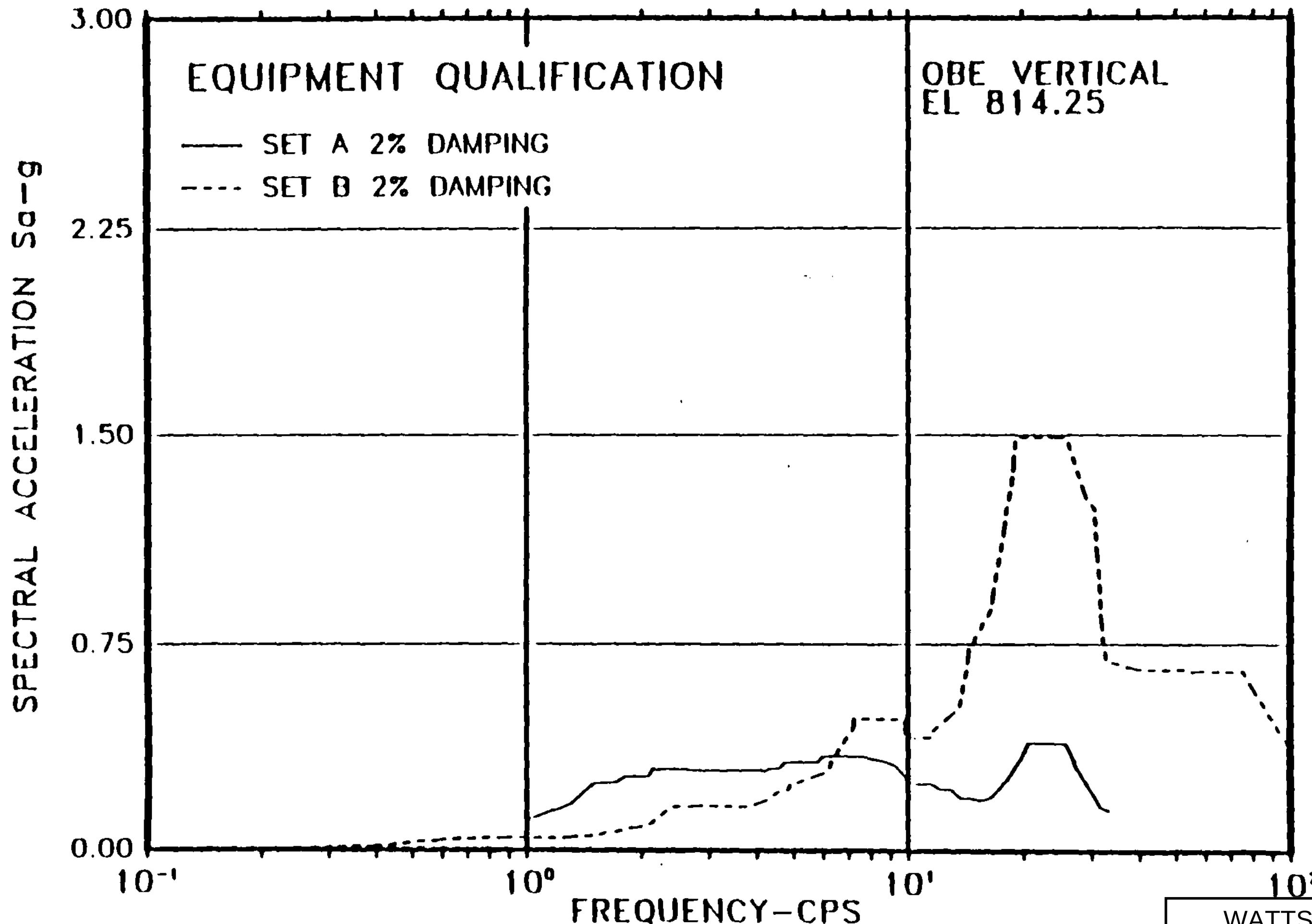
FIGURE 3.7-15G

TVA WATTS BAR SET A VS. SET B ARS COMPARISON
AUXILIARY-CONTROL BUILDING



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Set A VS. Set B FIGURE 3.7-15H

TVA WATTS BAR SET A VS. SET B ARS COMPARISON
AUXILIARY-CONTROL BUILDING



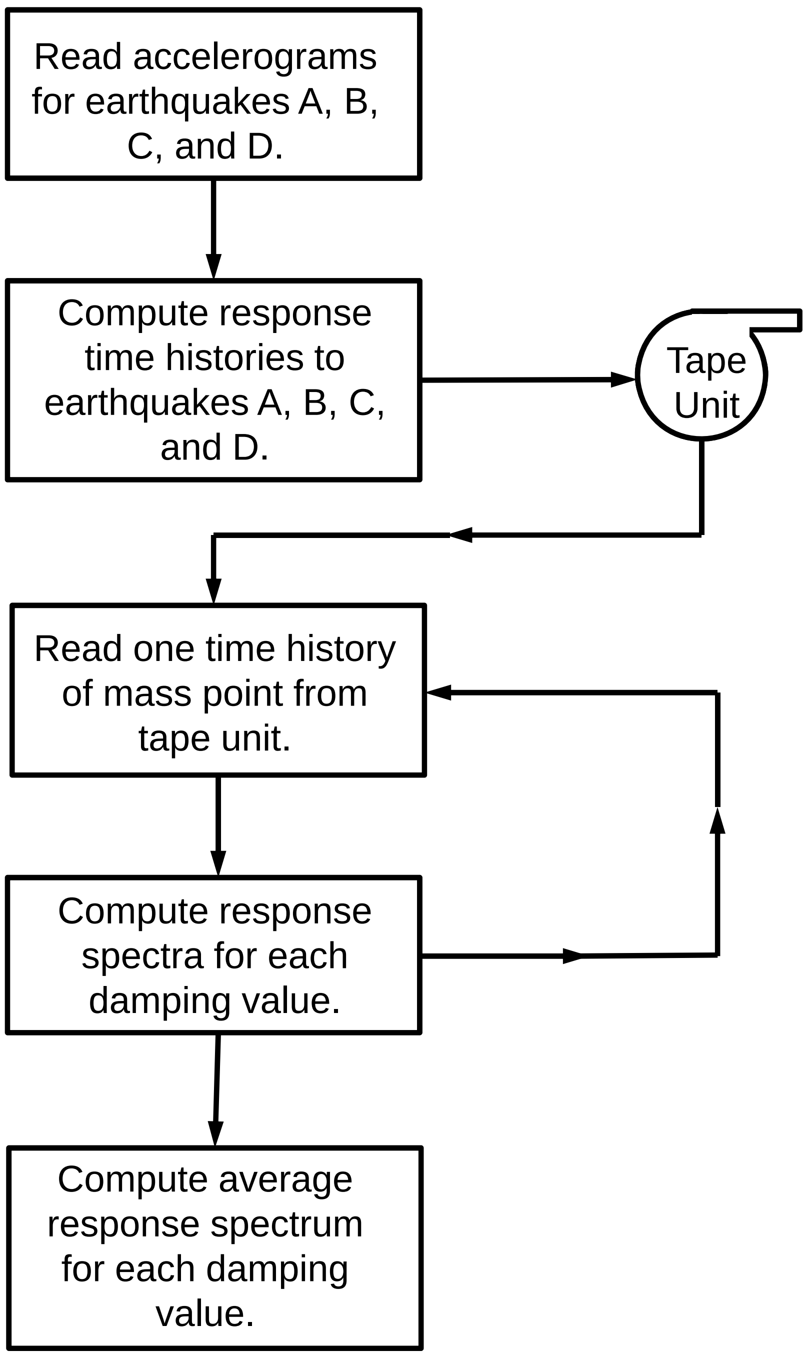
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

FIGURE 3.7-15I

FIGURE 3.7-16 THRU FIGURE 3.7-36

DELETED



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Flow Chart for
Development of Floor
Response Spectra

FIGURE 3.7-37

FIGURE 3.7-38

DELETED

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-39

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-40

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-41

CAD MAINTAINED DRAWING

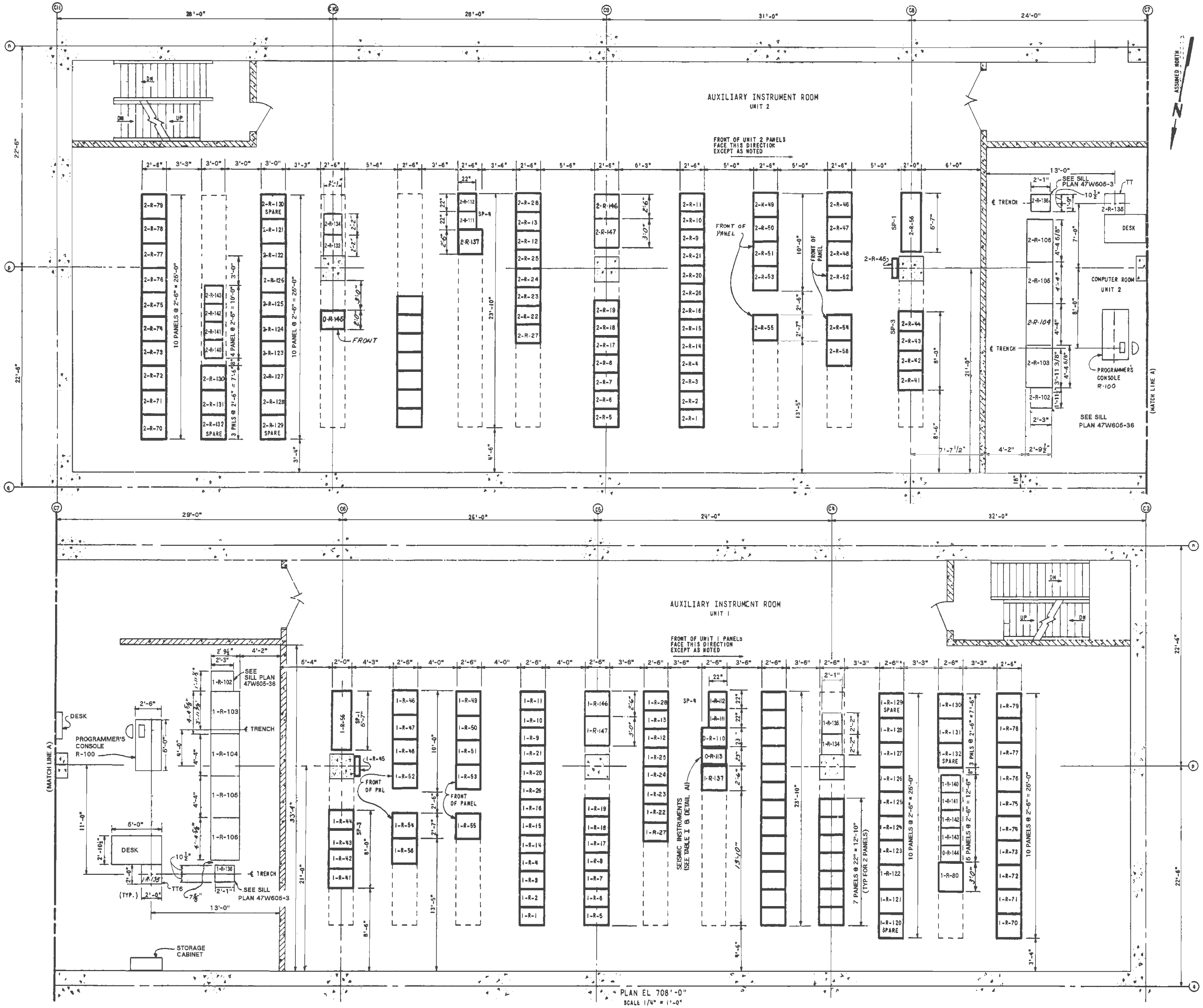
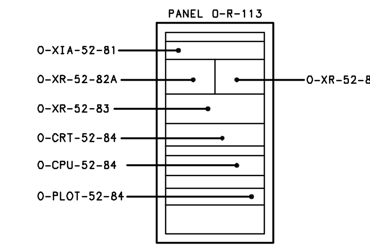


TABLE I
SEISMIC INSTRUMENTATION

INSTRUMENT NO.	FSAR		TVA IDENTIFICATION NO.	INSTRUMENT TYPE
	SECTION NO.	ITEM NO.		
	3.7.4.2		0-XIA-52-81	ALARM PANEL
			0-XR-52-82A	RECORDER PANEL
			0-XR-52-82B	RECORDER PANEL
			0-XR-52-83	RECORDER PANEL
			0-CRT-52-84	LCD PANEL
			0-CPU-52-84	CENTRAL CONTROLLER
			0-PLOT-52-84	PRINTER PANEL



DET A1
SEISMIC INSTRUMENT LOCATION

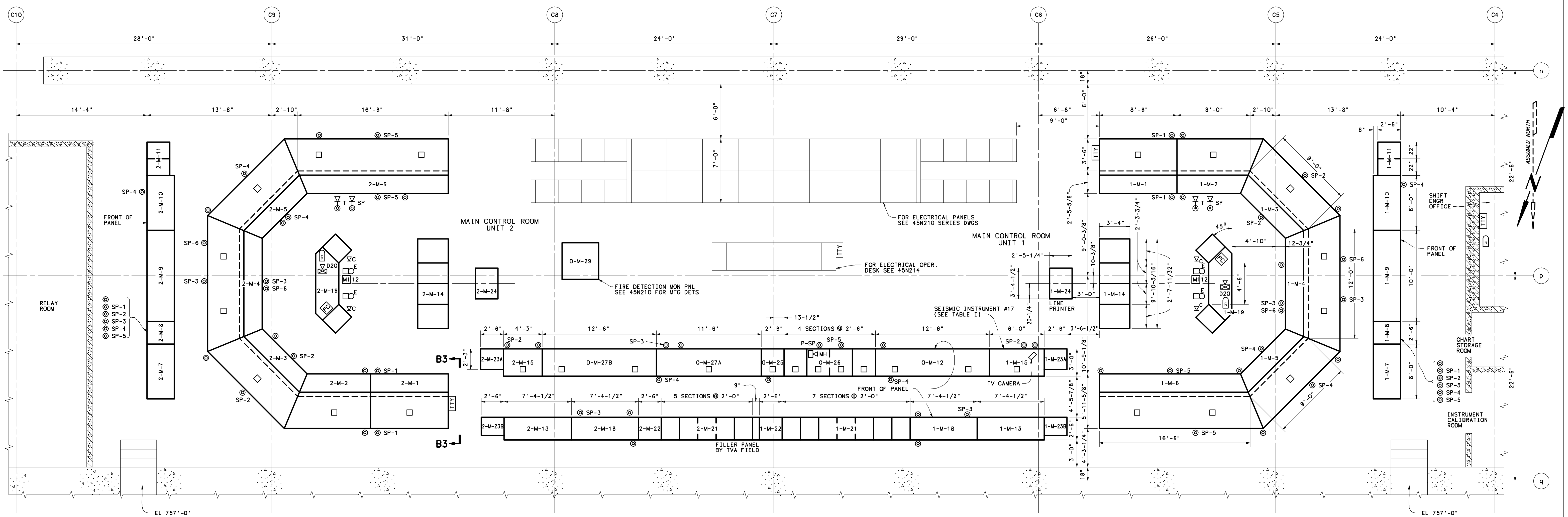
NOTES:
1. THIS DRAWING WAS ADAPTED FROM DRAWING NO. 47W605-2, REVISION 8.
2. REVISIONS OF DRAWING NO. 47W605-2 DO NOT REQUIRE UPDATING UNLESS EQUIPMENT IN TABLE I IS AFFECTED.

COMPANION DRAWINGS:
47W652-2,3,4
47N653-1,2,3
47B652

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

CONTROL BUILDING UNITS 1 & 2
SEISMIC INSTRUMENTATION
LOCATION OF SEISMIC INSTRUMENTS
AND PERIPHERAL EQUIPMENT
TVA DWG NO. 47W652-1 RD
FIGURE 3.7-42

CAD MAINTAINED DRAWING



PLAN EL 755'-0"

TABLE I
SEISMIC INSTRUMENTATION

INSTRUMENT NO.	FSAR SECTION NO.	FSAR ITEM NO.	TVA IDENTIFICATION NO.	INSTRUMENT TYPE	ORIENTATION
17	3.7.4.2	8	XA-55-15B	ANNUNCIATOR PANEL (SEE DRAWING 47W605-56)	

NOTES:

1. THIS DRAWING WAS ADAPTED FROM DRAWING NO. 47W605-1, REVISION 7.
2. REVISIONS OF DRAWING NO. 47W605-1 DO NOT REQUIRE UPDATING OF THIS DRAWING UNLESS EQUIPMENT IN TABLE I IS AFFECTED.

COMPANION DRAWINGS:
47W652-1,3,4
47N653-1,2,4
47B652

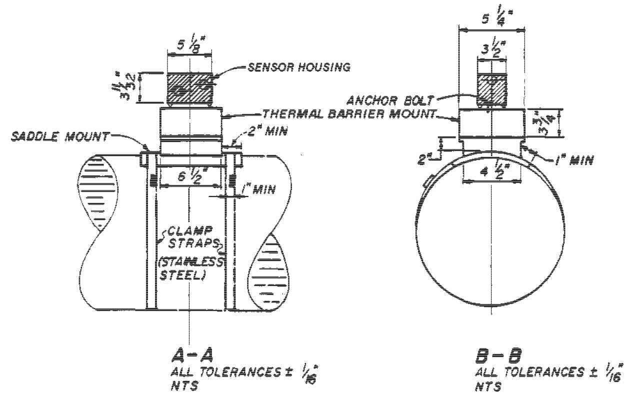
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

CONTROL BUILDING-UNITS 1 & 2
SEISMIC INSTRUMENTATION
LOCATION OF SEISMIC INSTRUMENTS
AND PERIPHERAL EQUIPMENT
TVA DWG NO. 47W652-2 RE
FIGURE 3.7-43

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-44

CAD MAINTAINED DRAWING



SADDLE MOUNT FOR PAR4CO-2 TRIAXIAL PEAK ACCELEROGRAPH

MOUNTING INSTRUCTIONS:

1. ALL PLATES OF THE SADDLE MOUNT ARE TO BE 1/2-INCH THICK STAINLESS STEEL.
2. LONG AXIS OF ACCELEROGRAPH UNIT IS TO BE ALIGNED WITH LONGITUDINAL AXIS OF PIPE.
3. SADDLE MOUNT AND STRAPS ARE TO BE ATTACHED DIRECTLY TO BARE PIPE.
4. THERMAL BARRIER MOUNT IS TO BE ENGDAHL ENTERPRISES MODEL NO. TBM400 AND WILL BE PROCURED BY THE FIELD.
5. SADDLE MOUNT IS TO HAVE FOUR 1/4" TAPPED THREAD HOLES IN A 4" x 4" SQUARE PATTERN. THERMAL BARRIER MOUNT IS TO BE ATTACHED TO SADDLE MOUNT WITH 1/4"-20UNC x 4" OR 4-1/4" STAINLESS STEEL BOLTS.

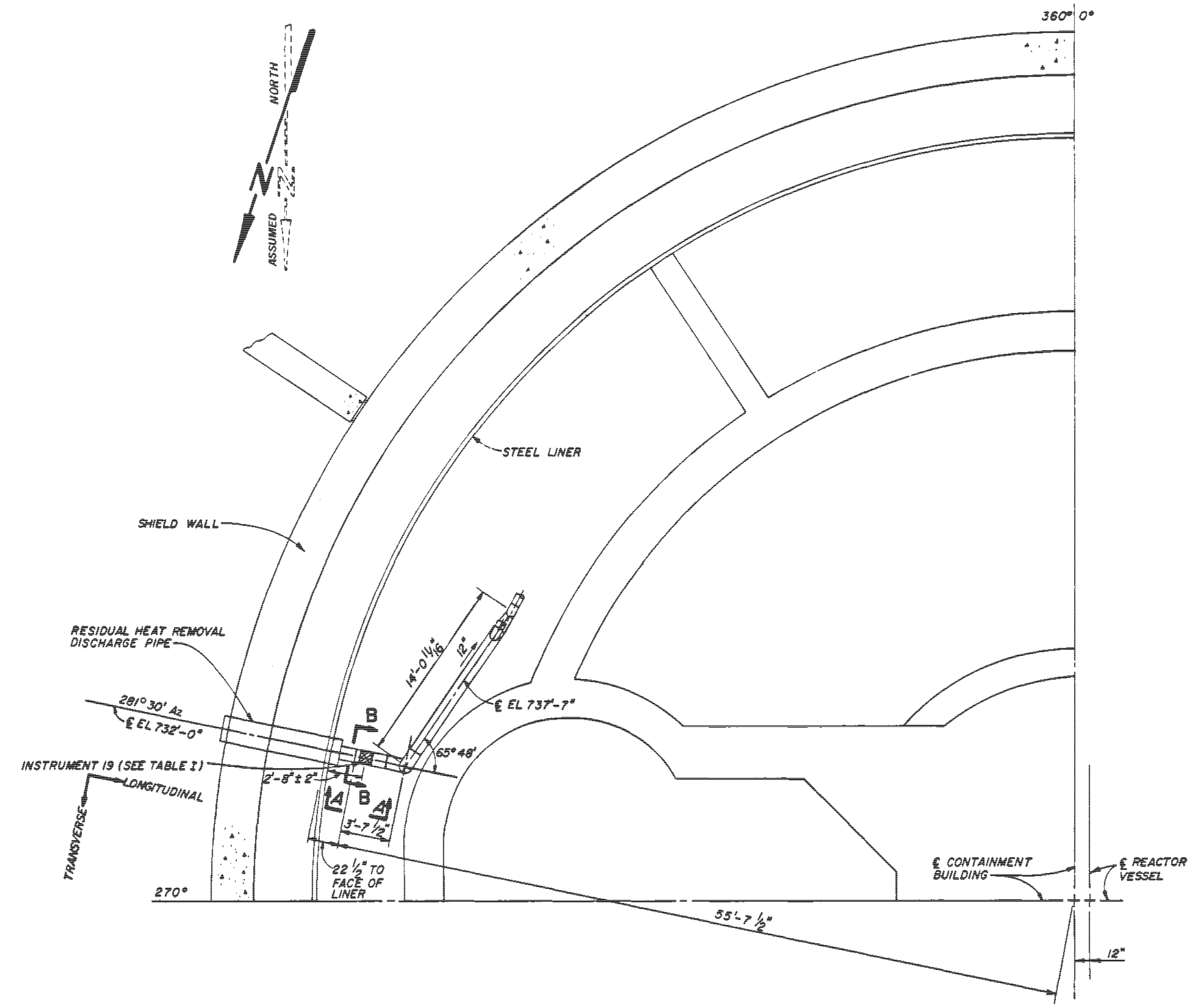


TABLE I
SEISMIC INSTRUMENTATION

INSTRUMENT NO.	FSAR SECTION NO.	FSAR ITEM NO.	TVA IDENTIFICATION NO.	INSTRUMENT TYPE
19			XR-52-76B ABANDONED IN PLACE	TRIAXIAL PEAK ACCELEROGRAPH PAR400-2 (SEE SECT. A-A & B-B FOR MOUNTING INSTRUCTIONS)

NOTES:

1. THIS DRAWING WAS ADAPTED FROM DRAWING NO. 47W432-3, REVISION 23.
2. REVISIONS OF DRAWING NO. 47W432-3 DO NOT REQUIRE UPDATING OF THIS DRAWING UNLESS EQUIPMENT IN TABLE I IS AFFECTED.
3. DESIGN INPUT DOCUMENTS: NRC REG GUIDE 1.12 AND ANSI/ANS-2.2, SECTION 4.1. THE LOCATION OF INSTRUMENT 19 DEVIATES FROM REG GUIDE 1.12 AS DOCUMENTED IN A MEMORANDUM DATED 3-5-79 (B45 880304 252).

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR UNIT 1
SEISMIC INSTRUMENTATION
LOCATION OF SEISMIC INSTRUMENTS
AND PERIPHEAL EQUIPMENT
TVA DWG NO. 47W652-4 RF
FIGURE 3.7-45

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Shield Building

The Shield Building is a Category I structure in its entirety and is designed to remain functional in the event of a Safe Shutdown Earthquake (SSE) or a tornado.

The Shield Building is designed as described in Sections 3.8.1.1 through 3.8.1.7. The evaluation and modification of the Shield Building reinforced concrete structure are optionally done using the ultimate strength design method in accordance with the codes, load definitions and load combinations specified in Appendix 3.8E.

3.8.1.1 Description of the Shield Building

The Shield Building, shown in Figure 3.8.1-1 through 3.8.1-7, is a reinforced concrete structure surrounding the steel containment structure and is designed to provide the following: radiation shielding from accident conditions, radiation shielding from parts of the Reactor Coolant System (RCS) during operation, and protection of the steel containment vessel from adverse atmospheric conditions and external missiles propelled by tornado winds. The Shield Building is a reinforced concrete cylinder supported by a circular base slab and covered at the top with a spherical dome. It is located adjacent to the concrete Auxiliary and Valve Room Buildings and is physically separated from them by a 1-inch fiberglass-filled expansion joint. There is a polyvinyl chloride seal placed in formed grooves on the face of the Shield Building where it abuts the Auxiliary Building, thus providing water tightness between the two buildings up to grade level of Elevation 728.0. The seal is embedded in the groove with epoxy adhesive mortar. The Shield Building is maintained watertight to Elevation 742.0. A sectional view through the Shield Building is shown in Figure 3.8.1-1. Only the base slab resists the LOCA pressure load which is transmitted to it through a steel plate liner anchored to its top face. For further discussion of the base slab see Section 3.8.5.

The cylinder wall is approximately 150 feet in height from the top of the base slab to the spring line of the dome. It has an inside diameter of 125 feet 1 inch and a thickness of 3 feet. Conventional steel reinforcing bars were used throughout the structure and were placed in a horizontal and vertical pattern in each face of the cylinder wall. The area of reinforcement in each direction of each face is not less than 0.0015 times the gross concrete area.

The effects of penetrations through the wall were considered. Penetrations, 12 inches or less in diameter, do not significantly disturb the reinforcing pattern in the wall. Therefore, no special reinforcing considerations were made at these areas.

For penetrations larger than 12 inches, reinforcing is terminated at the opening. Supplemental reinforcing is added, both vertically and horizontally, to replace the reinforcing, terminated at rectangular penetrations larger than 12 inches and circular penetrations larger than 24 inches. The amount of supplemental reinforcing added is equal to or greater than the amount of

reinforcing removed and is placed adjacent to the penetration. In addition, rectangular penetrations in the wall have diagonal reinforcing across the corners. Reinforcing bars were lap spliced in accordance with ACI 318-71 code requirements for strength design or have been cad-welded.

Reinforcing steel bars in the dome were arranged in a radial and circumferential pattern.

A ring tension beam is provided to resist the outward thrust from the dome roof. The tensile force in the ring beam is resisted by 24 No. 11 reinforcing bars. These bars are spliced with mechanical splices that are uniformly staggered at least 6 feet on center around the circumference of the ring beam. Therefore, at any cross section in a length of 6 feet, only three bars are spliced out of the total of 24 bars, and not more than two of these are in any one layer. That is, at any section, 21 bars are continuous and unspliced. These continuous, unspliced bars alone will carry the imposed load with only a 15 percent increase in stress. Stirrups enclosing the main reinforcement are spaced on 15-inch centers.

To facilitate removal of the old steam generators (OSGs) and installation of the replacement steam generators (RSGs) during the Unit 1 steam generator replacement (SGR), two construction
opening were cut in the concrete shield building dome. These openings were restored by splicing
new reinforcing bar to the existing reinforcing bar using Bar-Lock couples, Cadwelds, and/or welding and pouring new concrete to close the openings.

3.8.1.1.1 <u>Equipment Hatch Doors and Sleeves</u>

As shown in Figure 3.8.1-8, a double-leaf equipment door installed in a sleeve is provided for each Reactor Building. The steel sleeve forms an access through the Shield Building wall to the equipment hatch in the Containment Vessel. Each sleeve extends from inside the Shield Building to the shielded passageway leading to the Auxiliary Building floor Elevation 757.0. Each door is of the hinged, double-leaf, marine type with seals for providing an airtight closure between the annulus surrounding the steel containment vessel and the inside of the Auxiliary Building. A door will normally be opened only when the reactor is in the shutdown, depressurized condition such that secondary containment is not required.

The sleeves, embedded in the Shield Building walls, are of welded steel construction, rectangular in cross section, with corners fabricated to a radius. They form clear passageways 20 feet wide and 17 feet-8 inches high through the concrete walls of the Shield Buildings.

Floors in the sleeves are at Elevation 756.63 coinciding with the Elevation of the operating floors in the Reactor Buildings.

The doors are hinged to the sleeves on the end toward the outside of the Shield Building wall and are of welded construction consisting of structural shapes with a steel skin plate.

Sealing of a door when closed is by means of solid, molded rubber seals mounted on the door. The seals contact the edge of the sleeve at the top and sides, a removable seal bar at the floor level, and a sealing bar at the meeting line of the two leaves. Penetrations through the doors are sealed with solid rubber O-ring type seals.

The doors are opened and closed manually. Latching of the doors in the closed position is accomplished by hand-lever operated dogs acting on wedge surfaces around the perimeter and meeting edges of the door leaves.

The doors are part of the airtight closure between the annulus surrounding the Containment Vessel and the inside of the Auxiliary Building. These doors are to remain closed during unit operation and will only be opened during unit shutdown.

The door and sleeves will maintain their structural integrity and remain operational after being subjected to the environmental or accident conditions listed in Section 3.8.1.4.

3.8.1.2 Applicable Codes, Standards, and Specifications

The structural design of the reinforced concrete Shield Building is in compliance with the proposed ACI-ASME (ACI-359) Code for Concrete Reactor Vessels and Containment, Article CC-3000, as issued for trial use, April 1973, for the loading combinations defined in Table 3.8.1-1. Allowable stresses are based on this code with the exception of allowable tangential shear stresses in walls where the ACI 318-71 code is used. Detailing of reinforcing around opening of circular walls is based on the ACI Chimney Code (ACI 307-69), Sections 4.4.4 through 4.4.7. All reinforcing steel conforms to the requirements of ASTM Designation A615-72, Grade 60.

Unless otherwise indicated in the UFSAR, the design and construction of the Shield Building is based upon the appropriate sections of the following codes, standards, and specifications. Modifications to these codes, standards, and specifications are made where necessary to meet the specific requirements of the structures.

Where date of edition, copyright, or addendum is specified, earlier versions of the listed documents were not used. In some instances, later revisions of the listed documents were used where design safety was not compromised.

1. American Concrete Institute (ACI)

ACI 214-77	Recommended Practice for Evaluation of Strength Test Results of Concrete
ACI 318-71	Building Code Requirements for Reinforced Concrete
ACI 359	Code for Concrete Reactor Vessels and Containments, (Proposed ACI-ASME Code ACI-359 (Article CC-3000) As issued for trial use April, 1973)

ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting
ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete
ACI 307-69	Specification For the Design and Construction of Reinforced Concrete Chimneys

2. American Institute of Steel Construction (AISC)

   'Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings,' adopted February 12, 1969, except welded construction is in accordance with Item 4 below.

3. American Society for Testing and Materials (ASTM), 1975 Annual Book of ASTM Standards. Specific standards are identified in Section 3.8.1.6.

4. American Welding Society (AWS)

   Structural Welding Code, AWS D1.1-72 with Revisions 1-73 and 2-74 except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders.

   Visual inspection of structural welds will meet the minimum requirements of Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 as specified on the design drawings or other engineering design output. See Item 12 below.

   'Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Connections,' AWS D12.1-61.

5. Uniform Building Code, International Conference of Building Officials, Los Angeles, 1970 edition.

6. Southern Standard Building Code, 1969 edition, 1971 Rev.

7. 'Nuclear Reactors and Earthquakes,' USAEC Report TID-7024, August 1963.

8. <u>American Society of Civil Engineers Transactions</u>, Volume 126, Part II, Paper No. 3269, 'Wind Forces on Structures,' 1961.

9. <u>Code of Federal Regulations</u> Title 29, Chapter XVII, "Occupational Safety and Health Standards," Part 1910.

10. NRC Regulatory Guides;

RG 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures
RG 1.12	Instrumentation for Earthquakes
RG 1.15	Testing of Reinforcing Bars for Category I Concrete Structures
RG 1.31	Control of Ferrite Content in Stainless Steel Weld Metal
RG 1.55	Concrete Placement in Category I Structures.

11. <u>Nuclear Construction Issues Group (NCIG)</u>

NCIG-01, Revision 2 - Visual Welding Acceptance Criteria (VWAC) for Structural Welding

NCIG-02, Revision 0 - Sampling Plan for Visual Reinspection of Welds

The referenced NCIG documents may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS.

NCIG-02, Revision 0, was used as the original basis for the Department of Energy (DOE) Weld Evaluation Project (WEP) EG&G Idaho, Incorporated, statistical assessment of TVA performed welding at WBNP. Any further sampling reinspections of structural welds subsequent to issuance of NCIG-02, Revision 2, are performed in accordance with NCIG-02, Revision 2 requirements.

The applicability of the NCIG documents is specified in controlled design output documents such as drawings and construction specifications. Inspectors performing visual weld examination to the criteria of NCIG-01 are trained in the subject criteria.

12. <u>TVA Reports</u>

CEB 86-12	Study of Long Term Concrete Strength at Sequoyah and Watts Bar Nuclear Plants
CEB 86-19-C	Concrete Quality Evaluation

3.8.1.3 Loads and Loading Combinations

The Shield Building dome and cylinder wall are subjected to the following loads. Design loading combinations utilized to examine the effects of localized areas are shown in Tables 3.8.1-1 and 3.8.1-2.

Dead Load

This includes weight of the concrete structure plus any other permanent load contributing to stress, such as equipment, piping, and cable trays suspended from the structures.

Earth Pressure

The static soil pressure was computed using Earth Pressure Standards from TVA's General Standards which incorporate Coulomb's "wedge of pressure" theory.

Standard soil properties for fine grained rolled fill are as follows:

Angle of internal friction	= 32 degrees
Angle of friction between soil and building	= 16 degrees
Dry weight	= 120 lb/cu ft
Buoyant weight	= 65 lb/cu ft

Due to adjacent structures the soil does not completely surround the Shield Building but lies in a 185-degree segment around it. The soil was backfilled to a height of 31 feet above the base slab. A surcharge of 200 psf was used.

Hydrostatic Pressure

Uplift forces and lateral static pressure were computed using the full hydrostatic head measured from the water surface. Water surface elevations from the probable maximum flood (Section 2.4) were used in determining hydrostatic heads.

Due to water seals between the Shield Building and adjacent structures, the lateral hydrostatic pressure was applied only to one-half of the circumference for the drawn down ground water table. For the probable maximum flood the adjacent structures are allowed to flood and lateral hydrostatic pressure was applied around the full circumference.

Loss-of-Coolant Accident (LOCA)

In addition to the reactions of the containment vessel and interior concrete due to the LOCA pressure transients, the LOCA produced uplift forces on the steam generator or reactor coolant pump anchors in the base slab. The LOCA also increased the temperature in the annulus space between the Containment Vessel and the Shield Building. This produced a nonlinear temperature gradient across the cylinder wall and dome. A typical gradient in shown in Figure 3.8.1-9.

Normal Temperature Gradient

The temperature gradient for normal plant operation was considered as uniformly varying through the section. The maximum temperature gradient occurs just above grade when the plant is in operation and a minimum ambient temperature exists. The normal temperature difference across the wall varies from a minimum of 35°F below grade to the maximum of 85°F as shown on Figure 3.8.1-9.

Operational Basis Earthquake (OBE)

The plant was designed to remain operational under the OBE. The OBE has a maximum acceleration of 0.09g horizontally and 0.06g vertically. In addition to the maximum values of the structural response in terms of displacement, acceleration, shear, moment, torque and axial force, the soil pressure and hydrostatic pressures were increased due to seismic motions. The static soil pressure was increased 23% for a dry fill and 11% for a saturated fill. This incremental increase was a triangle of pressure with the apex at the rock surface and the maximum ordinate at the ground surface. The hydrostatic pressure of the water within the fill was increased by 11%.

This incremental increase was a triangle of pressure with the apex at the water surface and maximum ordinate at the rock surface. The magnitude of these increases were determined by shaking table experiments performed for another TVA project. The reaction from earthquake motion on the compressed expansion joint material separating the adjacent Auxiliary and Valve Room Buildings was also taken into consideration.

Safe Shutdown Earthquake (SSE)

The plant was designed to have the capability for safe shutdown for the SSE (maximum acceleration of 0.18g horizontally and 0.12g vertically). The incremental pressure increase for soil and hydrostatic pressure was twice that for the OBE.

Live Load

Live load includes non pipe hanger loads, plus any other permanent load such as crane loads, etc. Snow load of 20 psf was considered in the design live load.

Tornado

The tornado was assumed to have an "eye" whose pressure is 3 psi below ambient, a "funnel" having a rotational velocity of 300 mph, and a translational speed of 60 mph. The Shield Building was designed for wind loads corresponding to 360 mph and a maximum internal pressure of 3 psi. Maximum wind velocity and maximum internal pressure loading do not coincide as shown by Figure 3.3-1. The ultimate capacity of the structure in flexure or shear is not exceeded under the combined pressure and wind velocity loadings of Figure 3.3-1.

The adjacent structures disturb the air flow around the Shield Building. The only method to determine the actual pressure distribution on the structure is by a model test. In lieu of model test, several cases of extreme pressure distributions were analyzed in an attempt to bracket the actual stresses. The normal maximum wind loading was based on Figure 1(b), from ASCE Paper 3269, "Wind Forces on Structures."

Tornado missiles are described in Section 3.5.

Construction Loads - *Historical Information*

The dome was poured in two lifts. The first lift is a 9-inch pour supported by temporary shoring bearing on the Containment Vessel. The first lift was designed to support the wet concrete dead load of the second lift plus a construction load of 50 psf.

3.8.1.4 Design and Analysis Procedures

Base Slab

The base slab is discussed in Section 3.8.5.

Cylinder Wall and Dome

The stiffness of the cylinder wall was small in comparison to that of the base slab and the cylinder wall was assumed fixed at the base. The height of the wall was such that the effect of discontinuity at one end was negligible when considering discontinuity at the other end.

For symmetrical loadings, the edge forces at the point of discontinuity were determined by writing the equations of the primary system and the equation of compatibility. The discontinuity stresses from the edge forces were superimposed on the membrane stresses. The above analysis was checked by two independent computer analyses ("Axisymmetric Finite Element Analysis, AMG032" and GENSHL 2). Unsymmetrical loadings, such as wind, were analyzed by using computer code, GENSHL. These loads were approximated through a Fourier series.

Creep and Shrinkage Effects

Creep was not considered in the design of the Shield Building. Sustained loads are essentially the dead weight loads of the structure itself with subsequent stress levels too low to influence creep deformations to any significant degree particularly since these deformations do not cause differential settlements in the structure.

Shrinkage effects are considered in the design of all structures by estimating the temperature change from peak hydration temperatures to final operating temperature conditions. In addition drying shrinkage effects are considered in all members which have an average drying path of less than 15 inches. The methods used to consider these effects are explained in an ACI Committee 207 Report 70-45, "Effect of Restraint, Volume Change, and Reinforcement on Cracking of Massive Concrete" published in July 1973.

The effects of base restraint on the cracking of a circular structure is essentially the same as the effects on a wall of equal thickness whose length is equal to the outside diameter of the circular structure.

The Shield Building was not only designed to restrict shrinkage cracking, thus holding the cracks to a minimum acceptable size, but was also waterproofed on the exterior surface below grade to eliminate possible seepage. The portion above grade is essentially out of the restraint zone and will therefore be relatively free from shrinkage cracking.

Tangential Shear

The tangential shears induced by earthquake and wind forces were assumed to vary from zero over a thickness of wall located at the extremes of a diameter parallel to the line of action of the shearing force to a maximum on a wall thickness located at the extremes of a diameter normal to the line of action of the shearing force. Distribution was assumed proportional to the cosine of the polar angle measured from the diameter normal to the line of action of the shear force with a maximum allowable shear stress in the concrete limited to 247 psi according to special provisions for shear in walls in the ACI 318-71 code.

Seismic

See Section 3.7 for a detailed description of the seismic analysis.

Equipment Hatch Doors and Sleeves

For the closed position, the structural members of the door leaves were designed as simple beams under uniformly distributed loading with the end reactions carried by the sleeve. Loads at the dogging wedges were carried to the sleeve as concentrated loads.

For the open position, the door leaves were treated as cantilever structures, and the hinge members and sleeve were designed for the resulting concentrated loads.

Design of the doors and sleeves was by TVA without the use of a computer program.

Under normal operating conditions, air pressure equal to 5 inches of water is exerted on the Auxiliary Building side of the doors. Under accident or tornado conditions, the doors are subjected to air pressure. Environmental and accident conditions which were considered in the design of the doors and sleeves are as follows:

1. The OBE and the SSE with accelerations as hereinafter defined.

2. An inadvertent release of the cooling sprays in the Containment Vessel will cause a pressure drop within the annulus surrounding it and result in an air pressure load of 2 psi on the Auxiliary Building side of the doors and sleeves. Duration of this condition will be for a few hours maximum.

3. A tornado condition which causes a pressure drop within the Auxiliary Building will result in a pressure of 3 psi on the annulus side of the doors. Duration will be for 3 seconds.

4. A LOCA accident in the Containment Vessel which will result in a pressure equal to 3/4 inch of water on the Auxiliary Building side of the doors. A partial vacuum is created in the annulus by vacuum pumps, and this condition may exist for a period of several months.

Earthquake accelerations used in design of the doors and sleeves were determined by dynamic analysis of the supporting structure of the Shield Building. Accelerations at the centerline of the equipment hatch for the OBE are as follows:

Lateral (north-south)	0.16g
Lateral (east-west)	0.16g
Vertical	0.12g

Accelerations at the centerline of equipment hatch for a SSE are as follows:

Lateral (north-south)	0.36g
Lateral (east-west)	0.36g
Vertical	0.23g

These accelerations were used as static loads for determining component and member sizes. After establishing the component and member sizes, a dynamic analysis, using appropriate response spectrum, was made of each sleeve and its doors to determine that allowable stresses had not been exceeded.

3.8.1.5 Structural Acceptance Criteria

Controlling Conditions - Shield Building Structure

The SSE in combination with a LOCA (load combination 8) produced the largest overturning moment. For this combination, the percent of the base slab in compression was 51% and the factor of safety for overturning was 1.74.

The uplift on the equipment from the LOCA combined with the SSE controlled the design of the base slab.

Minimum steel requirements of 0.65 square inches per foot (minimum steel ratio of 0.0015 in each face and in both vertical and horizontal directions) controlled the inside face vertical steel requirements throughout the shell and the inside face horizontal steel requirements above grade.

The SSE in load combination 8 controlled the design of the outside face vertical reinforcement at the base of the cylinder wall. Due to earth and hydrostatic pressure, outside face horizontal reinforcement requirements were greatest 16 feet above the base of the cylinder wall at Elevation 713.0.

The construction loading controlled the reinforcement design in the dome and the upper portion of the cylinder wall.

The SSE produced a maximum tangential shear stress at the base of the wall of 189.7 psi which was 76.8% of the allowable.

The effects of repeated reactor shutdowns and startups during the plant's life will not degrade the above margins of safety because the Shield Building is minimally affected by these operations. The only effects from normal operations are from interior temperature changes which are insignificant compared to normal exterior temperature variations.

Equipment Hatch Doors and Sleeves

Allowable stresses for load combinations used for the various parts are given in Table 3.8.1-2. For normal load conditions, the allowable stresses provide safety factors of 1.67 (F_y/0.6 F_y) to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts. For a limiting condition such as a Safe Shutdown Earthquake (SSE), stresses do not exceed 0.9 yield.

3.8.1.6 Materials, Quality Control and Special Construction Techniques

General

The principal materials used in the construction of the Shield Building base slab, wall, and dome were concrete and reinforcing steel. Steel is used for the structural parts of the equipment hatch doors and sleeves with rubber used for the seals.

3.8.1.6.1 Materials

Concrete

Cement conformed to ASTM Specification C150-72 Type I. The guaranteed 28-day mortar strength was 5025 psi with a guaranteed standard deviation of 395 psi and a guaranteed maximum tricalcium aluminate content of 9.5%.

Aggregates conformed to ASTM Specification C-33-71a and were manufactured of crushed limestone.

Water for mixing concrete and also for washing the aggregates and curing concrete was tested prior to use in accordance with Corps of Engineers test method CRD-C400.

The fly ash used at Watts Bar is in general accordance with the ASTM C618-73, except for the loss of ignition and fineness of pozolanic index parameters. TVA specific requirements for loss of ignition are more restrictive while the fineness pozolanic index is less restrictive than the ASTM requirements. (See Section 3.8.3.2.1.a for more details). Sampling and testing was performed in accordance with ASTM C 311.

Air-entraining admixtures conformed to ASTM Specification C-260-69.

Water-reducing agent used for concrete mixtures containing fly ash was selected based on demonstrated achievement of TVA specified concrete strength of a control mix by actual testing.

Reinforcing Steel

Reinforcing steel conformed to ASTM Designation A615-72, Grade 60.

For the Unit 1 steam generator replacement, reinforcing steel used in the restoration of the shield building construction openings conformed to ASTM A615, Grade 60.

Bar-Lock Couplers

During the Unit 1 steam generator replacement, Bar-Lock couplers were used to splice the new reinforcing bar to the existing reinforcing bar during the restoration of the shield building construction openings. Bar-Lock couplers are manufactured from seamless hot-rolled steel tube conforming to ASTM A-519 specification, with minimum tensile strength exceeding 100,000 psi.

Equipment Hatch Sleeves and Doors

The structural parts of the sleeves and doors are fabricated from ASTM A36 steel.

3.8.1.6.2 Quality Control

Concrete

Concrete was produced in a central batch and mixing plant until 1977, and central batch and transit mix after 1977. A materials engineering unit was specifically responsible for control, documentation, and daily review of test data.

Aggregate gradation and deleterious material was checked daily. All coarse aggregate was rinsed and resized. The gradation of the fine aggregate and the amount finer than the No. 200 sieve conformed to specifications.

The other concrete material was also subject to periodic tests (see Section 3.8.3.2).

The specified strength of the concrete was 4000 psi at 28 days. Some concrete did not meet specification requirements. This was evaluated and documented in the Report CEB-86-19-C "Concrete Quality Evaluation." The results have been documented in affected calculation packages and drawings.

A testing program conducted at the site compared strengths of cylinders and concrete from 3-foot-thick wall sections subjected to exterior exposures. The results of this test program are documented in TVA report CEB 86-12, "Study of Long-Term Concrete Strength at Sequoyah and Watts Bar Nuclear Plants." These tests demonstrated the long term compressive strength gain with age which have occurred. The strength gain and age was generally 2600 psi beyond 28 days and 1300 psi beyond 90 days.

During the Unit 1 steam generator replacement, concrete used for the restoration of the shield building dome construction openings was provided in accordance with Specification 24900-C-321. The concrete was designed to achieve a minimum strength of 4000 psi at seven days.

Reinforcing Steel

Testing of reinforcing steel conformed to Regulatory Guide 1.15.

Cadweld splices conformed to Regulatory Guide 1.10.

Bar-Lock Couplers

Manufacturing processes and procedures for the Bar-Lock couplers used in the restoration of the Shield Building openings flowing installation of the replacement steam generators complied with the applicable provisions of ANSI/ASME N45.2. Qualification testing of the Bar-Lock couplers conformed to ASME Code, Section III, Division 2, CC-4333.2, “ Splice System Qualification Requirements”.

Equipment Hatch Doors and Sleeves

Design by TVA and erection by TVA were in accordance with TVA's quality assurance program. Design and fabrication by the contractor were in accordance with the contractor's quality assurance program which was reviewed and approved by TVA's design engineers. The contractor's quality assurance program covers the criteria in Appendix B of 10 CFR 50. Fabrication procedures such as welding and nondestructive testing were included in appendices to the contractor's quality assurance program. ASTM standards were used for the material specifications and certified mill test reports were provided by the contractor for materials used for load carrying members.
Material used for seals including O-rings, was certified by a rubber technologist as being capable of withstanding the radiation and temperature conditions existing during a LOCA accident. This certification is based on testing and evaluation of seal materials performed under contract for TVA by Presray Corporation.

3.8.1.6.3 Construction Techniques - *Historical Information*

The walls of the Shield Building from the base slab to the bottom of the ring beam were constructed using conventional forms. The concrete pouring was performed in two stages to facilitate other construction work in the building. The first stage consisted of concrete pours to Elevation 762.0 and the second stage consisted of the remaining height of wall. Concrete temperatures were monitored throughout for a minimum period of 3 days during cold weather to assure cold weather protection requirements.

The dome roof was placed in two lifts with each lift divided into three basic rings and each ring divided into radial segments. The Steel Containment Vessel (SCV) is designed to support the form work for the first 9-inch-thick lift and the first lift is then designed to support the remaining 15-inch lift with the form work removed. Delays are specified between adjacent lift pours in order to minimize the effects of initial volume changes. The second lift was not placed until the first lift had attained its specified strength.

The base slab, ring beam, and parapet wall were constructed using conventional methods.

3.8.1.7 Testing and Inservice Surveillance Requirements

Since the Shield Building is not a pressure containment, its wall and dome will not be pressure tested.

REFERENCES

None

TABLE 3.8.1-1 (SHEET 1 of 2)

LOADING COMBINATIONS, LOAD FACTORS AND ALLOWABLE STRESSES
FOR THE SHIELD BUILDING CONCRETE EXTERIOR CYLINDRICAL WALL, DOME AND BASE SLAB

COMBINATIONS(3)

LOADING		**1**	**2**	**2a**	**2b**	**3**	**4**	**5**	**6**	**7**	**8**	**9**
D DEAD LOAD		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	10
D EARTH PRESSURE		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
PMF PROBABLE MAXIMUM FLOOD												1.0
To NORMAL OPERATING TEMPERATURE		1.0	1.0	1.0	1.0	1.0	1.0					
Ta ACCIDENT TEMPERATURE								1.0	1.0	1.0	1.0	
Pa ACCIDENT PRESSURE								1.5	1.25	1.25	1.0	
Fegs SAFE SHUTDOWN EARTHQUAKE							1.0				1.0	
Fego OPERATIONAL BASIS EARTHQUAKE			1.0		1.0				1.25			
W NORMAL WIND				1.0	1.0					1.25		1.0
Wt TORNADO(2)						1.0						
L LIVE LOAD		1.0	1.0			1.0	1.0	1.0	1.0	1.0	1.0	1.0
CC CONSTRUCTION CONDITION		1.0										
Pv NEGATIVE INTERNAL PRESSURE			1.0	1.0			1.0					
YjYr PIPE BREAK JET AND REACTION LOAD									1.0	1.0	1.0	
ALLOWABLE STRESSES*	fc	.45fc'	.45fc'	.45fc'	.45fc'	.75fc'	.75fc'	.75fc'	.75fc'	.75fc'	.75fc'	.75fc'
	fs	.5 fy(1)	.5fy(1)	.5 fy(1)	.5fy(1)	.9 fy	.9 fy	.9 fy	.9 fy	.9 fy	.9 fy	.9 fy

*fc' = SPECIFIED STRENGTH OF CONCRETE
fc = ALLOWABLE FLEXURAL CONCRETE STRESS
fy = YIELD STRENGTH OF REINFORCING STEEL
fs = ALLOWABLE REINFORCING STEEL STRESS

FOOTNOTES:

(1) REINFORCING STEEL STRESSES MAY BE INCREASED BY 33% WHEN TEMPERATURE EFFECTS ARE COMBINED PROVIDED THE REQUIRED SECTION IS NOT REDUCED FROM THAT REQUIRED WITHOUT THE TEMPERATURE EFFECTS

(2) Wt INCLUDES TORNADO WIND, TORNADO POSITIVE INTERNAL PRESSURE, AND TORNADO GENERATED MISSILES.

TABLE 3.8.1-1 (SHEET 2 of 2)

FOOTNOTES (Continued):

(3) LOADING COMBINATIONS (COMPARED TO TABLE CC-3200-1 OF ACI-359, 1973)

1. Service - Construction
2. Service - Normal
3. Factored - Extreme
4. Factored - Environmental
5. Factored - Abnormal
6. Factored - Abnormal/Severe Environmental
7. Factored - Abnormal/Severe Environmental
8. Factored - Abnormal/Extreme Environmental
9. Factored - Extra Case

The following loads from Table CC-3200-1 of ACI-359, 1973, as issued for trial use, are not applicable to the Shield Building exterior wall and dome.
(F, Pt, Tt, Ro, Ra, Yr, Yj, Ym, Pa, Ta) = 0
The Structural Integrity Test ($D + L + P_t + T_t$) from the ACI-359, 1973 is not a controlling load case for the base slab.

TABLE 3.8.1-2 (SHEET 1 of 2)

SHIELD BUILDING EQUIPMENT HATCH DOORS AND SLEEVE
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Structural

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression****	Shear
I	Dead load plus 2-psi pressure	0.50 F_y	0.47 F_y	0.33 F_y
II	Dead load plus 3-psi pressure inside	0.90 F_y	0.90 F_y	0.60 F_y
III	Dead load plus 2-psi pressure outside plus *OBE	0.60 F_y	0.60 F_y	0.40 F_y
IV	Dead load plus 2-psi pressure outside plus *SSE	0.90 F_y	0.90 F_y	0.60 F_y
**V	Dead load plus *OBE	0.60 F_y	0.60 F_y	0.40 F_y
**VI	Dead load plus *SSE	0.90 F_y	0.90 F_y	0.60 F_y

Mechanical

No.	Load Combinations	Allowable Stresses (psi) Tension & Compression(****)	Shear
**I	Dead load	$\frac{\text{Ult}}{5}$	$\frac{2 \times \text{Ult}}{15}$
***Ia	Dead load plus* OBE	0.60 F_y	0.40 F_y
***II	Dead load plus *SSE	0.90 F_y	0.60 F_y

TABLE 3.8.1-2 (Sheet 2 of 2)

SHIELD BUILDING EQUIPMENT HATCH DOORS AND SLEEVE
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

III	Dead load plus 2-psi pressure outside	$\frac{\text{Ult}}{5}$	$\frac{2 \times \text{Ult}}{15}$
IV	Dead load plus 3-psi pressure inside	0.90 F_y	0.60 F_y
V	Dead load plus 2-psi pressure outside plus *OBE	0.60 F_y	0.40 F_y
VI	Dead load plus 2-psi pressure outside plus *SSE	0.90 F_y	0.60 F_y

* Acts in one horizontal direction only at any given time and acts in the vertical and horizontal directions simultaneously.

** Door open.

*** For hinges only with doors open.

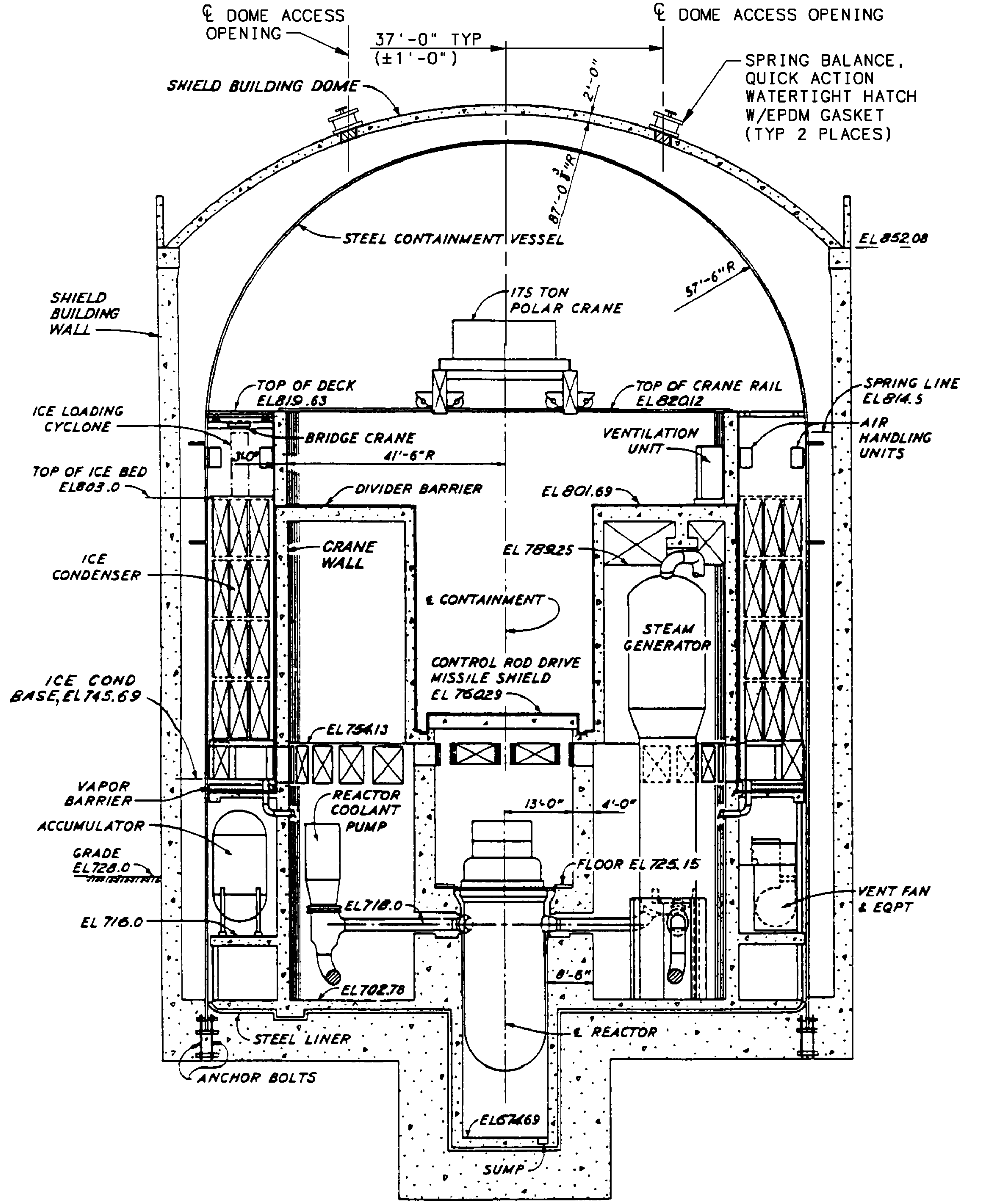
**** The value indicated for allowable compression stress is the maximum value permitted when buckling does not control. The critical buckling stress, Fcr, shall be used in place of Fy when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\,C_c^{\,2}} \right] \text{when } \frac{K1}{r} \leq C_c$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2 0} \text{ when } \frac{Kl}{r} > C_c$$

CAD MAINTAINED DRAWING



REACTOR BUILDING ELEVATION

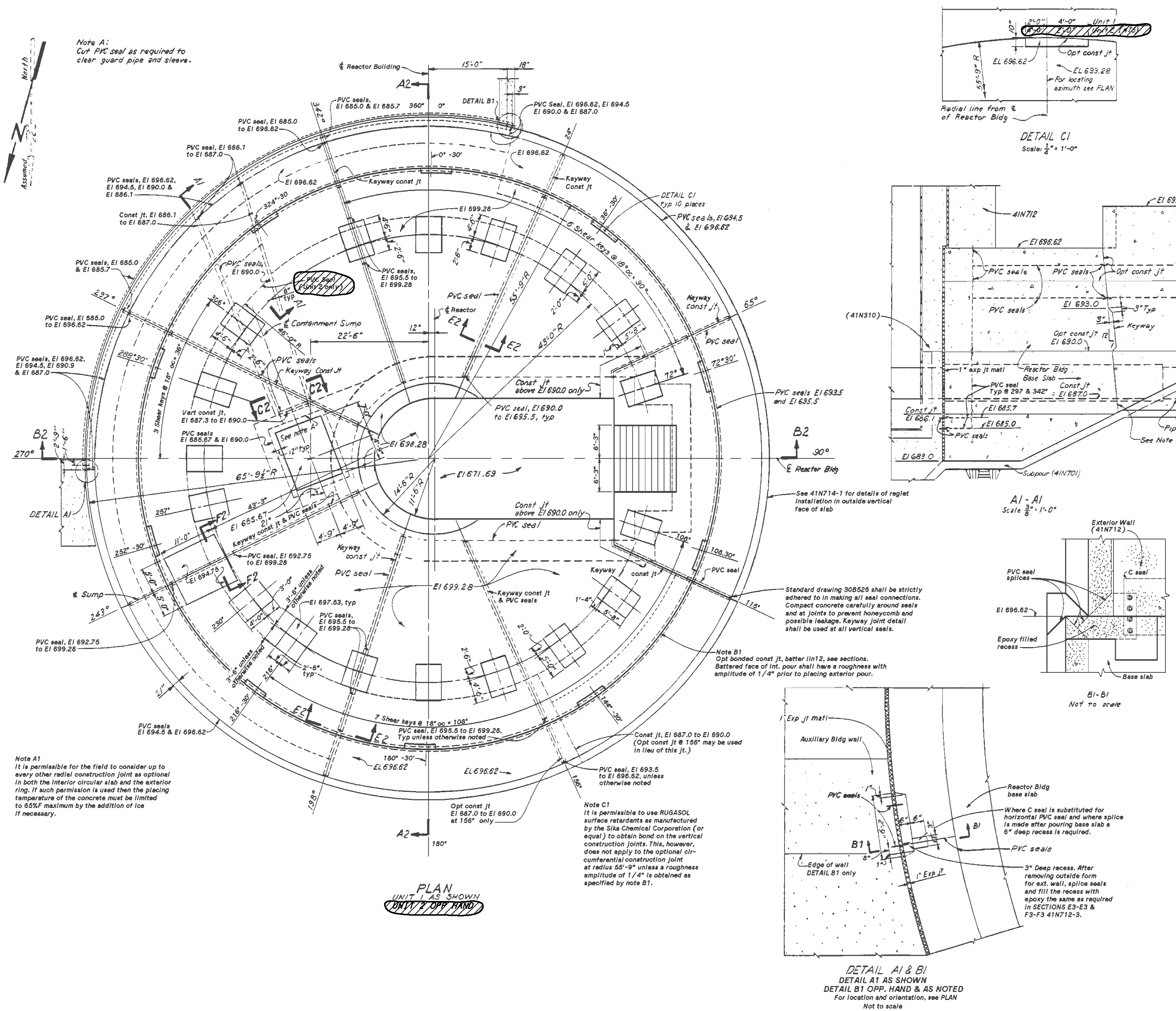
1 ft = 0.3048 m

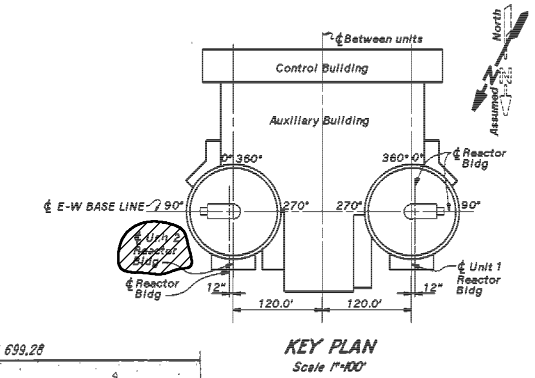
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT

REACTOR BUILDING ELEVATION

FIGURE 3.8.1-1

CAD MAINTAINED DRAWING





△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

NOTES:

1. For general notes, see 41N700-1.
2. △ Concrete shall be class 501.5 B FW, except below the construction joint at El 685.67 in the cavity where the concrete shall be class 301.5 A FW.
3. The minimum lenght of time between adjacent pours shall be 7 days. See note 14.
4. Dimensions shown are rough concrete. All surfaces except the vertical exterior face are construction joints and shall be roughened to insure bond. See note C1.
5. C seals may be substituted for PVC seals in horizontal construction joints.
6. For embedded pipes, see Mechanical drawings.
7. For embedded anchor bolts, see 48N401.
8. △ For limits of fill concrete to embed the containment liner frame, see 48N401. The fill conc shall be class 300.75 A FW. All horizontal surfaces of the fill concrete shall be screeded to insure a uniform bearing surface for containment liner. The fill concrete shall be poured before installation of containment liner.
9. Do not chamfer exposed edges.
10. Reactor building is a Class I structure, and quality assurance is required.
11. The containment vessel anchor bolts shall be pretensioned prior to the concrete attaining a compressive strenght of 5000 psi.
12. At the option of the field the structural steel liner assemblies for the Containment Sump, Containment Floor Drain Sump and Shear Keys may be placed prior to pouring the structural slab thus using them as forms in lieu of forming the recesses shown and inserting the assemblies later.
13. See note A1 this sheet.
14. If placing temperature of 65°F is used, the lenght of time between pours A1 & A2 and A2 & A3 may be shortened to 3 days at option of field. Pour designations are as shown on 41N10060-1, -11 R0.
15. See calculation WCG-1-1370 for engineering review and assessment of the Reactor Building concrete features.
16. Inleakage of ground water into the Auxiliary Building in the vicinity of the 0° azimuth shall be repaired in accordance with TVA specification "Specification Requirements for Sealing Water Leaks in Concrete Walls and Slabs" in contract P-95N3D-121900.

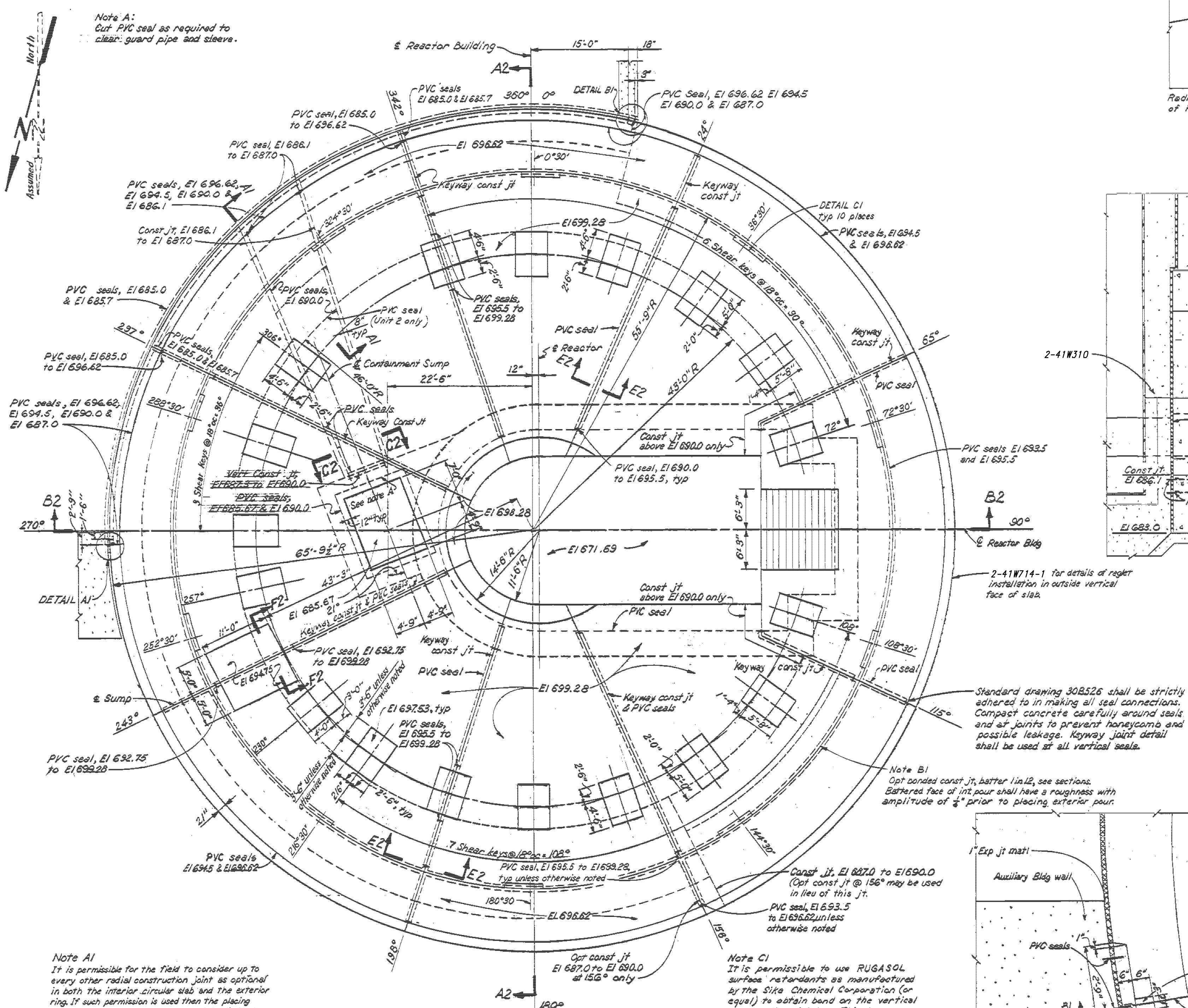
REFERENCE DRAWINGS:
41BM707----BILL OF MATERIAL

Scale 1/8" = 1'-0"
Except as noted

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
CONCRETE
STRUCTURAL SLAB
EL.699.28 OUTLINE
TVA DWG NO. 41N707-1 RE
FIGURE 3.8.1-2

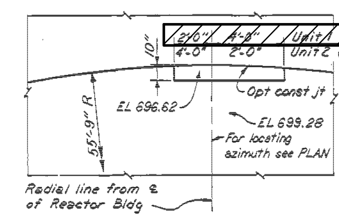
CAD MAINTAINED DRAWING



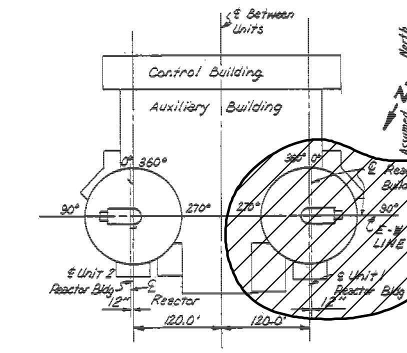
Note A1
It is permissible for the field to consider up to every other radial construction joint as optional in both the interior circular slab and the exterior ring. If such permission is used then the placing temperature of the concrete must be limited to 65°F maximum by the addition of ice if necessary.

Note C1
It is permissible to use RUGASOL surface retardants as manufactured by the Sika Chemical Corporation (or equal) to obtain bond on the vertical construction joints. This, however, does not apply to the optional circumferential construction joint at radius 55'-9" unless a roughness amplitude of 1/4" is obtained as specified by Note B1.

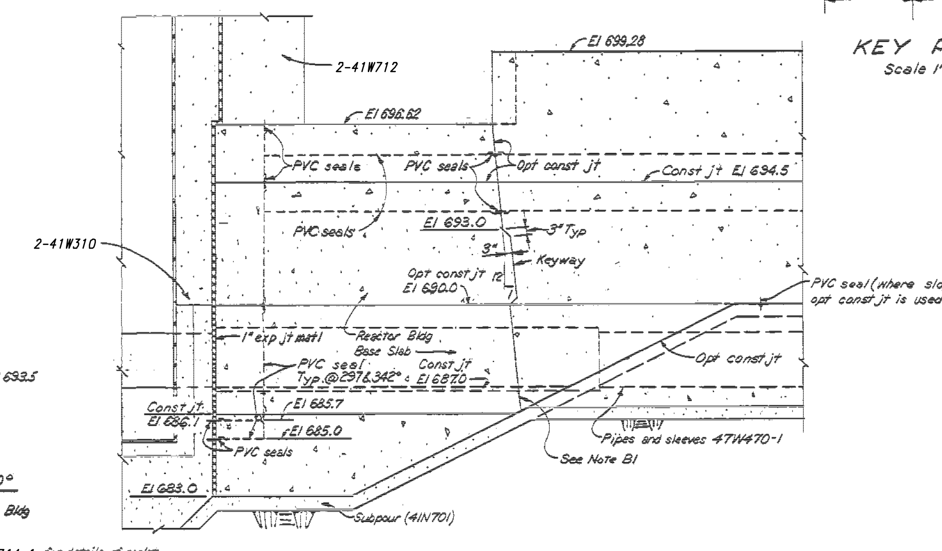
PLAN
~~UNIT 1 AS SHOWN~~
UNIT 2 OPP HAND



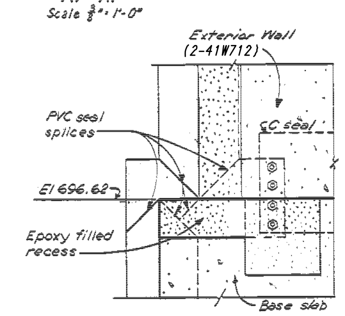
DETAIL C1
Scale 3/4"=1'-0"



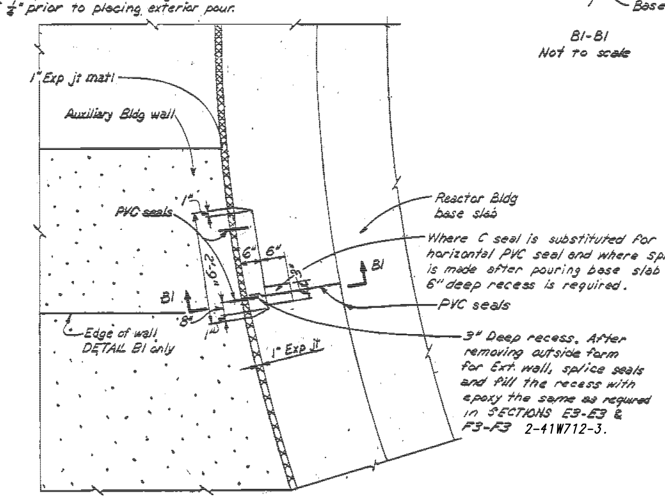
KEY PLAN
Scale 1"=100'



A1 - A1
Scale 3/8"=1'-0"



B1-B1
Not to scale



DETAIL A1 & B1
DETAIL A1 AS SHOWN
DETAIL B1 OPP HAND & AS NOTED
For location and orientation, see PLAN
Not to scale

NOTES:
1. For general notes, see 41N700-1.
△ 2. Concrete shall be class 501.5 B FW, except below the construction joint at EL 685.67 in the cavity where the concrete shall be class 301.5 A FW.
3. The minimum length of time between adjacent pours shall be 7 days. See note 14.
4. Dimensions shown are to rough concrete. All surfaces except the vertical exterior face are construction joints and shall be roughened to insure bond. See Note C1.
5. C seals may be substituted for PVC seals in horizontal construction joints.
6. For embedded pipes, see Mechanical drawings.
7. For embedded anchor bolts, see 48N401.
△ 8. For limits of fill concrete to embed the containment liner frame, see 48N401. The fill conc shall be class 300.75 A FW. All horizontal surfaces of the fill concrete shall be screeded to insure a uniform bearing surface for containment liner. The fill concrete shall be poured before installation of the containment liner.
9. Do not chamfer exposed edges.
10. Reactor building is a Class I structure, and quality assurance is required.
11. The containment vessel anchor bolts shall not be pretensioned prior to the concrete attaining a compressive strength of 5000 psi.
12. At the option of the field the structural steel liner assemblies for the Containment Sump, Containment Floor Drain Sump and Shear Keys may be placed prior to pouring the structural slab thus using them as forms in lieu of forming the recesses shown and inserting the assemblies later.
13. See Note A1 this sheet.
14. If placing temperature of 65°F is used, the length of time between pours A1 & A2 and A2 & A3 may be shortened to 3 days at option of field. Pour designations are as shown on 41N10060-1, -11 R0.

FOR HATCHED AREAS
SEE AC DWG 41N707-1

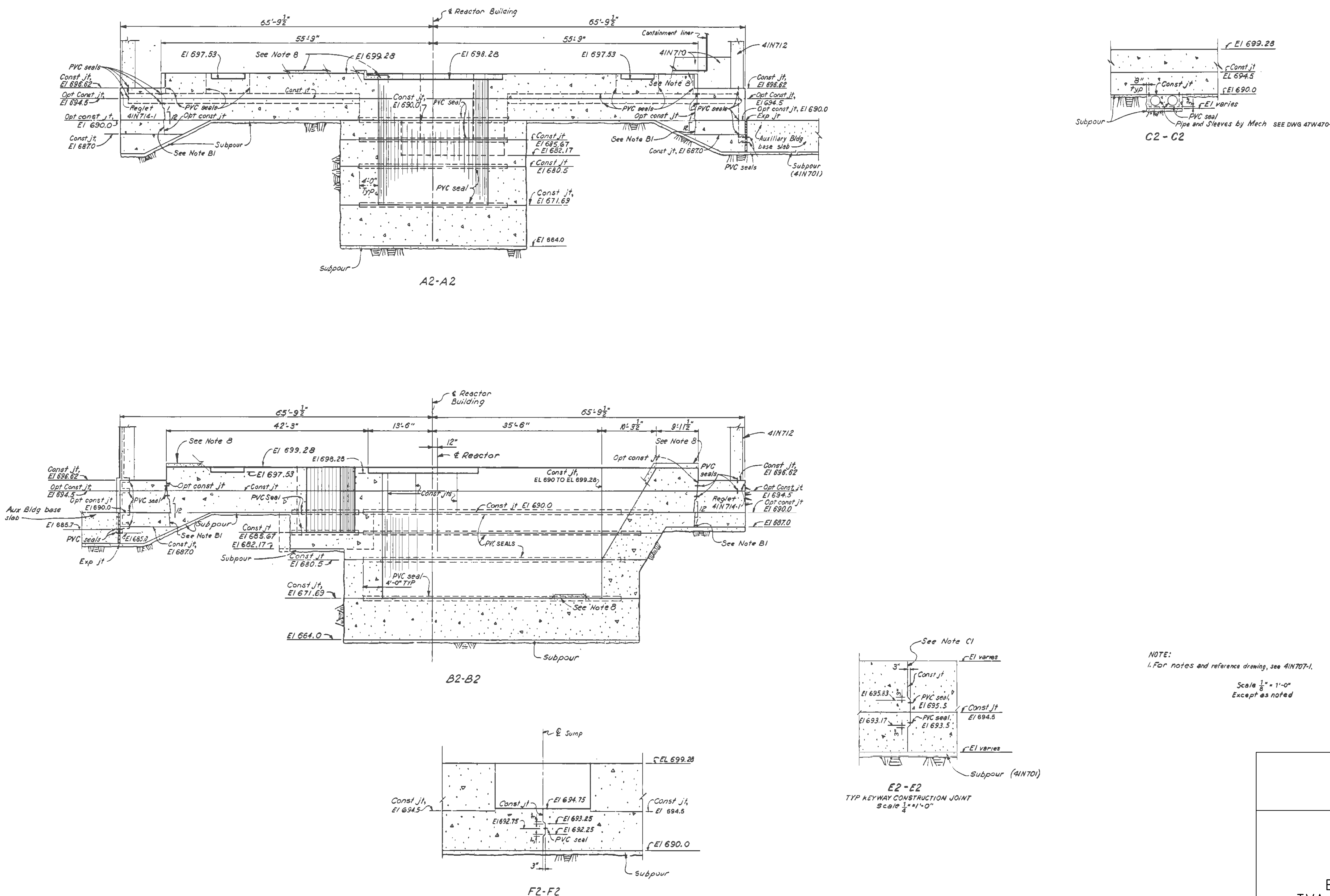
△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

COMPANION DWG: 2-41N707-2

REFERENCE DRAWINGS:
41BM707___ BILL OF MATERIAL

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
CONCRETE
STRUCTURAL SLAB
EL.699.28 OUTLINE
TVA DWG NO. 2-41N707-1 R0
FIGURE 3.8.1-2(U2)



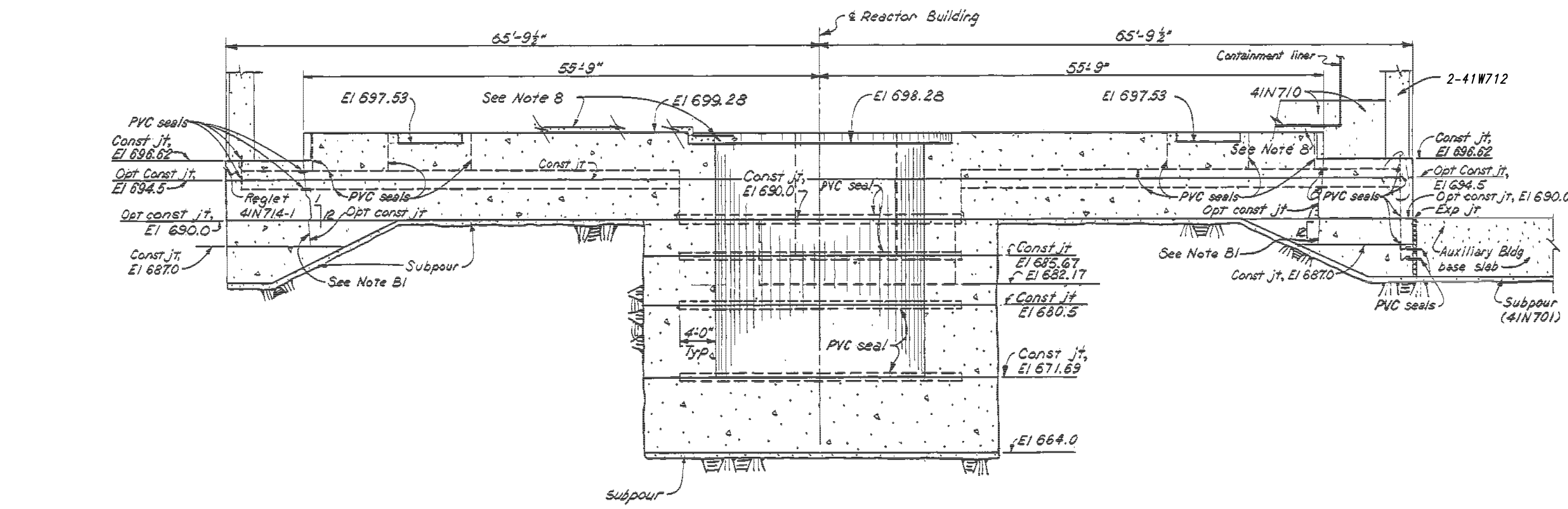
NOTE:
1. For notes and reference drawing, see 41N707-1.

Scale 1/8" = 1'-0"
Except as noted

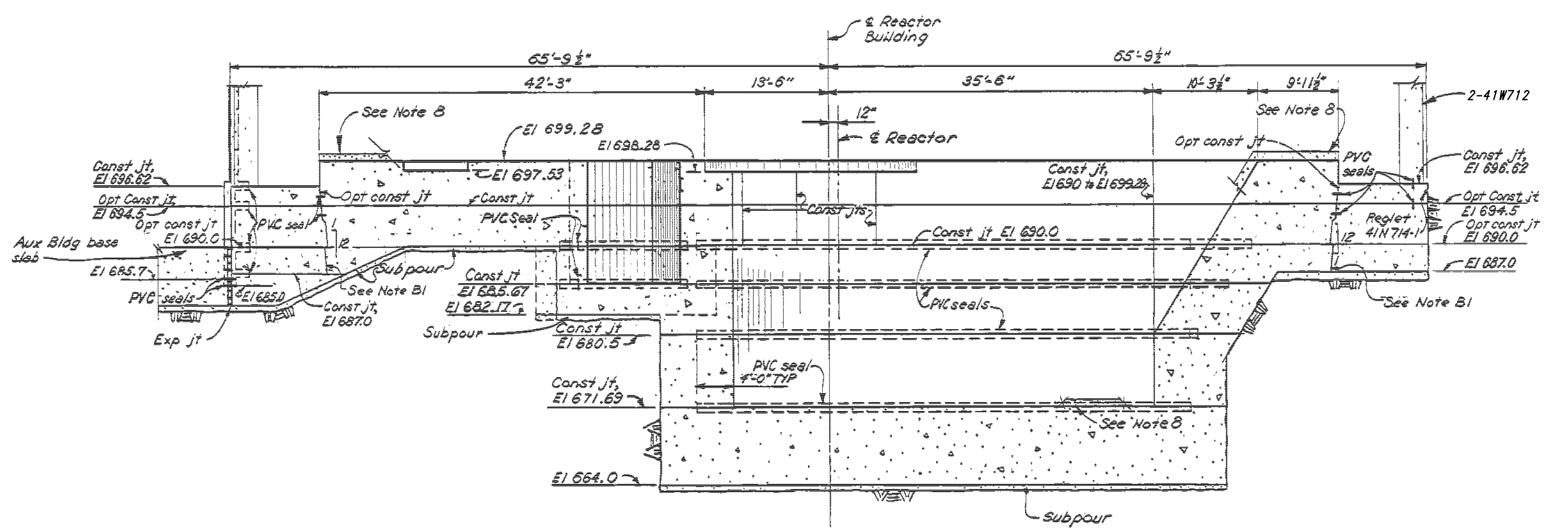
COMPANION DRAWING: 41N707-1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

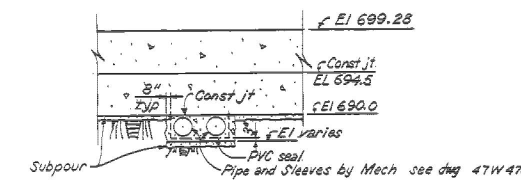
REACTOR BUILDING
UNIT 1
CONCRETE
STRUCTURAL SLAB
EL.699.28 OUTLINE
TVA DWG NO. 41N707-2 RC
FIGURE 3.8.1-3



A2-A2

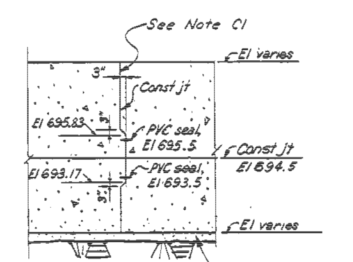


B2-B2

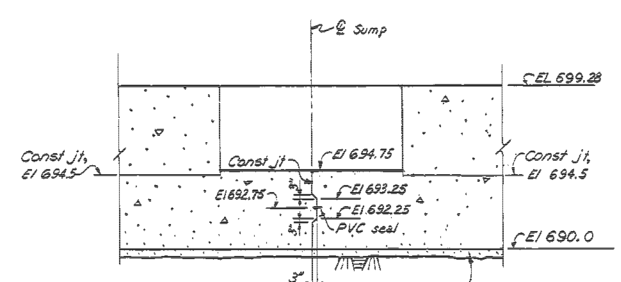


C2-C2

NOTE:
FOR NOTES AND REFERENCE DRAWING, SEE 2-41N707-1.



E2-E2
TYP KEYWAY CONSTRUCTION JOINT
Scale ¼"=1'-0"



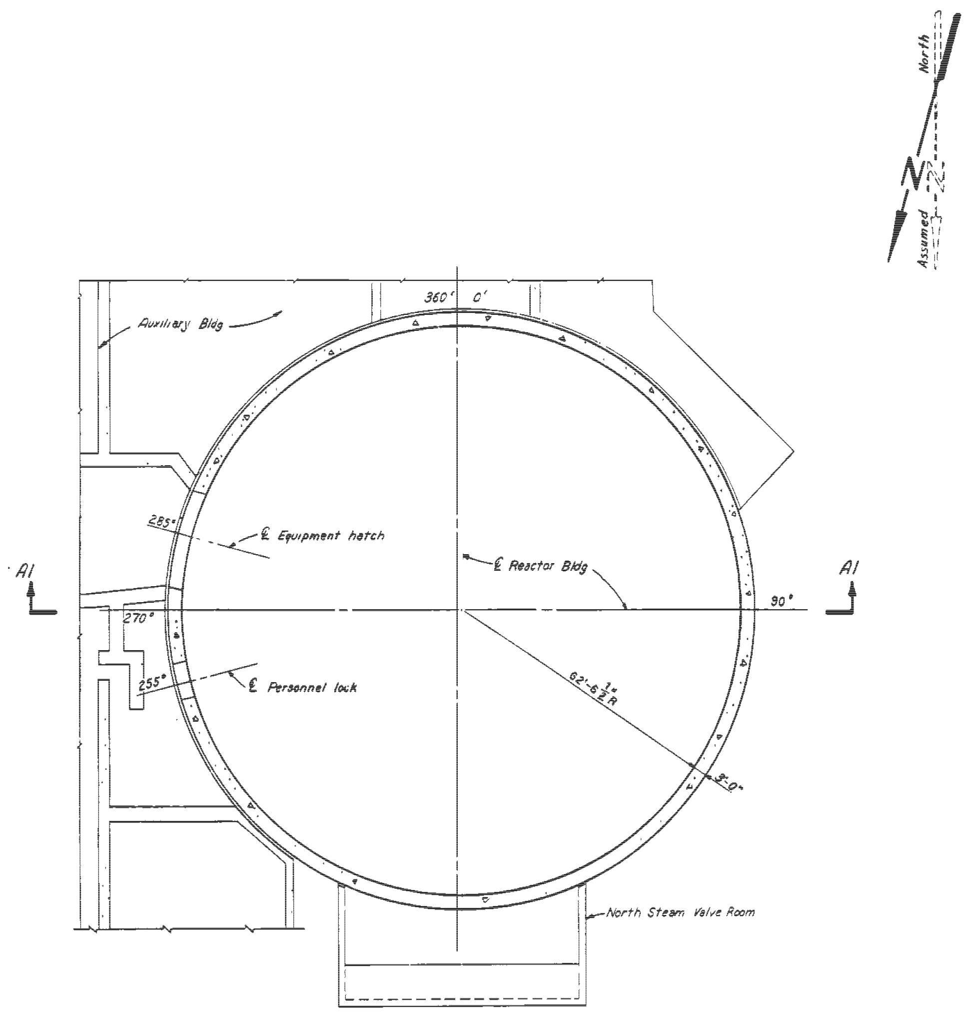
F2-F2
Scale ¾"=1'-0"

COMPANION DWG: 2-41N707-1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

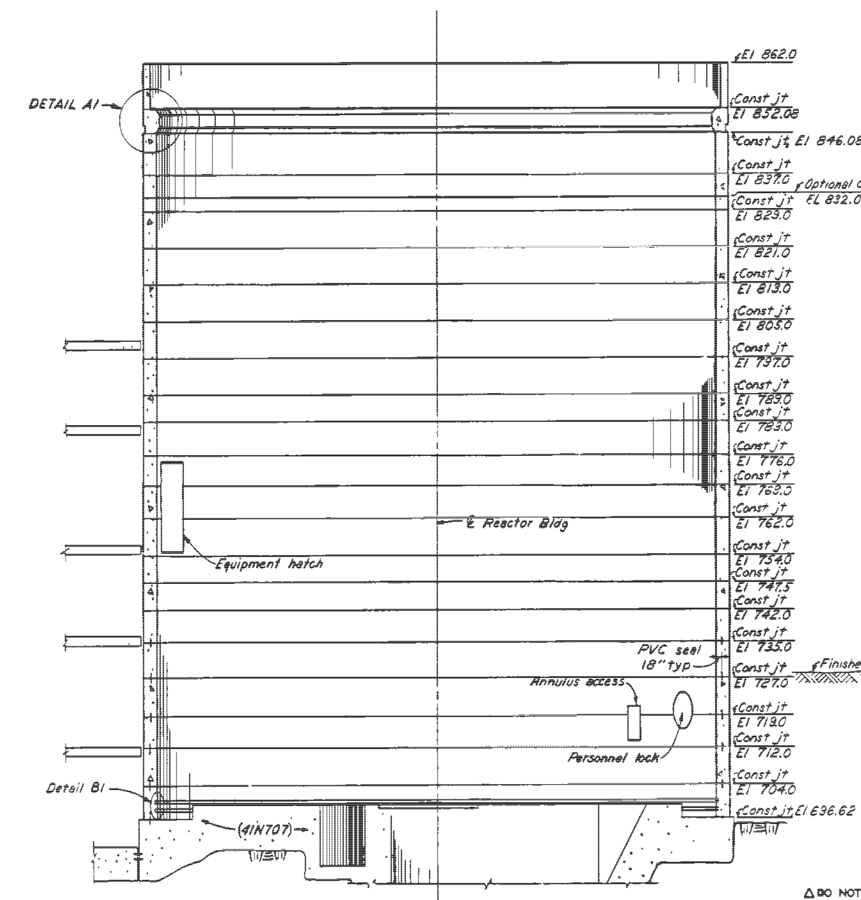
REACTOR BUILDING
UNIT 2
CONCRETE
STRUCTURAL SLAB
EL.699.28 OUTLINE
TVA DWG NO. 2-41N707-2 R0
FIGURE 3.8.1-3(U2)

CAD MAINTAINED DRAWING

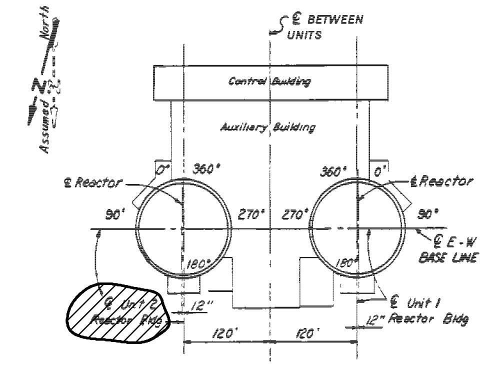


SECTIONAL PLAN - EL 757.0

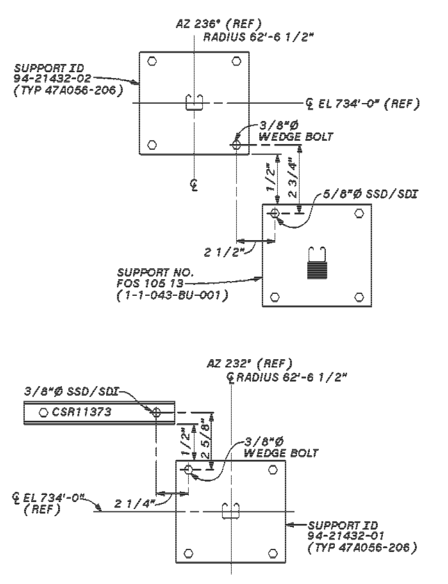
UNIT 1 AS SHOWN
UNIT 2 OPP HAND

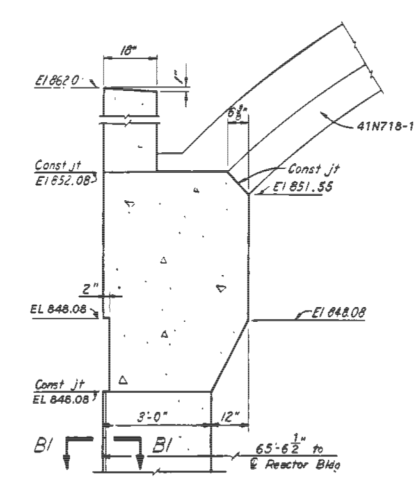


A1 - A1

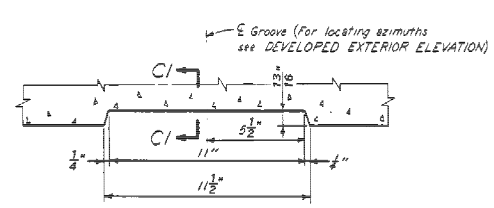


KEY PLAN
Scale 1" = 100'

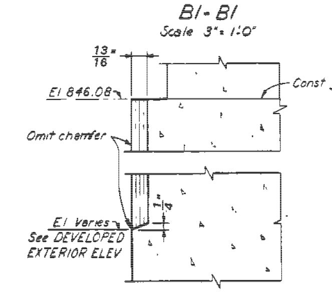




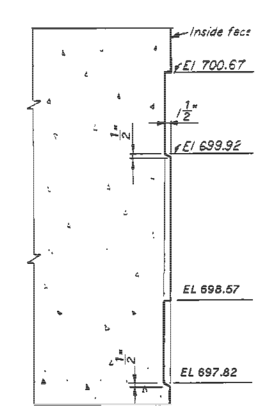
DETAIL A1
RING BEAM
Scale 1/2" = 1'-0"



B1 - B1
Scale 3" = 1'-0"



C1 - C1
Scale 3" = 1'-0"



DETAIL B1
SHEAR KEYS
Scale 1 1/2" = 1'-0"

△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

Note A:
The top 3' of the opening may be poured monolithically with the rest of the concrete if at least a 3' head is maintained above the top of the opening while pouring, or may be grouted separately in accordance with TVA General Construction Specification G51.

NOTE:
SEE CALCULATION WCG-1-1370 FOR ENGINEERING REVIEW AND ASSESSMENT OF THE REACTOR BUILDING CONCRETE FEATURES.

NOTES:
1. The Reactor Building is a Class I structure, and quality assurance is required.
△ 2. Concrete shall be class 4015A FW in accordance with TVA Construction Specification G2.
3. Epoxy adhesive mortar shall consist of Colma-Dur Gel or Sika Dur Gel and Colma Quartzite aggregate mixed in equal parts by volume. For proper adhesion all surfaces must be clean and dry when using this mortar.
4. Chamfer all exposed edges 3/4" unless otherwise noted.
5. For embedded parts, pipes and conduits, see Structural Steel, Mechanical and Electrical drawings
6. PVC seals shall be placed in all construction joints below El 740.0.
7. All dimensions given in the DEVELOPED EXTERIOR ELEVATION are chord dimensions between points at the outside circumference of the wall.
8. For embedded guy wire anchors, see Chicago Bridge and Iron Co drawing.
9. When concreting temporary construction openings, the top 3" shall be filled with concrete mortar. The mortar shall be hand tamped to ensure a uniform bearing surface for the wall above. The mortar shall consist of 1 part cement, 2 1/2 parts sand, and water not to exceed 5 gal. per sack of cement. As a substitute for concrete mortar a non-shrink non-metallic grout may be used in accordance with the manufacturers recommendations. For alternate method see Note A.
10. For Waterproofing details see dwg 41N714-1.

REFERENCE DRAWING:
41BM712 - - - - BILL OF MATERIAL

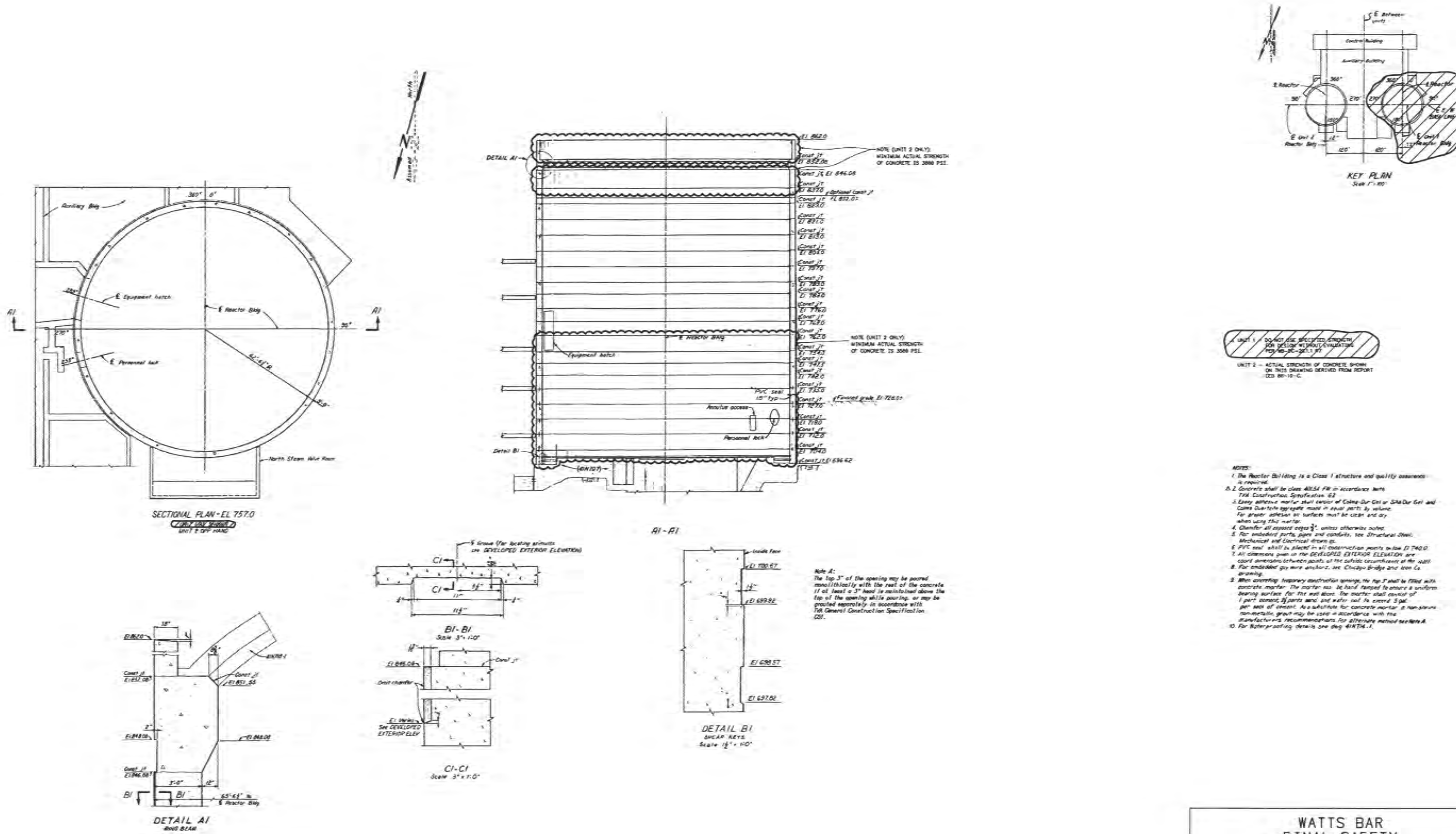
Scale 1/16" = 1'-0"
Except as noted

COMPANION DRAWINGS: 41N712-2 & 3

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
CONCRETE
EXTERIOR WALL
OUTLINE
TVA DWG NO. 41N712-1 RE
FIGURE 3.8.1-4

CAD MAINTAINED DRAWING



NOTES:
1. The Reactor Building is a Class 1 structure and quality assurance is required.
2. Concrete shall be class 4054 FW in accordance with TVA Construction Specification G2.
3. Epoxy adhesive mortar shall consist of Colma-Dur Gel or Sika-Dur Gel and Colma Quartzite aggregate mixed in equal parts by volume. For proper adhesion all surfaces must be clean and dry when using this mortar.
4. Chamfer all exposed edges ¾", unless otherwise noted.
5. For embedded parts, pipes and conduits, see Structural Steel, Mechanical and Electrical drawings.
6. PVC seal shall be placed in all construction joints below El 740.0
7. All dimensions given in the DEVELOPED EXTERIOR ELEVATION are chord dimensions between points at the outside circumference of the wall.
8. For embedded guy wire anchors, see Chicago Bridge and Iron Co. drawing.
9. When concreting temporary construction openings, the top 3" shall be filled with concrete mortar. The mortar etc. be hand tamped to ensure a uniform bearing surface for the wall above. The mortar shall consist of 1 part cement, 2½ parts sand and water not to exceed 5 gal. per sack of cement. As a substitute for concrete mortar a non-shrink non-metallic grout may be used in accordance with the manufacturers recommendations for alternate method see Note A.
10. For Waterproofing details see dwg 41N714-1.

Note A:
The top 3" of the opening may be poured monolithically with the rest of the concrete if at least a 3" head is maintained above the top of the opening while pouring, or may be grouted separately in accordance with TVA General Construction Specification G31.

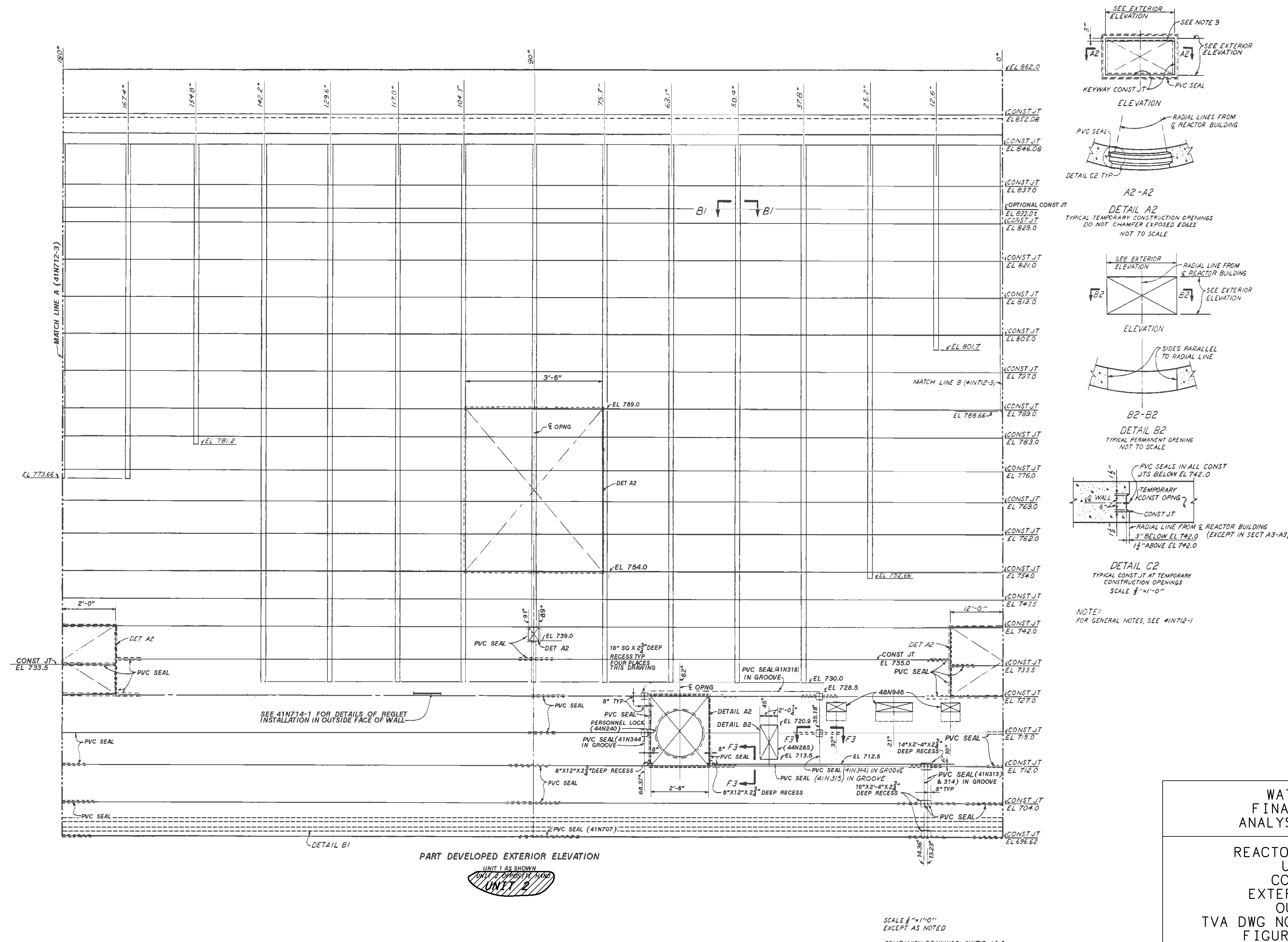
REFERENCE DRAWING:
41N712 ..... BILL OF MATERIAL
41N1868-12 ..... CONCRETE POUR DRAWING

FOR HATCHED AREAS SEE UNIT 1 DRAWING 41N712-1

COMPANION DWG: 2-41N712-2 & 3

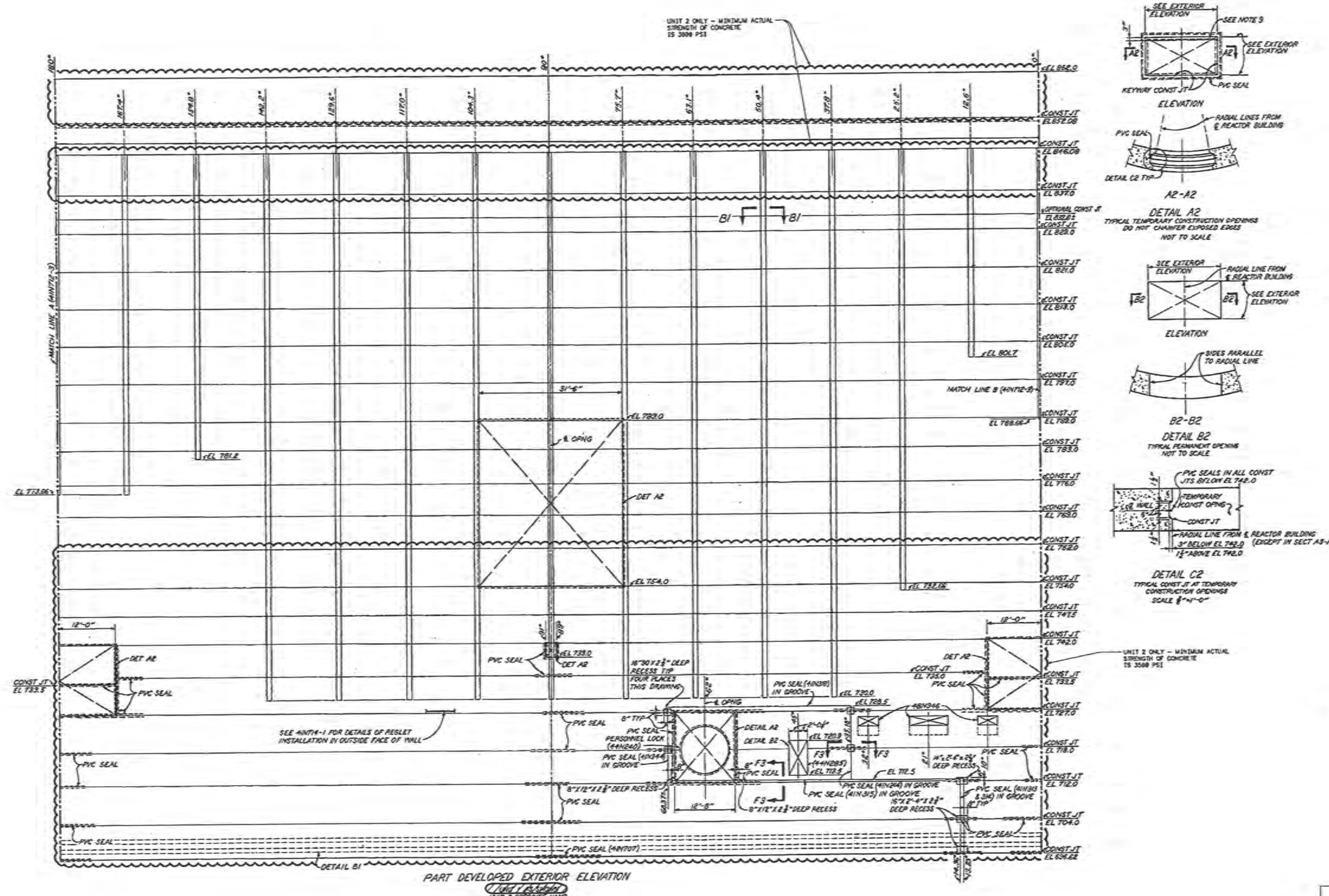
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
CONCRETE
EXTERIOR WALL
OUTLINE
TVA DWG NO. 2-41N712-1 R0
FIGURE 3.8.1-4(U2)



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
CONCRETE
EXTERIOR WALL
OUTLINE
TVA DWG NO. 41N712-2 RC
FIGURE 3.8.1-5



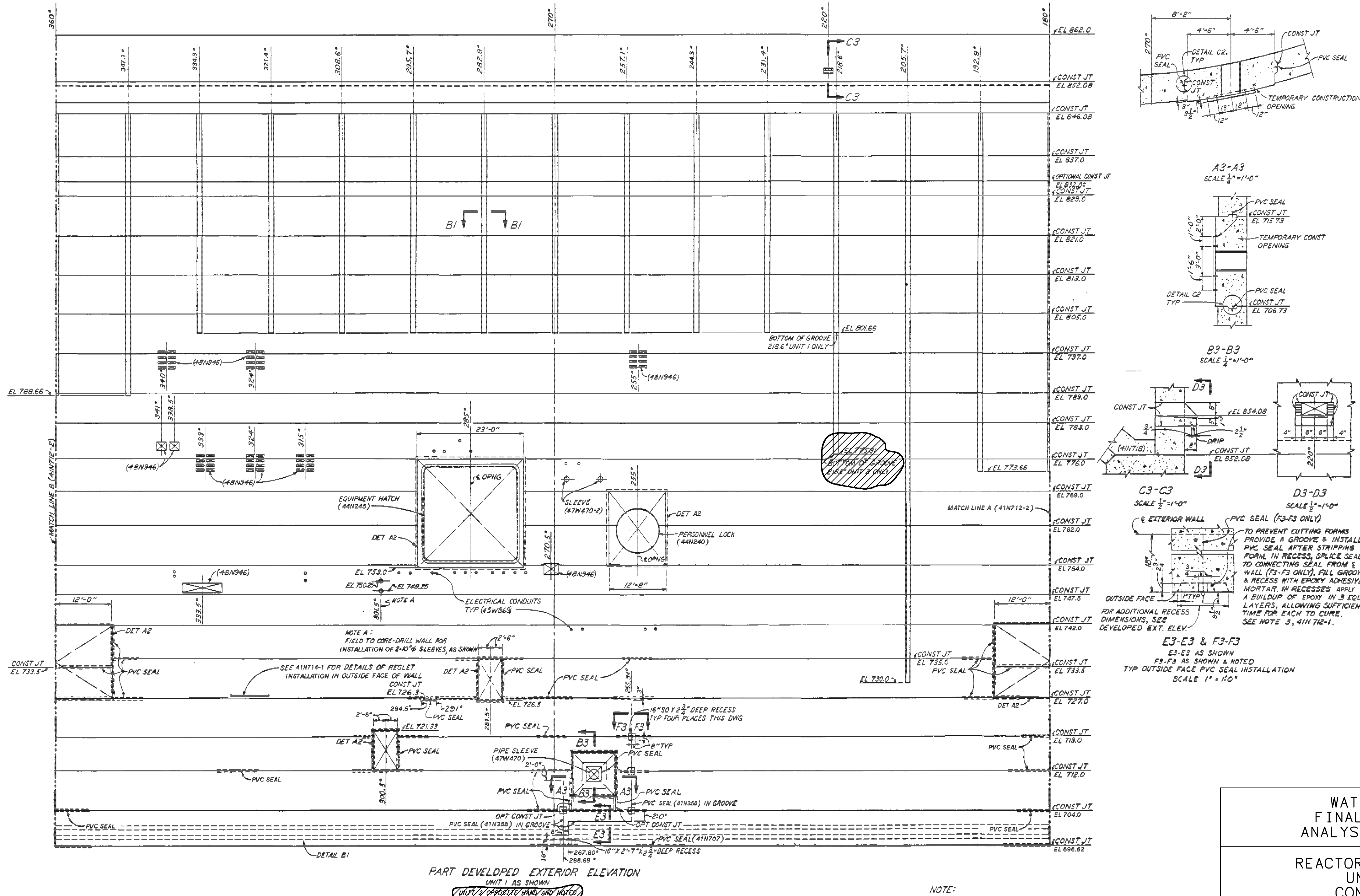
FOR HATCHED AREAS SEE UNIT 1 DRAWING 41N712-2

COMPANION DWG: 2-41N712-1 & 3

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
CONCRETE
EXTERIOR WALL
OUTLINE
TVA DWG NO. 2-41N712-2 RO
FIGURE 3.8.1-5(U2)

CAD MAINTAINED DRAWING



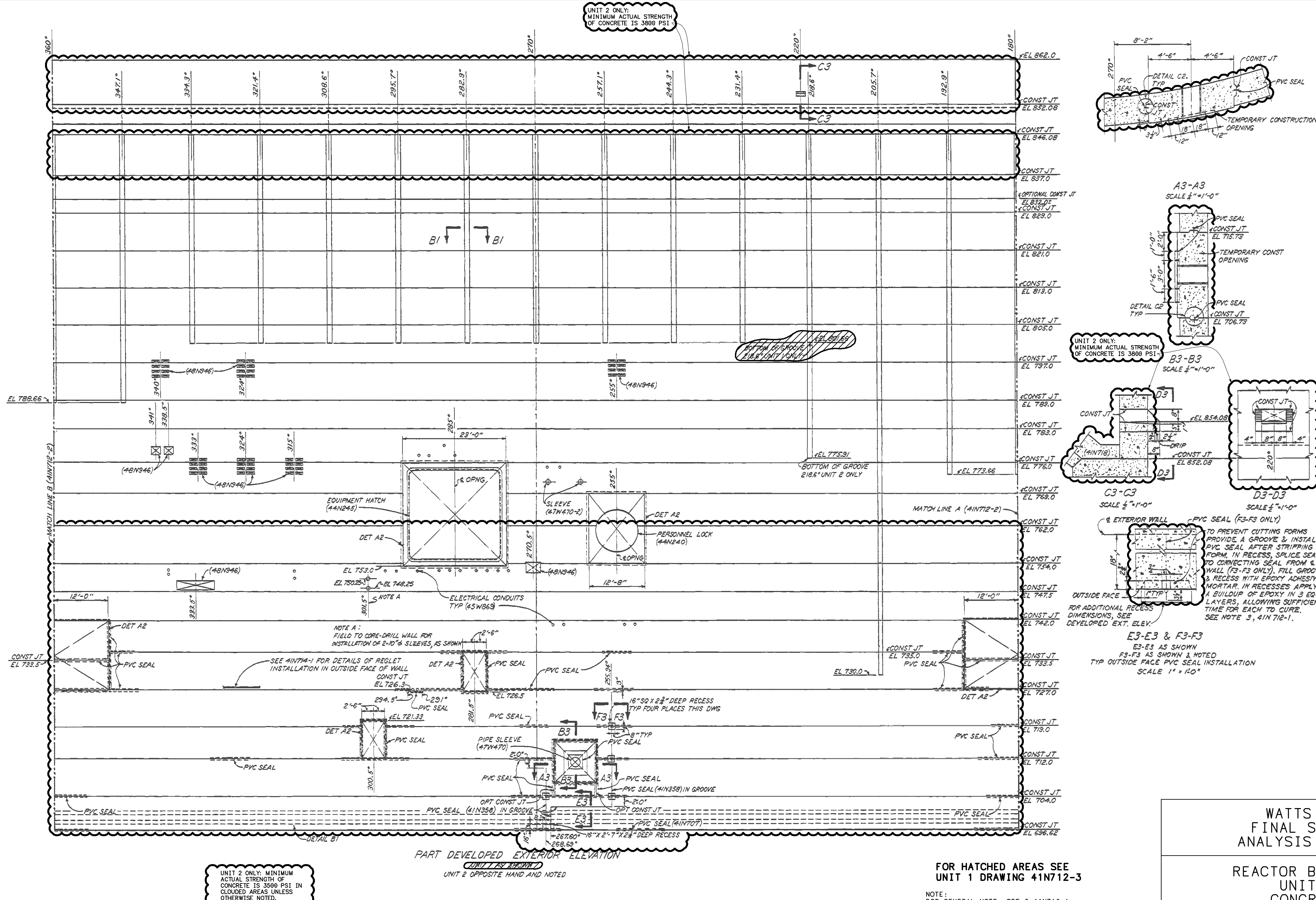
NOTE:
FOR GENERAL NOTE, SEE 41N712-1

SCALE 1/8" = 1'-0"
EXCEPT AS NOTED

COMPANION DRAWINGS: 41N712-1&2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
CONCRETE
EXTERIOR WALL
OUTLINE
TVA DWG NO. 41N712-3 RD
FIGURE 3.8.1-6



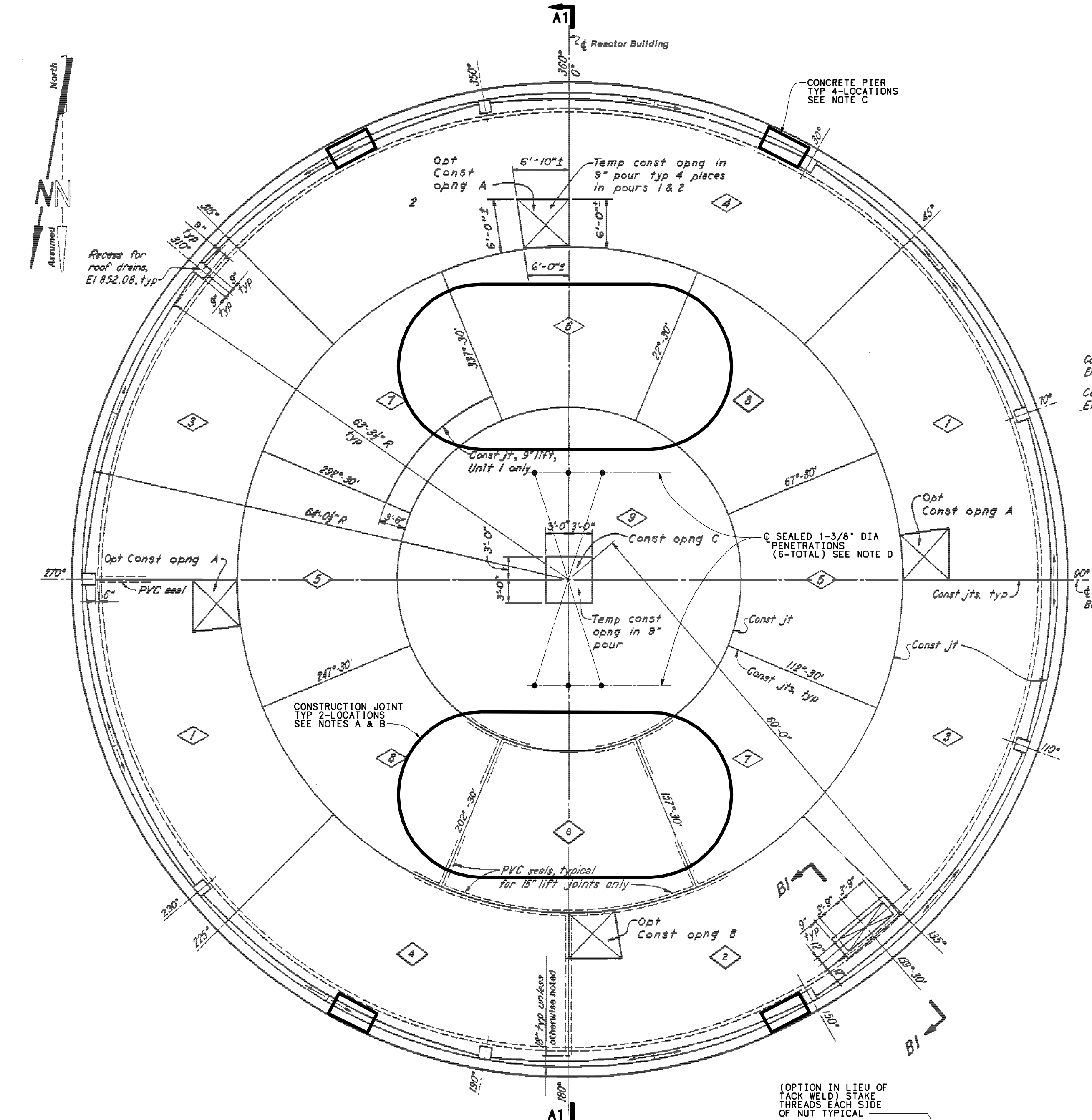
FOR HATCHED AREAS SEE UNIT 1 DRAWING 41N712-3

NOTE:
FOR GENERAL NOTE, SEE 2-41N712-1

COMPANION DWG: 2-41N712-1 & -2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

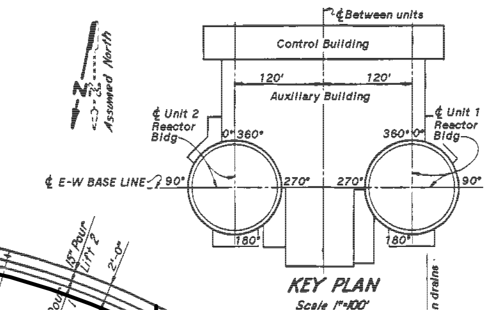
REACTOR BUILDING
UNIT 2
CONCRETE
EXTERIOR WALL
OUTLINE
TVA DWG NO. 2-41N712-3 R0
FIGURE 3.8.1-6(U2)



PLAN

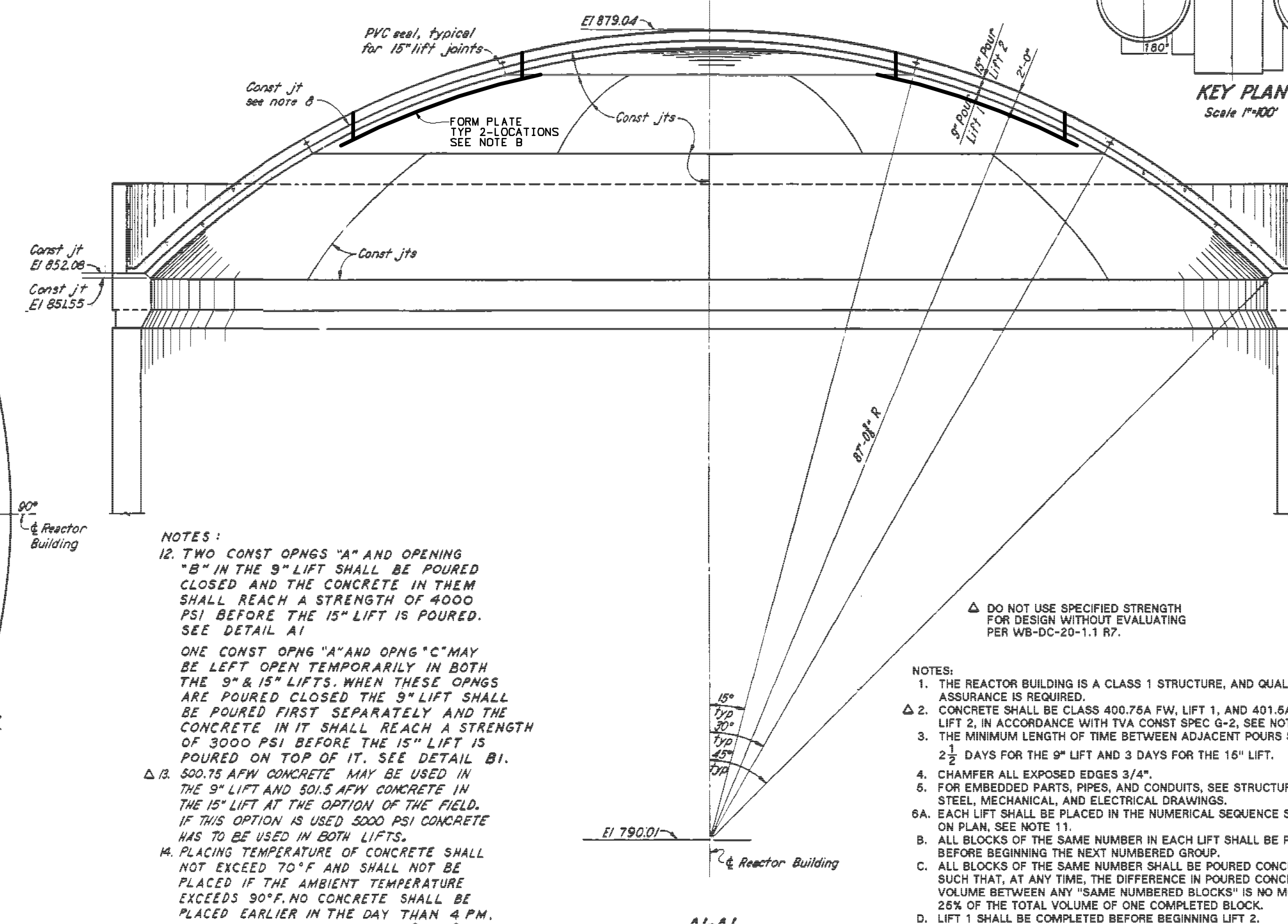
UNIT 1 AS SHOWN

Const jts shown are typical for both 1st & 2nd lifts.



KEY PLAN

Scale 1"=100'



A1-A1

NOTES:

12. TWO CONST OPNGS "A" AND OPENING "B" IN THE 9" LIFT SHALL BE POURED CLOSED AND THE CONCRETE IN THEM SHALL REACH A STRENGTH OF 4000 PSI BEFORE THE 15" LIFT IS POURED. SEE DETAIL A1

 ONE CONST OPNG "A" AND OPNG "C" MAY BE LEFT OPEN TEMPORARILY IN BOTH THE 9" & 15" LIFTS. WHEN THESE OPNGS ARE POURED CLOSED THE 9" LIFT SHALL BE POURED FIRST SEPARATELY AND THE CONCRETE IN IT SHALL REACH A STRENGTH OF 3000 PSI BEFORE THE 15" LIFT IS POURED ON TOP OF IT. SEE DETAIL B1.

△ 13. 500.75 AFW CONCRETE MAY BE USED IN THE 9" LIFT AND 501.5 AFW CONCRETE IN THE 15" LIFT AT THE OPTION OF THE FIELD. IF THIS OPTION IS USED 5000 PSI CONCRETE HAS TO BE USED IN BOTH LIFTS.

14. PLACING TEMPERATURE OF CONCRETE SHALL NOT EXCEED 70°F AND SHALL NOT BE PLACED IF THE AMBIENT TEMPERATURE EXCEEDS 90°F. NO CONCRETE SHALL BE PLACED EARLIER IN THE DAY THAN 4 PM.

15. CONST JTS BETWEEN POURS 7 9 AND 8 9 ARE OPTIONAL ON 9" LIFT ONLY.

16. SEE CALCULATION WCG-1-1370 FOR ENGINEERING REVIEW AND ASSESSMENT OF THE REACTOR BUILDING CONCRETE FEATURES.

△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

NOTES:

1. THE REACTOR BUILDING IS A CLASS 1 STRUCTURE, AND QUALITY ASSURANCE IS REQUIRED.

△ 2. CONCRETE SHALL BE CLASS 400.75A FW, LIFT 1, AND 401.5A FW, LIFT 2, IN ACCORDANCE WITH TVA CONST SPEC G-2, SEE NOTE 13.

3. THE MINIMUM LENGTH OF TIME BETWEEN ADJACENT POURS SHALL BE $2\frac{1}{2}$ DAYS FOR THE 9" LIFT AND 3 DAYS FOR THE 15" LIFT.

4. CHAMFER ALL EXPOSED EDGES 3/4".

5. FOR EMBEDDED PARTS, PIPES, AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.

6A. EACH LIFT SHALL BE PLACED IN THE NUMERICAL SEQUENCE SHOWN ON PLAN, SEE NOTE 11.

B. ALL BLOCKS OF THE SAME NUMBER IN EACH LIFT SHALL BE PLACED BEFORE BEGINNING THE NEXT NUMBERED GROUP.

C. ALL BLOCKS OF THE SAME NUMBER SHALL BE POURED CONCURRENTLY SUCH THAT, AT ANY TIME, THE DIFFERENCE IN POURED CONCRETE VOLUME BETWEEN ANY "SAME NUMBERED BLOCKS" IS NO MORE THAN 25% OF THE TOTAL VOLUME OF ONE COMPLETED BLOCK.

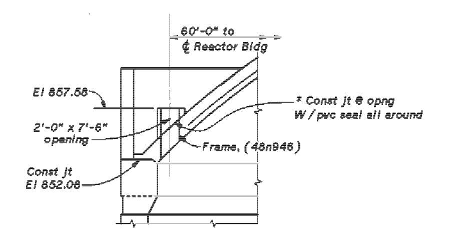
D. LIFT 1 SHALL BE COMPLETED BEFORE BEGINNING LIFT 2.

7. SHORING, INCLUDING THE STEEL TRUSSES (48N910), SHALL BE LEFT IN PLACE UNTIL STRENGHT OF 4000 PSI IS REACHED IN LIFT 1, AND SHALL BE REMOVED BEFORE START OF LIFT 2 CONCRETING.

8. THE TOP SURFACE OF THE FIRST LIFT SHALL BE INTENTIONALLY ROUGHENED TO AN AMPLITUDE OF APPROXIMATELY $\frac{1}{4}$".

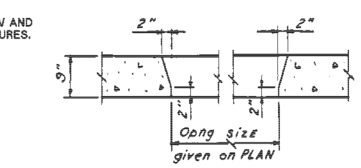
10. AN OPTIONAL VERTICAL CONSTRUCTION JOINT MAY BE PLACED RADIALLY IN BLOCKS 1 THROUGH 4 TO DRIVE EACH BLOCK INTO TWO EQUAL PARTS.

11. POUR SEQUENCE MAY BE ALTERED SUCH THAT ANY NUMBER OR ALL OF 1 AND 2, 3 AND 4, 5 AND 6, 7 AND 8 ARE INTERCHANGED.



B1-B1

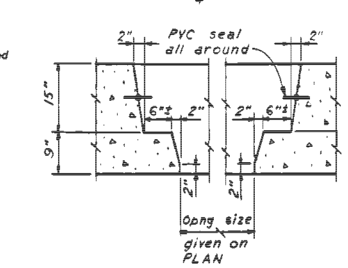
* Field not required to replace that portion of PVC seal removed during repair made to concrete (Unit 1 Only)



DETAIL A1

Typical section through opt temp const opng 9" lift

Scale $\frac{3}{4}$"=1'-0"



DETAIL B1

Typical section through temp const opng in both 9" and 15" lifts

Scale $\frac{3}{4}$"=1'-0"

Scale: $\frac{1}{8}$"=1'-0"

Except as noted

REFERENCE DRAWINGS:

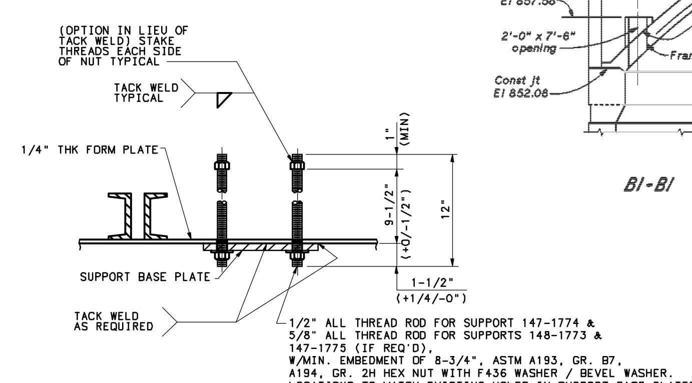
41BM718--- BILL OF MATERIAL

NOTE A:
- REPLACEMENT CONCRETE AND GROUT IS SAFETY RELATED.
- FOR SIZE AND LOCATION OF CONSTRUCTION OPENING, SEE DRAWING 24900-C-035, SH 1.
- FOR REBAR REINSTALLATION DETAILS SEE DRAWING 1-41W719-3.
- PRIOR TO POURING REPLACEMENT CONCRETE, THE SURFACE OF THE CONSTRUCTION JOINT SHALL BE PREPARED BY APPLYING LARSEN'S WELD-CRETE EPOXY BONDING AGENT IN ACCORDANCE WITH THE MANUFACTURER'S INSTRUCTIONS.
- REPLACEMENT CONCRETE STRENGTH SHALL BE 4000 PSI AT 7 DAYS AND SHALL BE FURNISHED IN ACCORDANCE WITH SPECIFICATION 24900-C-321 AND PLACED IN ACCORDANCE WITH SPECIFICATION G-2.
- REPLACEMENT CONCRETE SHALL BE PLACED IN A SINGLE 24" LIFT.
- RESIDUAL HOLES FROM FORMWORK SUSPENSION RODS SHALL BE FILLED WITH 4000 PSI (MIN) NON-SHRINK GROUT PURCHASED AND PLACED IN ACCORDANCE WITH TVA GENERAL ENGINEERING SPECIFICATION G-51.
- THE THICKNESS OF THE SHIELD BUILDING DOME IN THE VICINITY OFTHE CONSTRUCTION JOINT ALONG THE PERIMITER OF THE SGR CONSTRUCTION OPENINGS MAY BE INCREASED TO A NOMINAL 25-1/2" IN ORDER TO MAINTAIN THE EXISTING CONCRETE COVER WHEN REBAR IS SPLICED USING BAR-LOCK COUPLERS. THE LOCALIZED RIDGE FORMED FROM THE INCREASED THICKNESS SHALL BE WIDE ENOUGH TO COVER THE LENGTH OF THE BAR-LOCK COUPLER SLEEVES AND SHALL HAVE 45 DEGREE BEVELED EDGES.

NOTE B:
- FOR FORM PLATE AND SEALED SCUTTLE HOLE DETAILS, SEE DRAWINGS 24900-C-036, SHEET'S 1 & 2.
- THE FORM PLATE SHALL BE ABANDONED IN PLACE.
- HVAC SUPPORTS AT AZIMUTH 7 DEGREES MAY BE ATTACHED OVER THE FORM PLATE.
- MISCELLANEOUS ITEMS, NOT TO EXCEED 50 POUNDS IN DEAD WEIGHT, MAY BE ATTACHED DIRECTLY TO THE FORM PLATE.

NOTE C:
- FOR CONCRETE PIER INSTALLATION DETAILS, SEE DRAWING 24900-C-035, SH 4.
- THE CONCRETE PIERS SHALL BE ABANDONED IN PLACE.

NOTE D:
- FOR PENETRATION LOCATIONS AND DETAILS, SEE DRAWING 24900-C-039 SHEETS 1 & 2.
- PENETRATIONS SHALL BE FILLED WITH 4000 PSI (MIN) NON-SHRINK GROUT PURCHASED AND INSTALLED PER TVA GENERAL ENGINEERING SPECIFICATION G-51. NON-SHRINK GROUT IS SAFETY RELATED.



HEATING VENT & A.C. SUPPORTS

TYPICAL SUPPORT NO. 47A055-147 & 47A055-148

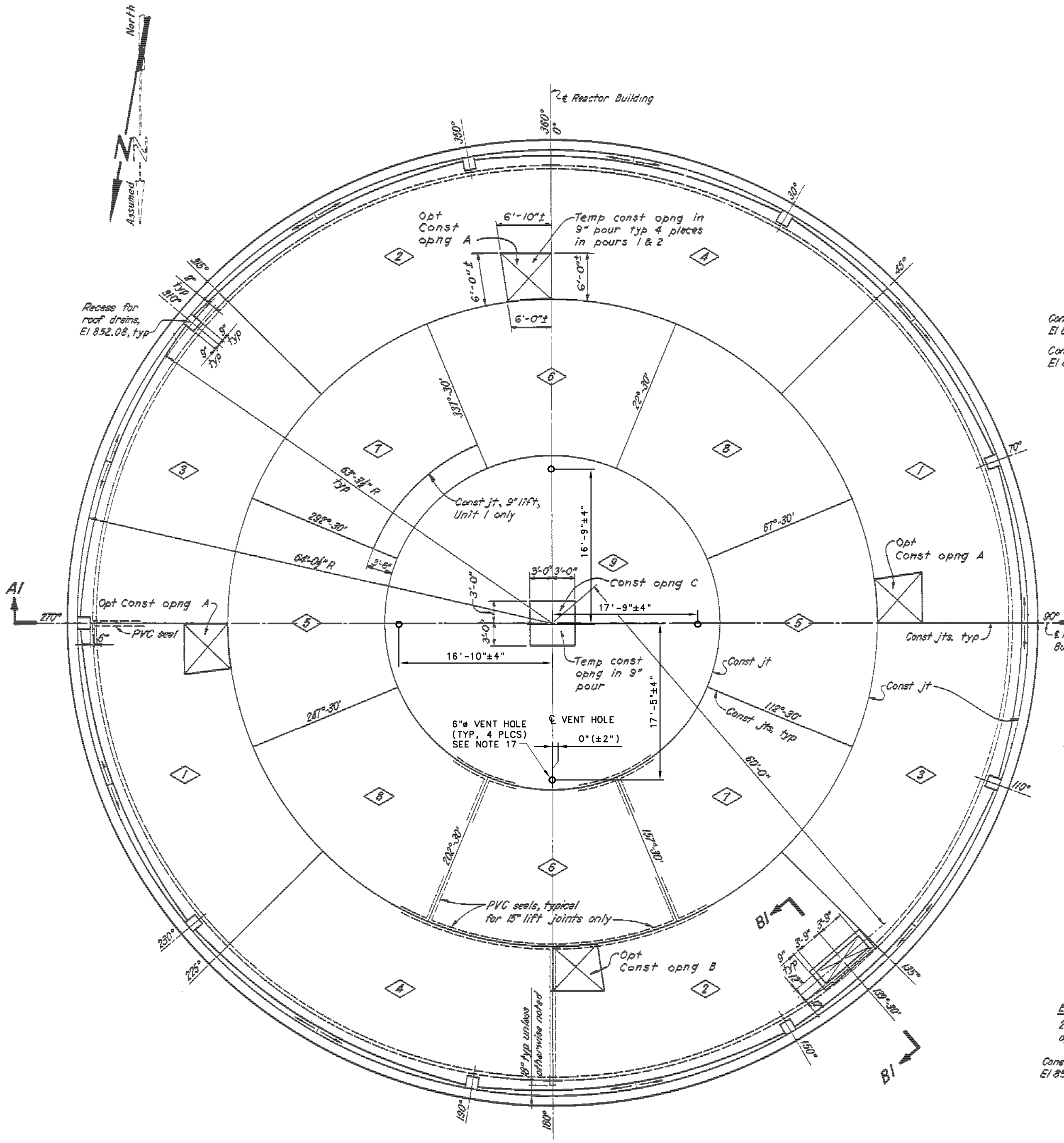
SUPPORTS REQUIRING THIS MODIFICATION ARE 148-1773, 147-1774 & 147-1775 (IF REQ'D) AS SHOWN IN SECTION A REFER TO: DCA-51729-172 (148-1773), DCA-51729-183 (148-1775) & DCA-51729-184 (148-1774) FOR DUCT SUPPORT DETAILS.

NOTE: USE FINGER SHIMS (A36) TO ACHIEVE AN APPROXIMATE 3"X 3" BEARING AREA AT BOLT LOCATIONS.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

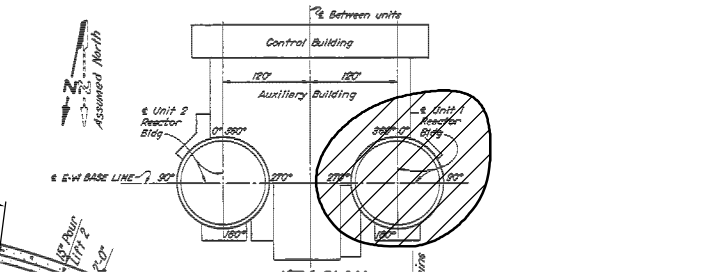
REACTOR BUILDING
UNIT 1
CONCRETE
DOME
OUTLINE
TVA DWG NO. 41N718-1 RF
FIGURE 3.8.1-7

CAD MAINTAINED DRAWING



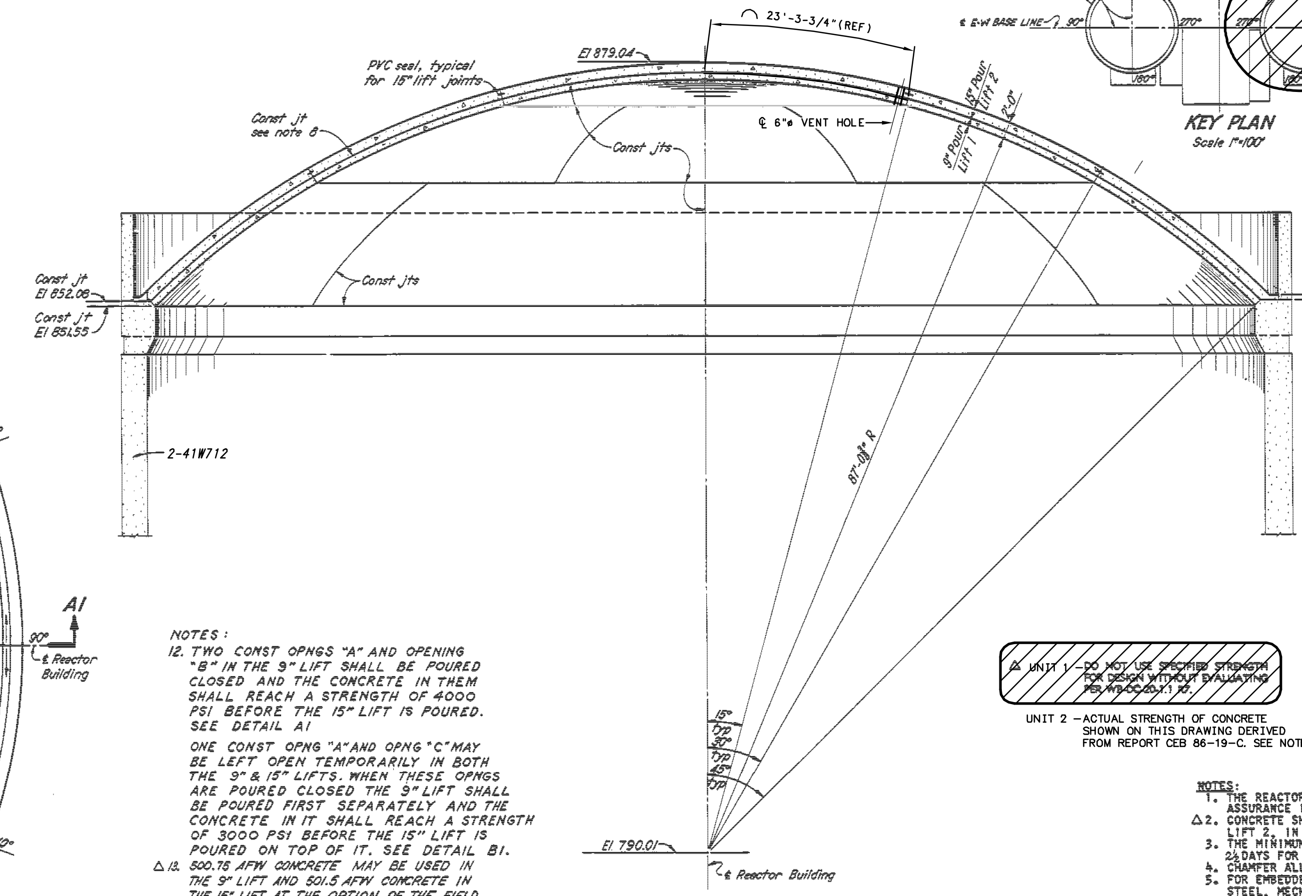
PLAN

Unit 1 As Shown
UNIT 2 OPP HAND
Const jts shown are typical for both 1st & 2nd lifts.



KEY PLAN

Scale 1"=100'



A1-A1

NOTES:

12. TWO CONST OPNGS "A" AND OPENING "B" IN THE 9" LIFT SHALL BE POURED CLOSED AND THE CONCRETE IN THEM SHALL REACH A STRENGTH OF 4000 PSI BEFORE THE 15" LIFT IS POURED. SEE DETAIL A1

    ONE CONST OPNG "A" AND OPNG "C" MAY BE LEFT OPEN TEMPORARILY IN BOTH THE 9" & 15" LIFTS. WHEN THESE OPNGS ARE POURED CLOSED THE 9" LIFT SHALL BE POURED FIRST SEPARATELY AND THE CONCRETE IN IT SHALL REACH A STRENGTH OF 3000 PSI BEFORE THE 15" LIFT IS POURED ON TOP OF IT. SEE DETAIL B1.

△ 13. 500.75 AFW CONCRETE MAY BE USED IN THE 9" LIFT AND 501.5 AFW CONCRETE IN THE 15" LIFT AT THE OPTION OF THE FIELD. IF THIS OPTION IS USED 5000 PSI CONCRETE HAS TO BE USED IN BOTH LIFTS.

14. PLACING TEMPERATURE OF CONCRETE SHALL NOT EXCEED 70°F AND SHALL NOT BE PLACED IF THE AMBIENT TEMPERATURE EXCEEDS 90°F. NO CONCRETE SHALL BE PLACED EARLIER IN THE DAY THAN 4 PM.

15. CONST JTS BETWEEN POURS 7 9 AND 8 9 ARE OPTIONAL ON 9" LIFT ONLY.

16. UNIT 2 ONLY - MINIMUM ACTUAL STRENGTH OF CONCRETE IS 4600 PSI FOR 1st LIFTS AND 4100 PSI FOR 2nd LIFTS.

17. FIELD TO FILL HOLES WITH EMACO S66 CI WHEN HOLES ARE NO LONGER REQUIRED.

△ UNIT 1 - DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

UNIT 2 - ACTUAL STRENGTH OF CONCRETE SHOWN ON THIS DRAWING DERIVED FROM REPORT CEB 86-19-C. SEE NOTE 16

NOTES:

1. THE REACTOR BUILDING IS A CLASS 1 STRUCTURE, AND QUALITY ASSURANCE IS REQUIRED.

△2. CONCRETE SHALL BE CLASS 400.75A FW, LIFT 1, AND 401.5A FW, LIFT 2, IN ACCORDANCE WITH TVA CONST SPEC G-2. SEE NOTE 13.

3. THE MINIMUM LENGTH OF TIME BETWEEN ADJACENT POURS SHALL BE 2½ DAYS FOR THE 9" LIFT AND 3 DAYS FOR THE 15" LIFT.

4. CHAMFER ALL EXPOSED EDGES 3/4".

5. FOR EMBEDDED PARTS, PIPES, AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.

6A. EACH LIFT SHALL BE PLACED IN THE NUMERICAL SEQUENCE SHOWN ON PLAN. SEE NOTE 11.

B. ALL BLOCKS OF THE SAME NUMBER IN EACH LIFT SHALL BE PLACED BEFORE BEGINNING THE NEXT NUMBERED GROUP.

C. ALL BLOCKS OF THE SAME NUMBER SHALL BE POURED CONCURRENTLY SUCH THAT, AT ANY TIME, THE DIFFERENCE IN POURED CONCRETE VOLUME BETWEEN ANY "SAME NUMBERED BLOCKS" IS NO MORE THAN 25% OF THE TOTAL VOLUME OF ONE COMPLETED BLOCK.

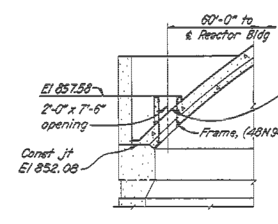
D. LIFT 1 SHALL BE COMPLETED BEFORE BEGINNING LIFT 2.

7. SHORING, INCLUDING THE STEEL TRUSSES (48N910), SHALL BE LEFT IN PLACE UNTIL STRENGTH OF 4000 PSI IS REACHED IN LIFT 1, AND SHALL BE REMOVED BEFORE START OF LIFT 2 CONCRETING.

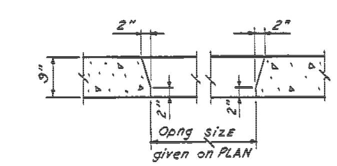
8. THE TOP SURFACE OF THE FIRST LIFT SHALL BE INTENTIONALLY ROUGHENED TO AN AMPLITUDE OF APPROXIMATELY ½".

10. AN OPTIONAL VERTICAL CONSTRUCTION JOINT MAY BE PLACED RADIALLY IN BLOCKS 1 THROUGH 4 TO DIVIDE EACH BLOCK INTO TWO EQUAL PARTS.

11. POUR SEQUENCE MAY BE ALTERED SUCH THAT ANY NUMBER OR ALL OF 1 AND 2, 3 AND 4, 5 AND 6, 7 AND 8 ARE INTERCHANGED.

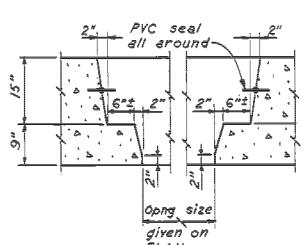


B1-B1



DETAIL A1

Typical section through opt temp const opng 9" lift
Scale ¾"=1'-0"



DETAIL B1

Typical section through temp const opng in both 9" and 15" lifts
Scale ¾"=1'-0"

* Field not required to replace that portion of PVC seal removed during repair made to concrete (Unit 1 Only).

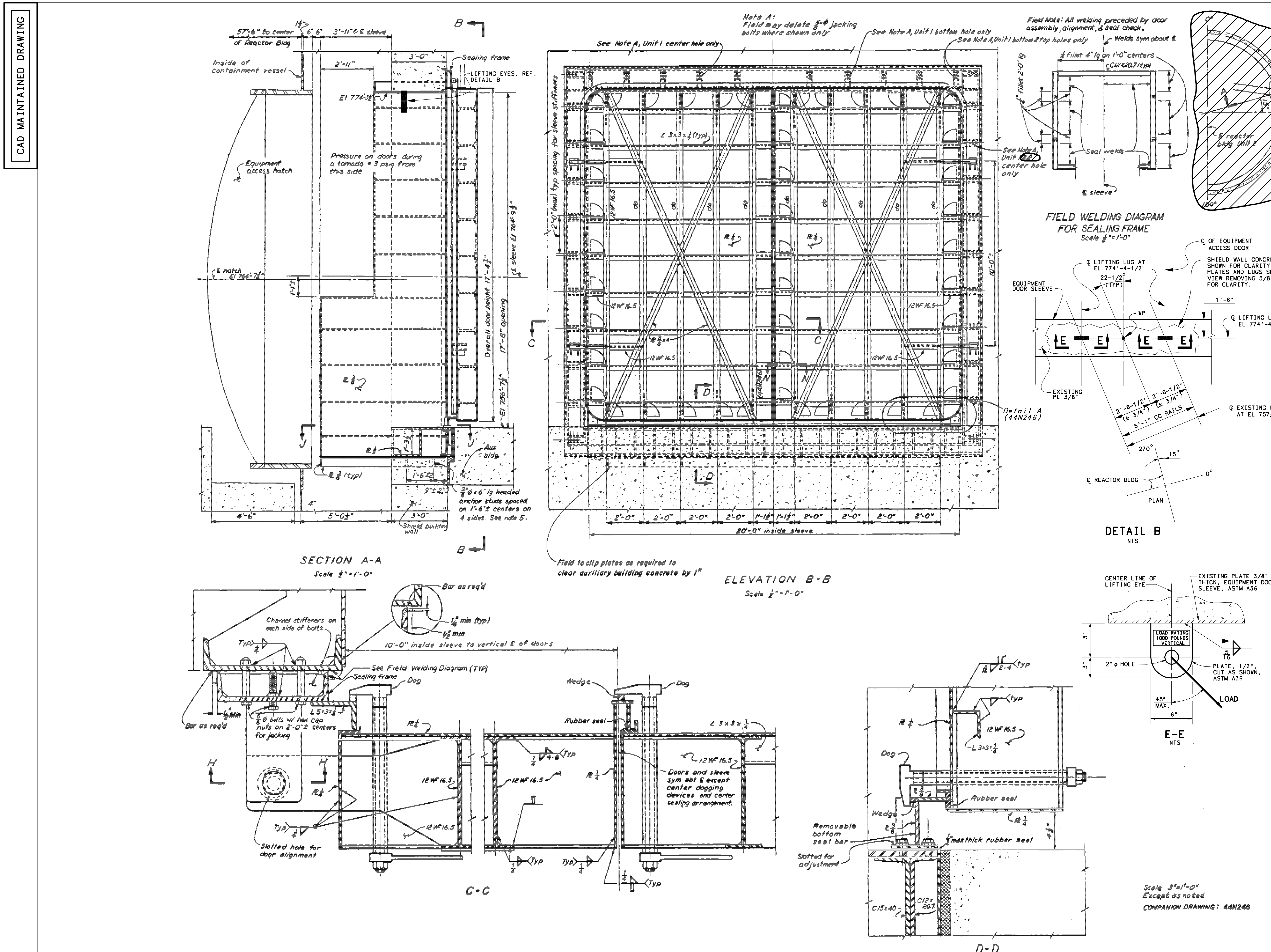
REFERENCE DRAWINGS:
41BW718 ______ BILL OF MATERIAL
41N1D060-19 ____ CONCRETE POUR DRAWING

FOR HATCHED AREAS SEE UNIT 1 AC DWG 41N718-1

UFSAR AMENDMENT 1

WATTS BAR FINAL SAFETY ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
CONCRETE
DOME
OUTLINE
TVA DWG NO. 2-41N718-1 R1
FIGURE 3.8.1-7(U2)



FIELD NOTES:

1. Field to modify seal bars by welding and/or grinding as required for fitup with seals when doors are dogged.
2. In the event a proper sealing can not be obtained using EPDM E401, Dow Corning 96-081 RTV adhesive sealant or 3145 RTV adhesive sealant may be used as a supplement to that seal. As an alternative to RTV, the sealing may also be achieved by the use of Polyurethane (E84-Class I), a two component type foam product of FOMO Products Inc. All these products are certified to withstand radiation.

NOTES:

1. One door and sleeve assembly required per unit - two total.
2. Field splice, if required, for sleeve and sealing frame to be on or near the vertical centerline.
3. For outline of concrete see 41N712-1 and 41N712-2.
4. For outline of hatch and hatch doors see 44N250 R1.
5. All embedded concrete anchors by TVA Field. ASTM A108.
6. All welds to be 1/4 continuous fillet unless otherwise noted.

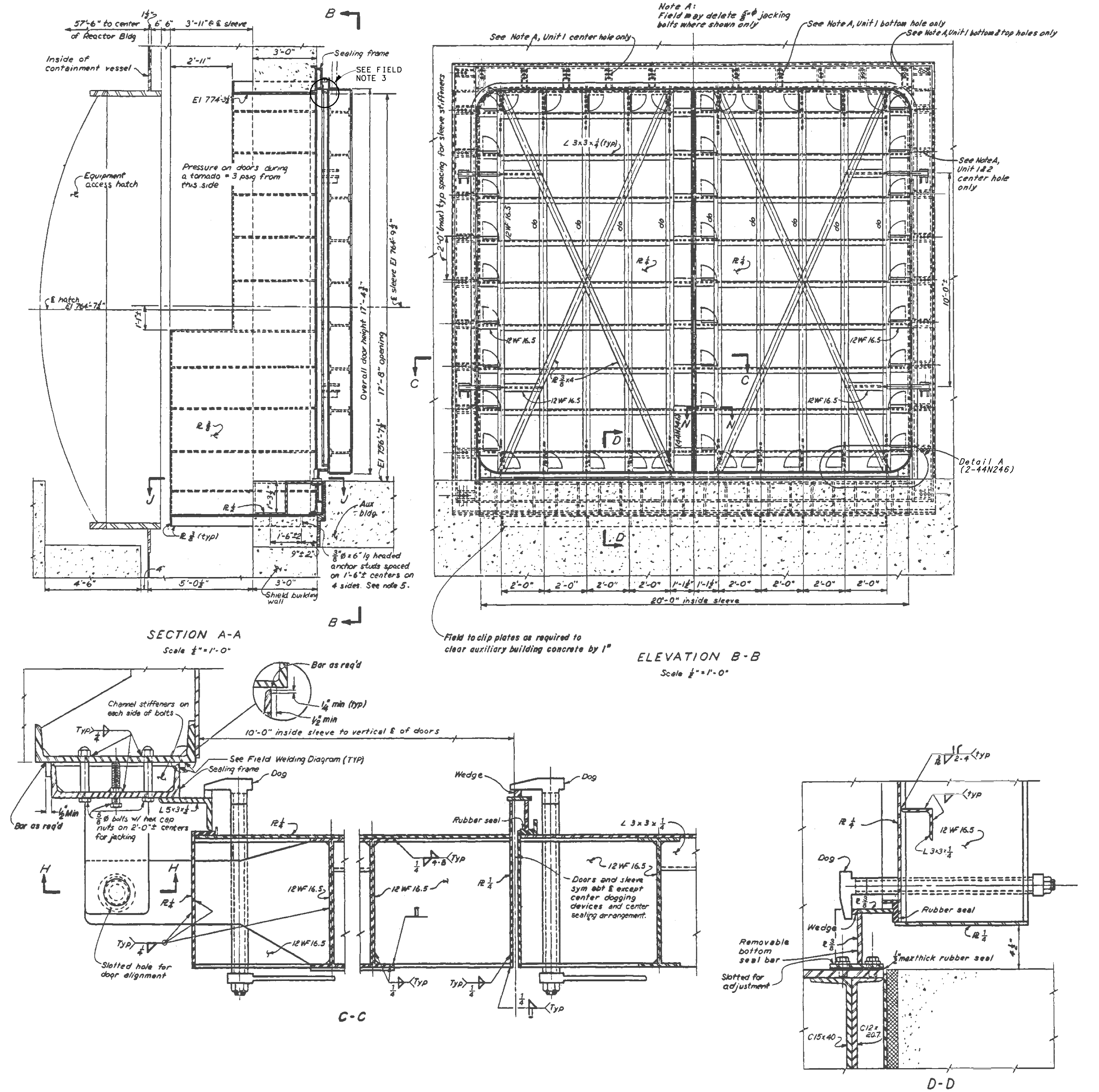
QUALITY ASSURANCE NOTES:

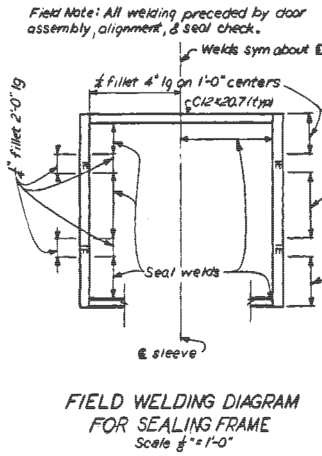
1. Doors and sleeves are Category I equipment.
2. Contractor's requirements per TVA Specification 2193.
3. Field welding and inspection of welds in accordance with General Construction Specification G-29C.
4. Field materials and work to be in accordance with Quality Level II of N36-881.

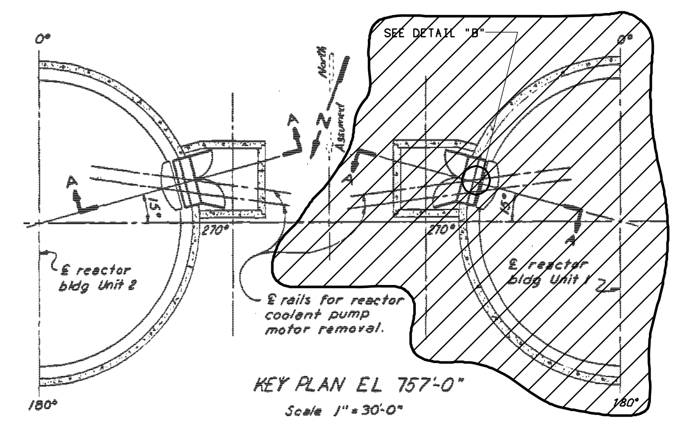
For manufacturer's details of equipment refer to E.H. Straus Co. file, TVA Contract no. 74C57-85881-N3H-9.

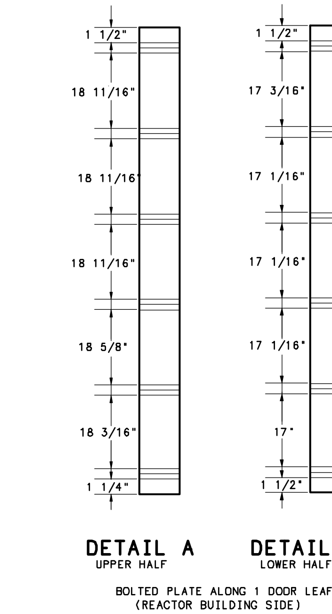
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

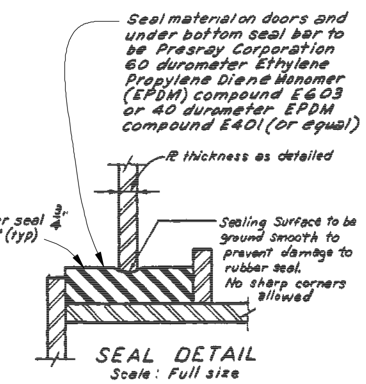
REACTOR BUILDING
UNIT 1
EQUIPMENT ACCESS DOORS
ARRANGEMENT & DETAILS-SHEET 1
TVA DWG NO. 44N245 RE
FIGURE 3.8.1-8











FIELD NOTES:

1. Field to modify seal bars by welding and/or grinding as required for fitup with seals when doors are dogged.
2. In the event a proper sealing can not be obtained using EPDM E401, Dow Corning 96-081 RTV adhesive sealant or 3145 RTV adhesive sealant may be used as a supplement to that seal.

   As an alternative to RTV, the sealing may also be achieved by the use of Polyurethane (E84-Class I), a two component type foam product of FOMO Products Inc. All these products are certified to withstand radiation.

3. During UNIT 2 completion, holes were drilled through the angle. Use materials specified in field note 2 to achieve proper sealing and prevent air leakage.

NOTES:

1. One door and sleeve assembly required per unit - two total.
2. Field splice, if required, for sleeve and sealing frame to be on or near the vertical centerline.
3. For outline of concrete see 2-41N712-1 and 2-41N712-2.
4. For outline of hatch and hatch doors see 2-44N250.
5. All embedded concrete anchors by TVA Field. ASTM A 108.
6. All welds to be 1/4 continuous fillet unless otherwise noted.

QUALITY ASSURANCE NOTES:

1. Doors and sleeves are Category I equipment.
2. Contractor's requirements per TVA Specification 2193.
3. Field welding and inspection of welds in accordance with General Construction Specification G-29C.
4. Field materials and work to be in accordance with Quality Level II of N36-881.

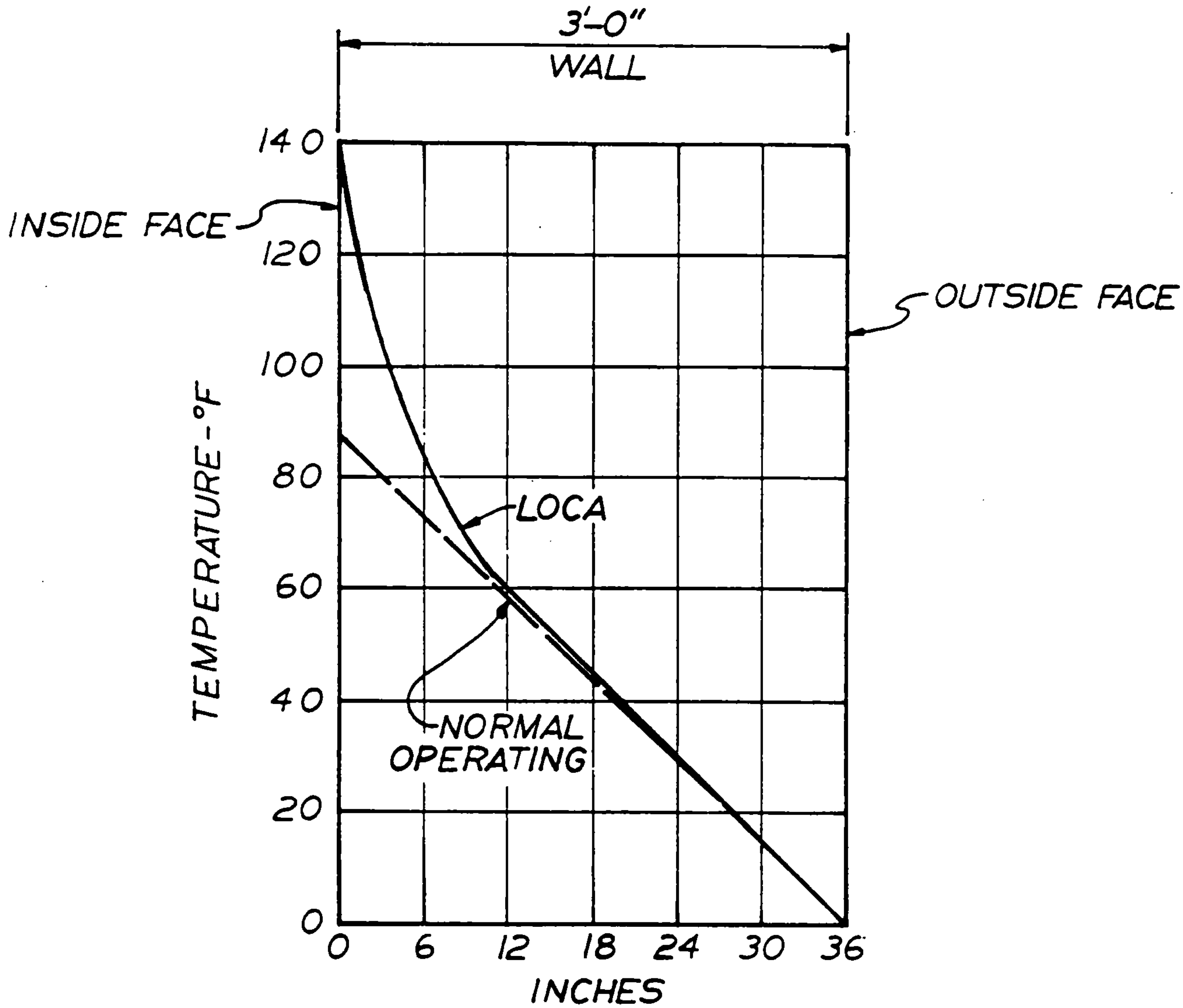
For manufacturer's details of equipment refer to E. H. Straus Co. file, TVA Contract no. 74C57-85881-N3H-9.

COMPANION DRAWING: 2-44N246

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
EQUIPMENT ACCESS DOORS
ARRANGEMENT & DETAILS-SHEET 1
TVA DWG NO. 2-44N245 R2
FIGURE 3.8.1-8(U2)



SHIELD BUILDING - EL 728 TO EL 745

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Shield Building Temperature Gradient Elevation 728-745 FIGURE 3.8.1-9

3.8.2 Steel Containment System

3.8.2.1 Description of the Containment and Penetrations

3.8.2.1.1 Description of the Containment

The steel containment vessel (SCV) for Watts Bar is a low-leakage, freestanding steel structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. Figure 3.8.2-1 shows the outline and configuration of the SCV.

The structure consists of side walls measuring 114 feet 8-5/8 inches in height from the liner on the base to the spring line of the dome and has an inside diameter of 115 feet. The bottom liner plate is 1/4 inch thick, the cylinder varies from 1-3/8 inch thickness at the bottom to 1-1/2 inch thick at the springline, and the dome varies between 1-3/8 inch thickness and 13/16 inch thickness with 15/16 inch thickness at the apex.

The bottom liner plate serves as a leak-tight membrane only (not a pressure vessel). The liner plate is anchored to the concrete by welding it continuously to steel plates embedded in and anchored into the base mat. The anchorage system of the cylindrical walls and the juncture of the cylinder to the base mat are shown in Figure 3.8.2-2.

The SCV dome is provided with a circumferential stiffener just above the springline supports, eight penetrations, and several attachments. Two penetrations are for the residual heat removal (RHR) spray system, two penetrations are for the containment spray system, and the remaining four penetrations are spares. The major attachments to the dome consist of lighting fixture supports, header supports for the RHR spray and containment spray systems, and the collector rail supports for the polar crane. Details of these penetrations and attachments are shown in Figure 3.8.2-3.

The SCV is provided with both circumferential and vertical stiffeners on the exterior of the shell. These stiffeners are required to satisfy design requirements for expansion and contraction, seismic forces, and pressure transient loads. The circumferential stiffeners were installed on approximately 10-foot centers during erection to ensure stability and alignment of the shell. Vertical stiffeners are spaced at 5° between the two lowest circumferential stiffeners. Other locally stiffened areas are provided at the equipment hatch and two personnel locks. Exterior pipe guides and restraints for the RHR spray and containment spray systems are attached to some of the circumferential stiffeners.

During the Unit 1 steam generator replacement, two construction opening were cut into the steel containment vessel. These construction openings were restored by reinstalling the removed steal sections and rewelding them to the remaining structure using full penetration welds. Abandoned-in-place reinforcement and support members added to stiffen the SCV during creation and use of the two construction openings are designed to remain attached to the SCV during a seismic event. The integrity of the restored vessel was verified by NDE and leak testing of the welds.

3.8.2.1.2 Description of Penetrations

Most penetration sleeves were preassembled into the SCV shell plates and stress relieved prior to installation of the plates into the SCV shell. Those penetration sleeves which required field installation were provided with insert plates of the same thickness as the shell plates and stress relieved as an assembly.

Equipment Hatch

The equipment hatch is composed of a cylindrical sleeve in the containment shell and a dished head 20 feet in diameter with mating bolted flanges. The flanged joint has double gasket seals with an annular space for pressurization and testing.

The equipment hatch door, sleeve, bolts, and attachments forming the pressure boundary were designed to Section III, Class MC of the ASME Code. The hatch guide system and hatch door hoisting support structure were designed to the AISC Design Specifications.

Details of the equipment hatch are shown on Figure 3.8.2-4.

Personnel Locks

Two personnel locks are provided for each unit. Each lock has double doors with an interlocking system to prevent both doors being opened simultaneously. Remote indication is provided to indicate the position of the far door. Quick-acting type equalizing valves are used to equalize the pressure inside the lock when entering or leaving the containment. Double seals are provided on the doors.

The personnel locks are completely prefabricated and assembled welded steel subassemblies designed, fabricated, tested and stamped in accordance with "Section III, Subsection NE" of the ASME Code.

Details of the personnel locks are shown on Figure 3.8.2-5.

Fuel Transfer Penetration

A 20-inch diameter fuel transfer penetration is provided for transfer of fuel between the fuel pool and the containment fuel transfer canal.

Expansion bellows were provided to accommodate differential movement between the connecting buildings. Figure 3.8.2-6 shows conceptual details of the fuel transfer penetration.

Spare Penetrations

Spare penetrations were provided to accommodate future piping and electrical penetrations. The spare penetrations consist of the penetration sleeve and head. Weld caps or closure plates are installed on spare penetrations to maintain containment integrity.

Purge Penetrations

The purge penetrations have one interior and one exterior quick-acting, tight-sealing isolation valve. A typical purge penetration arrangement is shown on Figure 6.2.3-2.

Electrical Penetrations

Medium voltage electrical penetrations for reactor coolant pump power use sealed bushings for conductor seals. The assemblies incorporate dual seals along the axis of each conductor.

Low voltage power, control and instrumentation cables enter the SCV through penetration assemblies which are designed to provide two leak tight barriers in series with each conductor.

All electrical penetrations are designed to maintain containment integrity for Design Basis Accident (DBA) conditions including pressure, temperature and radiation. Double barriers permit testing of each assembly as required to verify that containment integrity is maintained.

Qualification tests which may be supplemented by analysis, have been performed and documented on all electrical penetration assembly types to verify that containment integrity will not be violated by the assemblies in the event of a DBA. Existing test data and analysis on electrical penetration types may be used for this verification if the particular environmental conditions of the test were equal to or exceeded those for the Watts Bar Nuclear Plant.

Mechanical Penetrations

Typical mechanical penetrations are shown on Figures 3.8.2-7 and 3.8.2-8.

Mechanical penetration analysis is discussed in Section 3.8.2.4.6.

3.8.2.2 Applicable Codes, Standards and Specifications

3.8.2.2.1 Codes

The design of the containment vessel meets the requirements of the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NE, Winter 1971 Addenda and code cases 1431, 1517, 1529, 1493 and 1768.

The design of the bottom liner plates conforms to the requirements of the applicable subsections of the ASME Code, Section VIII, Division 1, and Section III, Paragraph NE-5120.

Nonpressure parts, such as supports, bracing, inspection platforms, walkways, and ladders were designed in accordance with the American Institute of Steel Construction (AISC) "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," Seventh Edition. The Eighth Edition is used for shapes not covered by the Seventh Edition.

Welding for these nonpressure parts was in accordance with the American Welding Society (AWS), "Structural Welding Code," AWS D1.1 (see Section 3.8.1.2, Item 4). Nuclear Construction Issues Group (NCIG) documents NCIG-01 and NCIG-02 (see Section 3.8.1.2, Item 11) may be used after June 26, 1985, to evaluate weldments that were designed and fabricated to the requirements of AISC/AWS.

The anchorage at the containment vessel meets the requirements of the ASME Code, Section III, with a maximum allowable stress for the anchor bolts of 2 x S_m.

All containment penetrations including the fuel transfer, purge, and mechanical within the jurisdiction of NE-1140 are designed to Section III, Class MC of the 1971 ASME Code. The penetration assemblies for those penetrations which attach to the nozzles out to and including the valve or valves required to isolate the system and provide a pressure boundary for the containment function are designed to Section III, Class 2 of the ASME Code. Spare penetrations including the nozzle caps are designed to Section III, Class MC of the ASME Code.

Two welds (1-074B-D045-01A and 1-074B-D045-08A) in the containment sleeves at the Unit 1 RHR sump have radiographic indications which have been interpreted as exceeding the radiographic acceptance criteria of ASME Section III. TVA has performed calculations (WBN-MTB-025 and CEB-CQS-415) which document the basis for the acceptability of these welds.

3.8.2.2.2 Design Specification Summary

Design Criteria

The containment vessel, including access openings and penetrations, is designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident (LOCA), including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in any emergency cooling system, will not result in undue risk to the health and safety of the public, and is designed to limit below 10 CFR 100 values the leakage of radioactive fission products from the containment under such LOCA conditions.

The basic structural elements considered in the design are the vertical cylinder and dome acting as one structure, and the bottom liner plate acting as another. The bottom liner plate is encased in concrete and is designed as a leak tight membrane only. The liner plate is anchored to the concrete by welding it continuously to steel members embedded and anchored in the concrete basemat.

On the exterior at approximately 20-foot centers the containment shell is provided with circular inspection platforms which also are designed as permanent circumferential stiffeners. Additional circumferential stiffeners are provided at personnel and equipment hatches and at other large attached masses, along with vertical stiffeners for some distance above and below these attachments. Also, additional permanent circumferential stiffeners were added for stability. Temporary stiffening was not required to meet tolerance requirements specified by TVA in the erection of the vessel. The design provides for movements of the vessel and supports due to expansion and contraction, pressure transient loads, and seismic motion. No allowance is made for corrosion in determining the material thickness of the vessel shell.

The following pressure and temperatures were used in the design of the vessel:

Overpressure test (1)	16.9 psig
Maximum internal pressure (2) (3) (4)	15.0 psig at 250°F
Design internal pressure (3)	13.5 psig at 250°F
Leakage rate test pressure	15.0 psig
Design external pressure	2.0 psig
Lowest service metal temperature	30°F
Operating ambient temperature	120°F
Operating internal temperature	120°F
Design temperature	250°F

In addition, the evaluations of the vessel design have considered a harsh environment temperature of 327°F.

1. 1.25 times design internal pressure as required by ASME Code, NE-6322.

2. See Paragraph NE-3312(b) of Section III of the ASME Code which states that the "design internal pressure" of the vessel may differ from the "maximum containment pressure" but in no case shall the design internal pressure be less than 90% of the maximum containment internal pressure.

3. Typical pressure transient curves are presented in Section 6.2.1. These curves show the transient pressure buildup in the compartments after a LOCA or DBA before a steady-state pressure of 15.0 psig is reached.

4. Shell temperature transient curves are presented in Appendix 3.8A. These curves show the shell temperature at the lower compartment wall, upper compartment wall, and ice condenser wall. The maximum containment wall temperature is 220°F.

5. A postulated main steam line break (MSLB) results in high environmental temperatures (327°F Unit 1 maximum, 325°F Unit 2 maximum) inside the lower compartment of the SCV. However, the coincident internal pressure is lower.[10]

In order to ensure the integrity of the containment, an analysis of the missile and jet forces due to pipe rupture was considered. This problem was eliminated by providing barriers to protect the containment vessel. Typical barriers are the main operating floor (Elevation 756.63) and the crane support wall. An example of a special barrier is the guard pipe enclosing the main steam and feedwater pipes between the Shield Building and the crane wall.

Allowable Stress Criteria

Allowable stress criteria for the containment vessel are shown in Table 3.8.2-1. The response of the containment vessel to seismic and pressure transient loadings results in a condition in which buckling of the steel shell may occur. Since the ASME Code does not define the allowable buckling stresses for this type of loading condition, an acceptable buckling criteria with appropriate factors of safety is given in Appendix 3.8B.

3.8.2.2.3 NRC Regulatory Guides

Applicable NRC Regulatory Guides are shown below. These guides were used as the basis for design of a number of safety oriented features.

Regulatory Guide 1.4: Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors

A dynamic analysis of the containment vessel was made for the pressure transient loadings. The containment vessel and penetrations were designed to withstand the maximum internal pressure that could occur due to a LOCA and the jet forces associated with the flow from the postulated pipe rupture.

Regulatory Guide 1.7: Control of Combustible Gas Concentrations in Containment

For Unit 1, a hydrogen recombiner is provided inside the containment to control the hydrogen buildup following a loss-of-coolant accident. In addition, the containment vessel has a hydrogen mitigation system designed to mitigate the effects of hydrogen releases after a LOCA (Section 6.2.5A).

For Unit 2, the containment vessel has a hydrogen mitigation system designed to mitigate the effects of hydrogen releases after a LOCA (Section 6.2.5).

Regulatory Guide 1.28: Quality Assurance Program Requirements (Design and Construction).

A Quality Assurance Plan for the Watts Bar Nuclear Plant was developed as a comprehensive plan for the design and construction of the Watts Bar Nuclear Plant. The Quality Assurance Plan of the Westinghouse Electric Corporation, the supplier of the Nuclear Steam Supply System, is also contained therein.

The plans were prepared to assure that the control of quality was achieved and documented for each phase of design, material selection, fabrication installation, and/or erection in accordance with the approved specifications and drawings. The plans relate principally to the reactor coolant and safety system, the containment and other components necessary for the safety of the nuclear portion of the plant.

The plan assures that:

1. Final design requirements and final detailed designs are in accordance with applicable regulatory requirements and design bases.

2. Components and systems to which this plan applies are identified and that the final design takes into account the varying degrees of importance of components and systems as evidenced by the possible safety consequences of malfunction or failure.

3. Purchased material and components fabricated in vendor shops conform to the final design requirements.

4. Components and systems are assembled, constructed, erected, and tested in accordance with the final design requirements and to requirements specified in the UFSAR.

5. The as-constructed plant can be operated and maintained in accordance with requirements specified in the UFSAR.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 Design Loads

The following loads are used in the design of the containment vessel and appurtenances. The loadings for the containment vessel were combined as in Section 3.8.2.3.2. The allowable stress criteria are shown in Table 3.8.2-1.

Dead Loads

These loads consist of the weight of the SCV, penetration sleeves, equipment and personnel access hatches, and attachments supported by the vessel. The weight of abandoned-in-place reinforcement and support members added to stiffen the SCV during creation and use of the two construction openings made for the Unit 1 steam generator replacement has been included in the evaluated load combinations.

Live Loads

Penetration loads as applicable.

Floor load of 100 psf or 1,000 pounds concentrated moving loads applied to the passage area of the personnel air locks.

Construction and snow loads at 50 psf, snow load at 20 psf during construction is considered but not simultaneously with other construction loads.

Floor load of 50 psf plus 225 pounds per linear foot for walkways.

Thermal Stresses During Design Basis Accident (DBA)

The containment vessel is designed to contain all the effluent which would be released by a hypothetical LOCA. This accident assumes a sudden rupture of the reactor coolant system which would result in a release of steam and a steam-air mixture in the vessel. It is calculated that this mixture would cause a lower compartment temperature of 250°F and an upper compartment temperature of 190°F, both occurring essentially instantaneously. After the accident, an internal spray system will commence spraying in the upper compartment only. The spray will discharge water on the interior of the upper compartment. For shell temperature transients refer to Appendix 3.8A.

A MSLB produces temperatures in the lower compartment of 327°F (Unit1) or 325°F (Unit 2) with coincident internal pressure and seismic loadings defined in load combinations 3A and 4A.

Hydrostatic Loads

The containment vessel is designed for three separate flood conditions. Hydrostatic load, Case 1B, accounts for the flooded condition due to ice melt from the ice condenser after the DBA.

After all the ice has melted the containment will be flooded to Elevation 719 feet - 3 inches. Also considered is the loading condition during meltdown (hydrostatic load, Case IA). Water will rise to a depth of 2 feet on the floor of the ice condenser. At this time, the depth of water on the containment cylindrical shell will be 9 feet - 3 inches.

Hydrostatic load, Case II, accounts for the post-accident fuel recovery condition. In order to remove fuel from the containment after the DBA, the containment vessel is designed for an internal hydrostatic head of 47 feet- 3 inches.

For hydrostatic load cases refer to Figure 3.8.2-1.

Ice Condenser Duct Panel Loads

The outer duct panels of the ice condenser are attached to the containment with threaded studs. These panels impart small horizontal and vertical forces on the containment shell under seismic conditions. The distribution of these loads to the shell is shown in Figure 3.8.2-1.

Equipment Loads

Equipment loads are those specified on drawings supplied by manufacturers of the equipment.

Overpressure Test

To test the structural integrity of the vessel an overpressure test of 125% of design pressure is applied under controlled conditions.

External Pressure Load

The containment vessel is stiffened and designed to withstand an external pressure of 2.0 psig.

Seismic Loads

Seismic loads are generated using the methodology discussed in Sections 3.7.1 and 3.7.2.

Wind Loads

The containment vessel and its penetrations are completely enclosed by the Shield Building, and are therefore not subject to the effects of wind and tornadoes.

However, during construction, the vessel dome was exposed to the elements for a short duration. For this construction condition, a wind load of 30 psf on the projected area of the vessel dome was considered.

Non-Axisymmetric Transient Pressure Loads

The division of the containment into compartments is described in Section 6.2.1 and in Section 3.8.2.4.4.

Pressure transient loads are considered for occurrence of the DBA (double- ended rupture of the reactor coolant system) in all 6 lower compartment volumes. The curves presented in Section 6.2.1 represent the containment pressure transients for the controlling break locations 1 through 6 for each of the 49 containment elements.

The pressures and differential pressures shown on these figures have no margin. The initial containment pressure was assumed to be 0.3 psig. This allows for an initial containment pressure before containment venting is required. The most severe containment pressure differences occur during the first 0.9 second of the blowdown.

For structural design purposes the pressures represented by the curves are increased by 45%. This allows for changes in such factors as equipment configuration and openings between compartments, which can influence the flow characteristics of the containment space, the effects of moisture entrainment, and tolerances in the analytical constraints used in the code. (The effects of moisture entrainment, investigated by TVA and Chicago Bridge and Iron Company (CB&I), do not control the design of the containment vessel for any loading condition).

Local loadings from commodities attached to the SCV are calculated using dynamic response spectra generated for each area of the vessel. These spectra reflect the response of the vessel to localized dynamic pressure loadings resulting from postulated high energy pipe breaks. See Sections 3.6A and 3.6B for discussions of how these high energy break locations are determined.

3.8.2.3.2 Loading Conditions

The following loading conditions are used in the design of the containment vessel:

1. Normal Design Condition
    - Dead load of containment vessel and appurtenances
    - Lateral and vertical load due to one-half SSE
    - Personnel access lock floor live load
    - Penetration loads
    - Design Internal Pressure or Design External Pressure
    - Design temperature

2. Normal Operation Condition Operating Basis Earthquake (OBE)
    - Dead load of containment vessel and appurtenances
    - Lateral and vertical load due to OBE
    - Penetration loads
    - Spray header and lighting fixture live loads
    - Walkway live loads
    - Personnel access lock floor live load
    - Internal temperature range 60°F to 120°F

3A. Upset Condition - DBA and OBE
    - Dead load of containment vessel and appurtenances
    - Design internal pressure
    - Lateral and vertical load due to OBE
    - Penetration loads
    - Thermal stress loads including shell temperature transients
    - Hydrostatic Load Case IA or IB
    - Internal temperature range 80°F to 250°F

3B Upset Condition - DBA and OBE

- Dead load of containment vessel and appurtenances
- Pressure transient loads
- Lateral and vertical load due to OBE
- Penetration loads
- Thermal stress loads including shell temperature transients
- Hydrostatic Load Case IA or IB
- Internal temperature range 60°F to 120°F

3C. Upset Condition MSLB

- Dead load of containment vessel and apurtenances
- Internal pressure coincident with MSLB[10]
- Lateral and vertical load due to 1/2 SSE.
- Spray header loads
- Ice condenser duct load
- Thermal load due to temperature range 80°F to 327°F (Unit 1) or 325°F (Unit 2)
- Penetration loads

4A. Emergency Condition - DBA and SSE

- Dead load of containment vessel and appurtenances
- Design internal pressure
- Lateral and vertical load due to SSE
- Penetration loads
- Thermal stress loads including shell temperature transients
- Hydrostatic Load Case IA or IB
- Internal temperature range 80°F to 250°F

4B <u>Upset Condition - DBA and SSE</u>

- Dead Load of Containment vessel and appurtenances
- Pressure transient loads
- Lateral and vertical load due to SSE
- Penetration loads
- Thermal stress loads including shell temperature transients
- Hydrostatic Load Case IA or IB
- Internal temperature range 60°F to 120°F

4C. <u>Emergency Condition MSLB</u>

- Loads are same as in Condition 3C except lateral and vertical load due to SSE

5. <u>Construction Condition at Ambient Temperature</u>

- Dead load of containment vessel and appurtenances
- Snow load at 20 psf
- Lateral load due to wind
- Temporary construction live loads on catwalks, platforms, and hemispherical head including support of the first pour of the concrete Shield Building dome.

6. <u>Test Condition at Ambient Temperature</u>

- Dead load of containment vessel and appurtenances
- Internal test pressure
- Weight of contained air

7. <u>Post-Accident Fuel Recovery Condition with Flooded Vessel</u>

- Dead load of containment vessel and appurtenances
- Hydrostatic Load Case II

3.8.2.4 Design and Analysis Procedures

3.8.2.4.1 Introduction

The design, fabrication, and erection of the SCV were contracted to Chicago Bridge and Iron Company (CB&I), Oakbrook, Illinois. The design of the vessel was reported by CB&I in a 12-volume stress report from which the following design and analysis procedures were taken. TVA reviewed the stress report as required by ASME Code Section NA-3260. Furthermore, TVA performed a complete design review of CB&I work to insure the adequacy of the design. As part of the design review, independent analyses were performed for seismic, thermal and pressure transient loading conditions.

Compressive stresses in the containment vessel are produced by dead, live, seismic, and pressure transient loads. But pressure transient loads are by far the most significant loads to the stability of the vessel. Therefore, buckling is addressed only in Section 3.8.2.4.4.

3.8.2.4.2 Static Stress Analysis

A detailed stress analysis of all major structural components was prepared in sufficient detail to show that each of the stress limitations of the ASME Boiler and Pressure Vessel Code, Section III, Section NE-3000 was satisfied when the vessel is subjected to the loading combinations enumerated in this section.

Details of the juncture of the cylinder to the base mat are shown in Figure 3.8.2-2. In the analysis, the juncture was considered to be a point of infinite rigidity. The cylinder at this point cannot expand or rotate under the internal pressure and temperature load conditions; hence, shear and moment are introduced into the cylinder wall.

At the point the knuckle is welded to the vessel, a backup stiffener is used. This stiffener gives added rigidity at the point of the weld. Additional protection of the knuckle is accomplished by encasing the knuckle in 'Fiberglass' before floor concrete placement.

The embedded knuckle was designed to take interior pressure plus internal or external hydrostatic loads. It was assumed that cracks can occur in the concrete allowing pressure loads on the embedded knuckle. Anchor bolts were post-tensioned to prevent any cracking of the concrete. Thermal and pressure discontinuity stresses in the containment occur one foot above the last weld of the knuckle.

The stresses due to dead loads internal, and snow loads were determined at a sufficient number of locations to define the state of stress in the vessel under these loadings. Wind, snow and external support loads on the dome occurred during construction. Stresses due to dead loads, internal and external pressure were determined by hand calculations using classical strength of materials theory. Detail stresses in the embedment region at the base of the vessel were determined from a shell model of the vessel using CB&I computer program 781 described in Appendix 3.8C. The circumferential stiffeners on the embedment region were modeled as horizontal elements and the effect of vertical stiffeners was considered by modeling the shell plate as an orthrotropic material. Forces and bending moments due to the various loads were given by CB&I computer program 781, whereas the resulting detailed stress distribution was calculated using actual geometry of the vessel and stiffening in this region.

Design of spherical and cylindrical vessels for internal and external pressure is explicitly treated in Section NE of the ASME Boiler and Pressure Vessel Code. The vessels as designed are in full compliance with the Code requirement for internal and external pressure and provisions applicable to other load conditions.

3.8.2.4.3 Dynamic Seismic Analysis

The SCV dynamic analysis is discussed in Section 3.7.2.1.

3.8.2.4.4 Non-Axisymmetric Pressure Loading Analysis

The non-axisymmetric pressure loading (NASPL) results from an assumed sudden rupture in the reactor coolant system. The associated pressure loads are dynamic in nature and vary with time in both the circumferential and meridional directions in the vessel. The loads are non-axisymmetric for a short period culminating in uniform internal pressure throughout the containment. For analysis purposes, the containment was subdivided into forty-nine volumes and pressure-time histories determined for each volume for the postulated rupture, i.e., each break in the reactor coolant system. The pressure histories for each of the volumes were computed by the Westinghouse Electric Corporation using the TMD code network documented in Section 6.2.1.3. Figures 3.8.2-10 and 3.8.2-11 show the volumes used to characterize the pressure in the containment.

Dynamic analyses were made by CB&I for twelve breaks in the reactor coolant piping, six hot leg and six cold leg breaks. Two separate and distinct analysis methods were used in the design process. The overall vessel response was determined by a dynamic analysis treating the vessel as a lumped mass cantilever beam and by a dynamic shell analysis which considered the effects of local vibration modes.

1. Beam Analysis

   In the CB&I lumped mass beam analysis, each mass represented the mass of the vessel stiffeners and attached masses. The cantilever beam model was loaded with the forces from the NASPL. The forces were resolved into X and Y components and applied as mass point loads in the north-south and east-west directions. The response of the model to non-axisymmetric pressure transients was calculated by CB&I Program 1642 described in Appendix 3.8C. It employs the method of numerical integration and solves for natural frequencies, accelerations, overturning moments, and shears.

2. Shell Analyses

   Independent dynamic shell analyses of the containment were performed by both CB&I and TVA. The shell model used by CB&I is shown in Figure 3.8.2-12. The method of analysis involves a numerical integration technique operating on the governing differential equations. Linear behavior and axisymmetric geometry were assumed. The total transient response was calculated by the sum of the harmonic responses with the input loads being represented by Fourier Series. A full explanation of the method is given in Reference [1]. A number of CB&I proprietary programs, all described in Appendix 3.8C, were employed to arrive at the final shell responses. Figure 3.8.2-13 is a flow diagram of the analysis process with a brief description of the function accomplished by each computer program. CB&I Program 1624 (also in Appendix 3.8C) calculated acceleration response spectra at various elevations and azimuths from the acceleration histories.

TVA performed an independent shell analysis of the transient pressure response. A finite element model was used and the solution calculated by numerical integration. The agreement with the CB&I analysis was good. Since the TVA shell analysis was merely a check on the CB&I analysis, full documentation of the process and the programs used are not included herein.

The pressures were factored by 1.45 for computing responses to be used to ensure compliance with the buckling criteria in Appendix 3.8B. A factor of 1.80 was used in the design of the anchorage (see Section 3.8.2.4.8).

3.8.2.4.5 Thermal Analysis

A thermal analyses was performed on the containment for a loss-of-coolant accident. The shell temperature transients due to a double end rupture of a reactor coolant pipe are described in Appendix 3.8A. The tolerable temperature rise for the steel containment is well above the temperatures shown, since the steel shell was designed for the basic stress limits of Section NB-3221 and Section NB-3222.2 of the ASME Boiler and Pressure Vessel Code, Section III, for ASME SA-516, grade 70 steel at 300° F.

Also, as seen by these curves, the containment shell will experience an unbalanced temperature loading for the three compartments. The temperature difference between any two adjacent points on the vessel is held within the limits of Section NB-3222.4 of the code.

TVA performed a study to determine the effect of MSLB temperature on the SCV. The impact of the thermal movements on attached penetrations and appurtenances was also accounted for in this study. This study indicated that the SCV and attachments are still within acceptable ASME stress limits under MSLB.

3.8.2.4.6 Penetrations Analysis

The vessel manufacturer is responsible for the design of the steel containment including the reinforcement required at the penetrations. The specifications required the manufacturer to submit all preliminary design calculations for TVA's review before any material was detailed or fabricated. Penetrations requiring requalification after CB&I completed their contract were analyzed by TVA. TVA used essentially the same methodology and design criteria as CB&I. However, TVA used its own in-house developed computer program (PNA100 or TPIPE) to sum load combinations and hand calculations to calculate nozzle stresses and a public domain program (WERCO) to calculate shell stresses. The WERCO program employs the methodology of the Welding Research Council Bulletin No. 107.

Also, TVA performed an independent analysis of the steel containment, including the reinforcement required at penetrations.

Secondary and local stresses at penetrations subjected to applied loads were analyzed by CB&I programs 1027 and 1036, which are described in Appendix 3.8C. These programs employ the methods of the Welding Research Council Bulletin No. 107 in the analysis of the containment shell.

Penetrations not subjected to applied loads were designed in accordance with Section NE-3332 of Section III, ASME Code. Most penetrations were preassembled into the containment vessel shell plates and stress relieved prior to installation of the plate into the containment vessel shell. All other penetrations were installed in insert plates of the same thickness at the perimeter as the shell plates and stress relieved as assemblies. As a result, no reinforcement is provided in excess of that available in the shell and neck. Large penetrations, such as the large equipment hatch and personnel access locks, require stiffeners for reinforcement.

The penetrations subjected to external loads are supplied with pipe of sufficient wall thickness to resist these loads. Where one or more externally loaded penetrations are in close proximity to another externally loaded penetration or pad plate, the shell was analyzed for the interactive effects of these loaded penetrations.

The external loads were assumed to be reversible and the maximum stress combination was determined. Since pressure affects the design of the penetrations, a pressure equal to the internal design pressure is considered to act in conjunction with the externally applied loads.

Figure 3.8.2-14 shows the stresses assumed to be present in the analysis of the shell in the vicinity of the penetrations. These assumed stresses, which are due to internal containment pressure, are added to the stresses resulting from the externally applied loads before determining the stress intensities. The assumed stresses are employed as shown in Figure 3.8.2-14 for most of the penetrations. However, it is permissible to reduce these initial stresses when the penetration is provided with greater reinforcement than is required by Section III. At the point of intersection of the shell and penetration, a factor equal to the ratio of the area required for reinforcement within the two-thirds limit to the area available for reinforcement may be used to reduce the assumed initial stresses. At points in the shell away from this intersection, the factor becomes the ratio of required shell thickness to actual shell thickness. This reduction method was used on penetrations which were over-stressed when the assumed initial stresses used were as shown in Figure 3.8.2-14. While the factor for all penetrations using this method was less than 0.5, the minimum factor used in the analysis was 0.5.

The neck of the penetration was analyzed using CB&I Program 1392, described in Appendix 3.8C. This program computes the stresses in the neck at two points. The first point is located at a distance from the shell that is outside the normal limits for area replacement. The stresses at this point are due to the external loads and to the containment design pressure acting within the pipe. The second point is located within the area considered for area replacement. In addition to the stresses due to external loads and containment pressure, an assumed stress is also included. This assumed stress is as outlined above at the point of intersection of the shell and penetration and may be modified as discussed above. Permanent caps for spare penetrations are designed in accordance with ASME rules. Flanged penetrations are provided with double gasket details which permit the testing of the gaskets by pressurizing the air space between the gaskets.

The Heating, Ventilation and Air Conditioning (HVAC) penetrations were also analyzed by TVA. The entire piping assembly from the flexible connection in the Reactor Building to the flexible connection in the annulus was modeled including pipe, isolation valves, and pipe supports using discrete finite element representation. Shell flexibility was taken into account at the nozzle/shell intersection and at the hanger/shell attachments by inputting equivalent translational and rotational stiffness rates.

A response spectrum modal analysis was performed for the seismic and design basis accident condition using the floor response spectra nearest to the penetration locations. The total stress in the nozzle was calculated using the absolute summation of dead load, seismic, and DBA. The stress in the shell was analyzed by inputting the loads above into WERCO.

Other non-process and electrical penetrations were also analyzed by TVA. These penetrations were analyzed using the static acceleration technique in which the weight is multiplied by the peak accelerations from the seismic and DBA spectra times a 1.5 amplification factor and applied at the mass center of the assembly. The resulting stresses in the nozzle and shell were calculated using the technique used for qualifying mechanical penetrations.

3.8.2.4.7 Interaction of Containment and Attached Equipment

Some items rigidly attached to the containment respond in a non-rigid manner due to the local flexibility of the containment. This effect was analyzed for a number of penetrations and other attachments, but was found to be significant only for the equipment hatch, two personnel locks, and the HVAC penetrations.

The following procedure was followed in the equipment hatch and personnel lock analyses:

1. Linear and rotational mass moments of inertia were calculated in the radial, circumferential, and longitudinal directions. (The rotational degrees-of-freedom were considered because the centers of mass did not lie in the plane of the containment shell).

2. The local stiffnesses of the hatch and locks were calculated for the above degrees-of-freedom. A method developed by Bijlaard[2] was used.

3. The periods of vibration were calculated for motions in the radial (push-pull) direction, and in the circumferential and longitudinal (swinging) directions by the equation:

$$T = 2\pi\sqrt{\frac{I_o}{K}}$$

   where I_o and K are the mass moments of inertia and stiffnesses, respectively.

4. The response accelerations for seismic excitation and the pressure transients were taken from the spectra described in Sections 3.8.2.4.3 and 3.8.2.4.4, respectively.

5. The total structural response was found by the sum of the effects of the seismic, pressure transient, and dead weight loads.

All of the above calculations were performed by hand. The periods of vibration of the equipment hatch and the personnel locks in the three principal directions were all greater than 0.03 seconds, which is used as the demarcation between rigid and non-rigid vibration.

3.8.2.4.8 Anchorage

The containment vessel anchorage system consists of anchor bolts, an embedded anchor plate, and an anchor bolt bearing ring which attaches to the first shell ring. Details of the anchorage are shown in Figure 3.8.2-2.

Two rows of 3-1/2 inch anchor bolts are provided with one row on the outside of the shell and one row on the inside of the shell. The bolts in each row are spaced at two degrees and located in pairs on radial lines. The rows are located at equal distances from the center line of the shell.

The anchor bolts are embedded in the concrete to the maximum depth available. The majority of the bolts are embedded to a depth such that the lowest point on the bolts is slightly above Elevation 687.0. The remainder of the anchor bolts, located in the area of the pipe sleeves which extend from the penetration for the containment sump, are embedded with their lowest points at Elevation 689.3 being slightly above the sleeves. An embedded anchor plate at the lower end of the bolts is provided to transfer the bolt load to the concrete. The design of the bolt is based on using an allowable stress of 2 x S_m. Allowable stresses in the concrete are based on a specified strength of 5000 psi.

Loads considered in the design consist of dead loads, seismic loads, and NASPL loads. The NASPL loads have been increased by 80% for the design of the anchorage.

The anchor bolts were pretensioned during construction to assure fixity of the base during an operating accident. Since the concrete is subject to creep over a period of time, the effects of creep were calculated and bolt preload was increased accordingly. The initial bolt strain was calculated based on this preload.

The embedded anchor plate is a ring designed to transfer the bolt loads to the concrete. The design assumed that the ring is discontinuous at points midway between bolts. This approach permits the butts in the ring to be unwelded.

The tensile loads in the shell are greater than the compressive loads. Since the bolts are preloaded, the effect is that the anchorage is placed in compression. As a result, the anchorage system was designed for the bolt pre-load plus the compressive shell load.

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 Margin of Safety

A certified stress report was prepared by CB&I for the vessel in accordance with the requirements of the ASME Code. This report contains several hundred pages and therefore is not included in this report.

Design values for transient pressure loads were determined by multiplying the calculated values by 1.45 as described in Section 3.8.2.3.1. In addition, the buckling criteria, in Section 5 of Appendix 3.8B, require a load factor of 1.25.

Non-pressure parts such as walkways, handrail, ladders, etc., were designed in accordance with AISC "Manual of Steel Construction," seventh edition, so that the stress in the members and welds does not exceed the allowable stress criteria as set forth in the February 1969, AISC "Specifications for Design, Fabrication, and Erection of Structural Steel for Buildings." The factor of safety of these allowable stresses with respect to specified minimum yield points of the material used are as defined in Section 1.5 of "Commentary on the Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings."

Local areas, such as the personnel and equipment hatch areas, were checked for deformations to avoid a resonant condition. The vessel as a whole was not designed to deformation limits.

Shutdowns and startups do not occur with a frequency to require a design for fatigue failure. The number of load cycles will not affect the containment vessel service life.

The stability of the containment vessel was evaluated by the criteria of Appendix 3.8B. This criteria is applicable to stiffened circular and spherical shells and independent panels. A factor of safety was used in the design related to buckling. Loading conditions which included SSE used a factor of safety of 1.1. The factor of safety for external pressure was provided by the ASME Code. The factor of safety for all other loading conditions was 1.25.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials - General

Materials for the containment vessels, including equipment access hatches, personnel access locks, penetrations, attachments, and appurtenances meet the requirements of the following specifications of the issue in effect on the date of invitation for bids. Impact test requirements were as specified in the ASME Boiler and Pressure Vessel Code, Section III for maximum test metal temperature of 0°F. Charpy V-notch specimens, SA-370, type A, were used for impact testing materials of all product forms in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. In order to provide for loss of impact properties during fabrication, all materials were either furnished with an adequate test temperature margin below the minimum NDT temperature, or the specified minimum values were effectively restored by heat treatment in accordance with ASME Code requirements.

<u>Material Designations</u>

<u>Plate for Vessels</u>

Carbon steel	SA-516, Grade 70 carbon steel plates for pressure vessels for moderate and lower temperature service.
Austenitic stainless steel	SA-240, Type 304

<u>Forgings</u>

Carbon steel	SA-350, Grade LF1 for welding
Austenitic stainless steel	SA-182, Grade F304 or 316

Carbon steel (for fittings or couplings)	SA-105, SA-181, Grade II, or SA-234, Grade WPB.
Austenitic stainless steel for fittings or couplings)	SA-403, WP316, or SA-234, Grade WPB
<u>Pipe</u>	
Carbon steel	SA-333, Grade 1 or 6, seamless, or SA-155, Grade KCF70, electric fusion-welded.
Austenitic stainless steel	SA-312, Grade TP316, seamless, SA-358, Class 1, Grade 316, electric fusion-welded.
Carbon steel (for leak chase piping and platform handrail piping)	SA-53 or SA-106
<u>Castings</u>	
Carbon steel	SA-216, Grade WCB, or SA-352, Grade-LCB
Carbon steel (for lock and hatch mechanisms)	ASTM A27, Grades 70-36
Austenitic stainless steel (for personnel lock equalizing valve bonnet, ball, and body)	SA-351, Grade CF8M
Cold finished steel (for lock and hatch mechanisms)	ASTM A108, Grades, 1018 to 1050 inclusive
Bar and machine steel (for lock and hatch mechanisms)	ASTM A576, special quality, carbon content not less than 0.30 percent.
<u>Fasteners</u>	
Carbon steel	SA-320, Grade L7 or L43; SA-193, Grade B7; or SA-194, Grade 2H or 7
Austenitic stainless steel	SA-193, Grade B8, or SA-194, Grade 8

Carbon steel (for platform bolts and nuts)	A307, Grade B

Welding Electrodes

Carbon steel	SFA-5.1, E 70 Classification Submerged Arc SFA-5.17, EL or EM; Gas metal Arc SFA-5.18, E70-S-1 through E70-S-6; Gas Tungsten Arc SFA-5.18, E70-S-1 through E70-S-6.
Austenitic stainless steel	SFA-5.4 E308 or E309 Classification; SFA-5.9, ER308 or ER309 Classification

Structural Steel

Plates, bars, and shapes (other than vessel plates)	ASTM A36, A283, Grade C, A514 Type F A537, Class 1
Plates (leak chase and built-up sections)	SA-516, Grade 70.
Plates (platform walkways and personnel lock floor plate)	Regular quality carbon steel nonskip S400

FittingsA105, A181, Grade II

Gasket materials, including O-ring seals and flexible membrane seals, shall be of Ethylene Propylene Diene Monomer (EPDM) material, Presray Type E603, E603-A or other suitable elastomers in continuous rings and with a Shore A durameter of hardness of 50-70 prior to exposure at operational conditions.

Installed seals, packages spares and replacement are to be examined after delivery. Prior to initial startup and then at 18-month intervals thereafter, the installed seals are to be examined. Visual examination is required to determine if there is any evidence of cracking which would result in establishing a leak path for air. If any cracking of the seal is observed, the seal is to be replaced.

Minimum values of seal material properties are to be as following:

	ASTM Spec	Before Exposure	After Exposure
Durometer	D2240	50-70	45-75
Min. tensile	D412	1800 psi	900 psi
Min. elongation	D412	400%	150%
Max. compression set	D095	20%	30%

Seals and gasket materials are required to withstand radiation of 10^8 Rads.

3.8.2.6.2 Corrosion Protection

Potential corrosion of the steel containment has been considered at both the embedded bottom liner in conjunction with the concrete, at the inner face in the region of the ice condenser, and at the outer face exposed to the annulus atmosphere.

The conditions which determine corrosion are basically the electro-potential of the materials involved, the presence of oxygen and an electrolyte, temperature and may induced electro-potential, from extraneous sources. These have been evaluated in the determination of corrosion.

The containment material is to specification SA-516, Grade 70, being a 1% manganese, 0.3% silicon low carbon steel, and has interfaces with concrete. Thus no unfavorable electro-potentials exist in the materials.

The climatic conditions for Chattanooga, Tennessee, show an ambient annual temperature of 0°F to 100°F [3]. The corresponding temperature for the steel containment in the region of the ice condenser are approximately 32°F to 120°F.

The corrosion of the steel containment face in contact with the containment concrete is not a design consideration since portland cement concrete provides good protection to embedded steel. The protective value of the concrete is ascribed to its alkalinity and relatively high electrical resistivity in atmospheric exposure.

Reference [4] identifies three basic conditions as being conducive to the corrosion of steel in concrete.

1. The presence of cracks extending from the exposed surface of the concrete to the steel.

2. Corrosion cells arising from electro-potential differences in the concrete itself.

3. Electrolysis by induced currents in the concrete or steel.

With respect to condition (1) the base consists of a 3-foot thick concrete embedment surrounding all the steel containment. The cracking under the worst of cases is considered minimal. This quantity far surpasses minimum cover recommended by ACI 201-1 in the most corrosive marine environment.

The potential for developing corrosion cells was kept to a minimum by limiting the soluble salts and chlorides in the concrete. Further, the continuing corrosion of iron under these conditions requires that the hydrogen deposited at the cathode is freed or combined with oxygen. Since both these mechanisms are prevented by the concrete, the corrosion cells are polarized, and the reaction is brought to a standstill.

To preclude the development of induced electric currents and in keeping with good construction practice, all electrical equipment and structures are grounded as determined by the resistivity of the foundation materials for the site. Foundation material resistivity surveys were made and the result considered in the design and determination of the extent of the grounding mat.

The seasonal variation of steel containment temperature in the region of the ice condenser gives rise to a range of relative humidity from 4% at 120°F to 45% at 32°F. This is based on saturated air leaking from the cooling ducts at a temperature of 10°F and rising to the steel containment temperature at the containment surface.

The annular region exterior to the steel containment is essentially airtight. Only during periods of shutdown during which access doors are open will this seal be broken. In the event of a pipe rupture in the annular region, water would be removed by a drainage system at the base of the annulus.

Any ingress of moisture to the interior steel containment face is prevented by sealing the outer periphery of the ice condenser adjacent to the steel containment, and by the vapor barrier on the inside face of the duct panels at the boundary of the ice bed. In the event of any abnormal ingress of moisture through the seal, the leakage air from the cooling ducts has the capacity to absorb moisture up to the limits of the relative humidities quoted above. In addition, any moisture remaining will have a tendency to migrate to the colder end of the temperature gradient; i.e., for all steel containment temperatures above 10°F, moisture will migrate towards the cooling air ducts, where it will be evaporated as the cooling air increases in temperature in the course of its passage through the ducts.

For steel containment temperatures below 32°F any moisture at the steel containment face will be frozen, this condition pertaining to relative humidities greater than 45% and steel containment temperatures below 10°F when the migration of moisture could take place from the air cooling ducts to the steel containment.

In the event of actuation of the containment spray, water would be applied to the interior surface of the steel containment. Most of the water would be removed by the drainage system and the small amount of moisture remaining would be removed from the steel containment surface by evaporation.

Several references have been established which give corrosion data for the limits of the conditions described above.

For low alloy steels in any industrial atmosphere long-term tests indicate a maximum total corrosion of 0.016 inch in 40 years (based on 14g/sq dm in 18 years[5]).

For dry inland conditions which more closely simulate the steel containment conditions the total corrosion for the plant lifetime is approximately 0.010 inch.[7] This is accounted for by the fact that below relative humidity of 65%, iron oxide itself forms an adherent film affording good protection to further corrosion.[6][8] Furthermore, at temperatures below freezing, ion transport in the electrolyte is almost entirely inhibited, obviating the mechanisms of corrosion.[9]

It is concluded that the maximum total corrosion for any exposed internal surface of the steel containment in the region of the ice condenser is 0.010 to 0.015 inch over the lifetime of the plant. In general, the corrosion in the region of the ice condenser is expected to be less than in other areas of the containment, which can be readily inspected.

3.8.2.6.3 Protective Coatings

Protective coatings were applied to all exposed steel surfaces of the containment vessel. Surfaces embedded in concrete will not be coated. For coating systems used on the inside of the containment, see Section 6.1.2.

As part of the steam generator replacement, two openings in the dome of the containment vessel were created. To support cutting of the openings and reinstallation of the cut section of the containment vessel, the protective coatings on both sides of the containment vessel near the opening cut lines were removed. Areas where the coating on the outside (annulus side) of the containment vessel was removed were recoated following completion of welding, NDE and pressure testing of the containment vessel. Areas where the coating on the inside of the containment vessel was removed were not recoated. These uncoated areas will be periodically inspected as part of the containment in-service inspection program to verify that unacceptable amounts of corrosion of the containment vessel have not occurred.

[Historical Information - All exterior vessel shell surfaces and metal surfaces of platforms, floor plate, ladders, walkways, attachments, and accessories located in the annular space surrounding the containment vessel were cleaned in accordance with the requirements of Steel Structures Painting Council Surface Preparation Specification No. 6, Commercial Blast Cleaning, latest edition. After cleaning and having passed inspection, one complete prime shop coat of Carboline Carbozinc 11 paint (dry film thickness was not less than 2-1/2 mils) was applied in accordance with the manufacturer's instructions.]

[Historical Information - All interior surfaces of the containment vessel shell and metal surfaces of attachments thereto, except those parts embedded in the base slab and identified as the liner and areas within 2 inches of field-welded joints, were given one prime coat of Carboline Carbozinc 11 within 8 hours after blast cleaning in accordance with Steel Structures Painting Council Surface Preparation Specification No. 10, Near-White Blast Cleaning, latest edition. The primer was top-coated by TVA field forces with an epoxy coating as recommended. The surfaces of the vessel in the annular space were coated with materials selected for the ability to provide protection against atmospheric corrosion.]

3.8.2.6.4 Tolerances

The containment vessel as constructed does not exceed the applicable tolerance requirements of the ASME Code for fabrication or erection.

The out-of-roundness tolerance does not exceed 0.5% of the nominal inside diameter.

The deviation from a vertical line of the vertical cylindrical portion adjacent to the ice condensers is limited to ±2 inches for the height of the ice condensers.

Threaded studs for attachment of ice condenser outer duct panels do not vary from their theoretical location by more than ±1/4 inch.

Penetrations do not vary from their theoretical location by more than ±1/2 inch.

3.8.2.6.5 Vessel Material Inspection and Test

ASTM standard test procedures were employed for the liner and shell plates to ascertain compliance with ASTM specifications. Certified copies of mill test reports of the chemical and physical properties of the steel were submitted to TVA for approval. Tests for qualifying welding procedures and welders were also submitted for approval. All vessel pressure boundary material was tested (one test for each heat of steel) to determine its Nil Ductility Transition Temperature (NDTT). These tests were conducted to meet the requirements of ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB-2300. The tests were conducted at a maximum temperature of 0° F.

Ultrasonic inspection was required for all pressure boundary plates subjected to tensile forces normal to the plate surface. This inspection was performed in accordance with ASME Boiler and Pressure Vessel Code, Section III, NB-2530.

3.8.2.6.6 Impact Testing

Charpy V-notch impact tests were made of material, weld deposit and the base metal weld heat affected zone employing a test temperature of not more than 30°F below minimum operating temperature. The requirements of the ASME Code, Paragraph NB-2300, were met for all materials under jurisdiction of the code. All weld procedure qualifications for procedures used on the containment vessel shell also meet code requirements for ductility.

3.8.2.6.7 Post-Weld Heat Treatment

Field welded joints did not exceed 1½ inches and therefore, the containment vessel as a completed structure did not require field stress relieving. Insert plates at penetration openings did not exceed 1½ inches in thickness and stress relieving was not required by ASME Code before or after they were welded to adjacent plates. Post-weld heat treatment, where required, was performed as required by and in accordance with the ASME Code.

3.8.2.6.8 Welding

All welding procedures were qualified under provisions of Part A of Section IX of the ASME Code. Welding procedures were submitted to TVA for approval before welding was started. All welding was performed by welders qualified in accordance with Part A of Section IX of the ASME Code.

3.8.2.7 Testing and Inservice Inspection Requirements

3.8.2.7.1 Bottom Liner Plates Test - *Historical Information*

Before concrete was placed over the bottom liner, the leak tightness of this liner was verified. All liner plate welds were vacuum box tested for leak tightness. Upon completion of a successful leak test, the welds were covered with channels, and the channels were leak tested by pressurization to 15 psig.

3.8.2.7.2 Vertical Wall and Dome Tests - *Historical Information*

Welds in the cylinder wall and dome in ASME Code Section III, Categories A and B, were 100% radiographed. Welds in Categories C and D were examined by magnetic particle, liquid penetrant, or by ultrasonic methods.

3.8.2.7.3 Soap Bubble Tests - Historical Information

Upon completion of the construction of the containment vessel, a soap bubble test was conducted with the vessel pressurized to 5 psig. Soap solution was applied to all weld seams and gaskets, including both doors of the personnel airlocks.

A second soap bubble inspection test was made at 13.5 psig upon completion of the overpressure test in accordance with the requirements of the ASME Code.

Any leaks detected by soap bubble test which could affect the integrity of the vessel or which could result in excessive leakage during the leakage rate tests were repaired prior to proceeding with the tests.

3.8.2.7.4 Overpressure Tests - *Historical Information*

After successful completion of the initial soap bubble test, a pneumatic pressure test was made on the containment vessel and each of the personnel airlocks at a pressure of 16.9 psig. Both the inner and the outer doors of the personnel airlocks were tested at this pressure. The test pressure in the containment vessel was maintained for not less than 1 hour.

3.8.2.7.5 Leakage Rate Test - *Historical Information*

Following the successful completion of the soap bubble and overpressure tests a leakage rate test at 15 psig pressure was performed on the containment vessel with the personnel airlock inner doors closed.

CB&I performed the leak rate testing by the "Absolute Method," which consists of measuring the temperature, pressure, and humidity of the contained air, and making suitable corrections for changes in temperature and humidity.

Equipment and instruments were calibrated and certified before any pressure tests were initiated.

Continuous hourly readings were taken until it was satisfactorily shown that the total leakage during a consecutive 24 hour period did not exceed 0.1% of the total contained weight of air at test pressure at ambient temperature in accordance with the requirements of 10 CFR 50, Appendix J.

CB&I reviewed the leakage rate data during the test to determine adequacy of the test, authorize termination, or require continuation of the test.

3.8.2.7.6 Operational Testing - *Historical Information*

After completion of the airlocks, including all latching mechanisms, interlocks, etc., each airlock was given an operational test consisting of repeated operation of each door and mechanism to determine whether all parts are operating smoothly without binding or other defects. All defects encountered were corrected and retested. The process of testing, correcting defects, and retesting was continued until no defects were detectable.

3.8.2.7.7 Leak Testing Airlocks - *Historical Information*

The airlocks were pressurized with air to 16.9 psig. All welds and seals were observed for visual signs of distress or noticeable leakage. The airlock pressure was then reduced to 13.5 psig, and a thick soap solution was applied to all welds and seals and observed for bubbles or dry flaking as indications of leaks. Leaks and questionable areas were clearly marked for identification and subsequent repair. During the overpressure testing, the inner door was locked with hold-down devices to prevent upsetting of the seals.

The internal pressure of the airlock was reduced to atmospheric pressure and all leaks repaired after which the airlock was again pressurized to 13.5 psig with air and all areas suspected or known to have leaked during the previous test were retested by above soap bubble technique. This procedure was repeated until no leaks were discernible by this means of testing.

3.8.2.7.8 Penetration Tests - Historical Information

Type B tests were performed on all penetrations with test bellows and/or pressure taps in accordance with the requirements of 10 CFR 50, Appendix J. See Section 6.2.6 for imposed leak rates and tests performed on penetrations.

3.8.2.7.9 Inservice Inspection Requirements

3.8.2.7.9.1 Components Subject to Examination and/or Test

All ASME Code Class MC and metallic liners of Code Class CC components shall be examined and tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and as required by 10 CFR 50.55a(b)(2)(x), 50.55a(g)(4), and 50.55a(g)(6)(ii)(B), except where specific written relief has been requested. The inservice inspection requirements are contained in Section 3.8.5.1.1 for ASME Code Class CC concrete components. The inservice inspection requirements are contained in Section 5.2.8 for ASME Code Class 1 components and Section 6.6 for ASME Code Class 2 and 3 components. Inservice leakage rate tests and inservice surveillance of the containment vessel are discussed in Section 6.2.6.

3.8.2.7.9.2 Accessibility

Watts Bar design was established prior to the publication of Subsection IWE of ASME Section XI, however, accessible Class MC and metallic liners of Class CC components will be inservice examined in accordance with the guidelines of Subsection IWE of ASME Section XI.

3.8.2.7.9.3 Examination Techniques and Procedures

The examination procedures used by TVA are performed in accordance with the guidelines of Subarticle IWA-2200 of ASME Section XI.

3.8.2.7.9.4 Inspection Intervals

An inspection schedule for Class MC components will be developed in accordance with Subarticle IWE-2400 of ASME Section XI and the requirements of 10 CFR 50.55a(g)(6)(ii)(B).

3.8.2.7.9.5 Examination Categories and Requirements

The examination categories and requirements for Class MC components will be in accordance with Subsection IWE of ASME Section XI and 10 CFR 50.55a(b)(2)(x) to the extent practicable.

3.8.2.7.9.6 Evaluation of Examination Results

Evaluation of examination results shall be in accordance with Article IWE-3000 of ASME Section XI. Components with unacceptable indications will be repaired or replaced in accordance with Article IWA-4000 of ASME Section XI.

3.8.2.7.9.7 System Pressure Tests

The program for Class MC and metallic liners of Class CC components system pressure tests shall be in accordance with article IWE-5000 of ASME Section XI.

REFERENCES

1. Kalnins, A., "Analysis of Shells of Revolution Subjected to Symmetrical and Non-Symmetrical Loads," Trans. of the ASME Journal of Applied Mechanics, September, 1964, pp. 467-476.

2. Bijlaard, P. P., "Stresses from Radial Loads and External Moments in Cylindrical Pressure Vessels," Welding Journal 34(12), 1955.

3. American Society of Heating, Refrigeration and Air Conditioning Engineers, Handbook of Fundamentals, 1967.

4. American Concrete Institute Proceedings, Volume 59 Report No. 201, December 1962, H. R., "Durability of Concrete in Service."

5. "Long-Time Atmospheric Corrosion Tests on Low Alloy Steels," H. R. Copson, American Society for Testing Materials Proceedings, Volume 60, 1960, pp. 650-666.

6. "Corrosion," Metals Handbook, Volume 13, ninth edition, Metals Park, Ohio, 1987, pp 82-83.

7. Lauobe, C. P., "Corrosion of Steel in Marine Atmospheres," Trans. Electro Chemical Society, Volume 87, 1945, pp. 161-182.

8. Rozenfield, I. L., "Atmospheric Corrosion of Metals," Houston, TX: NACE, 1973, pp 104-106.

9. Evans, Ulrick R., The Corrosion and Oxidation of Metals: Scientific Principles and Practical Applications. London, Edward Arnold (Publishers) Ltd., 1960, pp. 27-37.

10. TVA drawings 47E235-44 through 48, "Containment Harsh Environment."

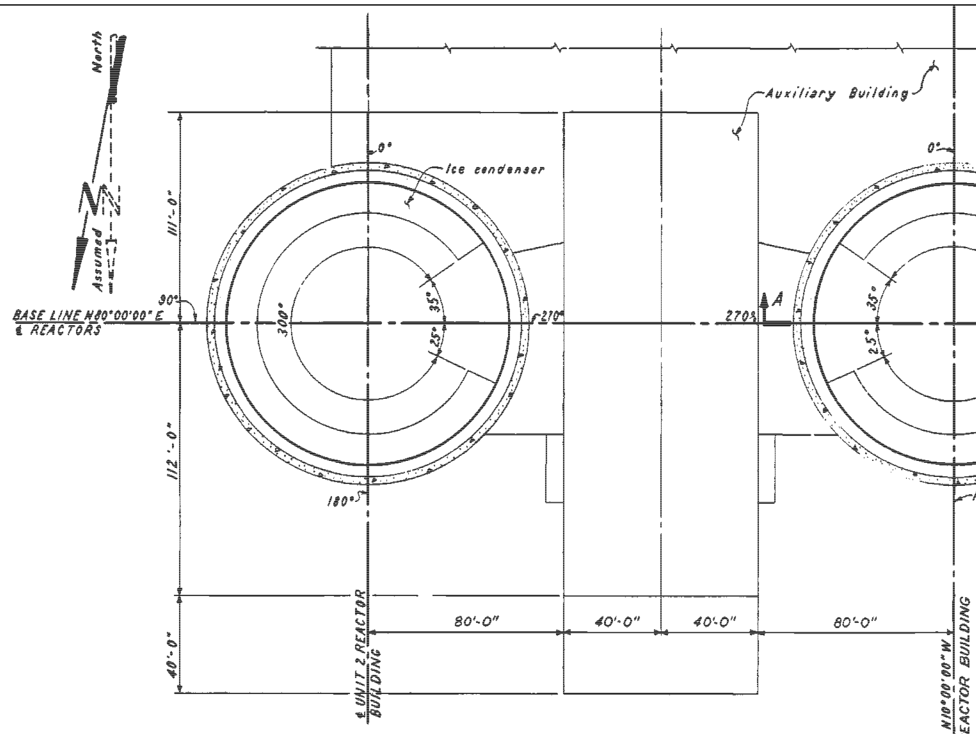
TABLES 3.8.2-1

ALLOWABLE STRESS CRITERIA - CONTAINMENT VESSEL

Material: SA-516, Grade 70

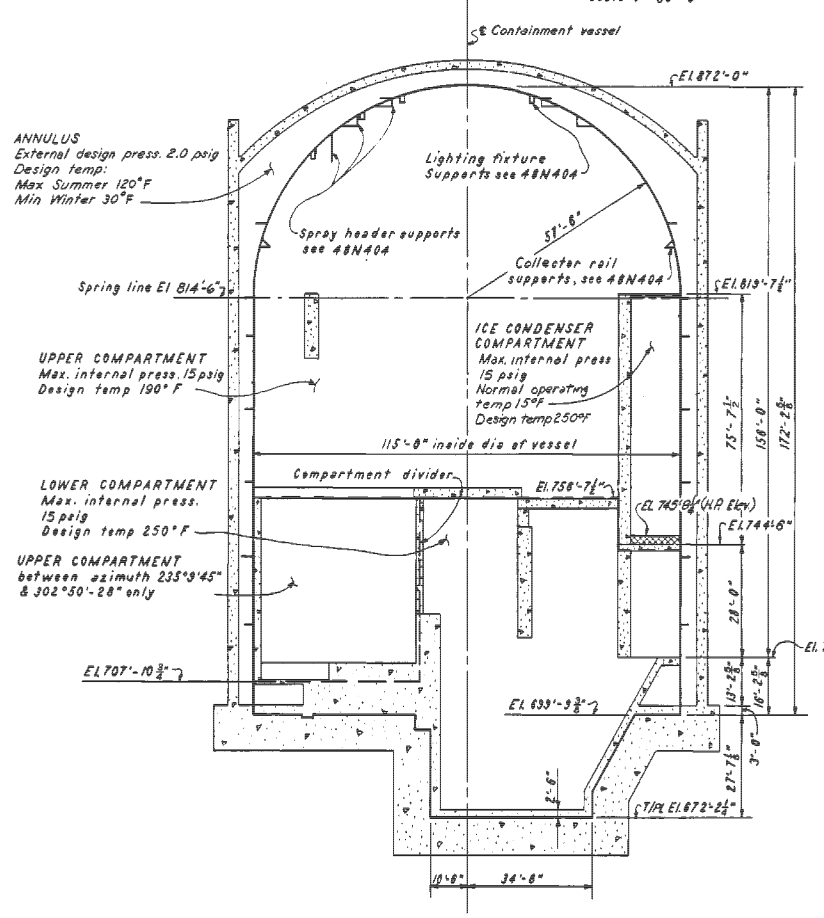
Loading Conditions	Applicable ASME Code Reference for Stress Intensity
1. Normal Design Condition	NB-3221
2. Normal Operation Condition	NB-3222
3. Upset Operation Condition	NB-3223
4. Emergency Operation Condition	NB-3224
5. Construction Condition	NB-3221
6. Test Condition	NB-3226
7. Post-Accident Fuel Recovery Condition	NB-3224

[1]All references are to the ASME Boiler and Pressure Vessel Code, 1971 Edition, Section III.



PLAN

Scale 1" = 30'-0"

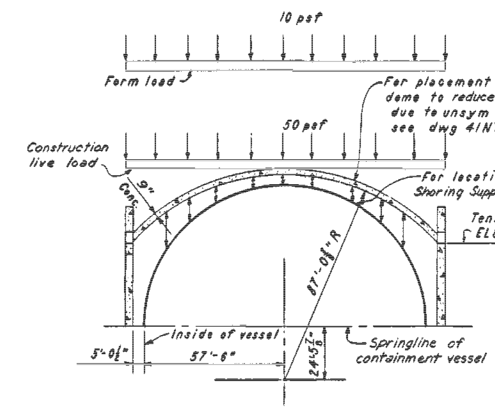


SECTION A-A

DESIGN PRESSURES & TEMPERATURES

UNIT 1 AS SHOWN

UNIT 2 OPP. HAND



CONSTRUCTION LOADS

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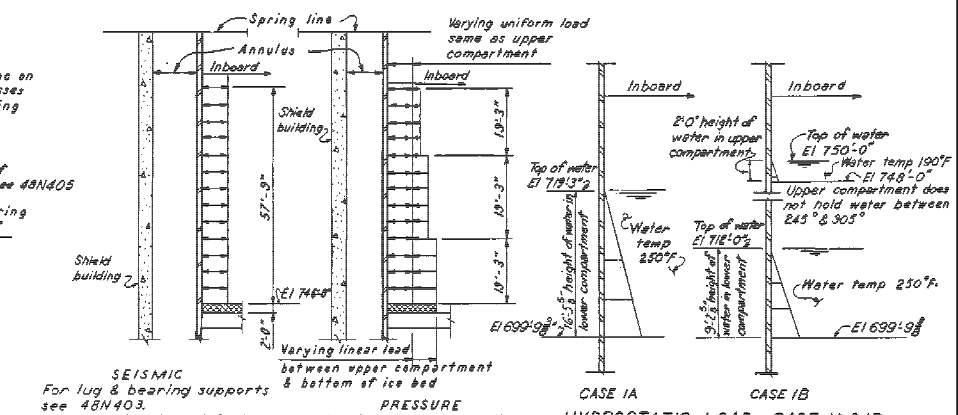


SEISMIC

For lug & bearing supports see 48N403.

Each lug shall be subjected to a maximum load of 200# in tension and 200# in shear due to the Design Basis Earthquake.

PRESSURE

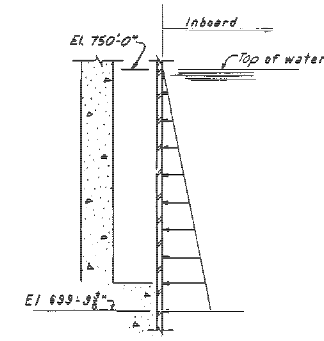
For bearing supports for linear load see 48N403. For loads see press. transients in specs.

ICE CONDENSER DUCT LOADS

HYDROSTATIC LOAD - CASE IA & IB

Hydrostatic load occurs when pressure in both upper and lower compartments is 15 psi and temp is 190°F in the upper compartment and 250°F in the lower compartment. Allowable stress shall conform to loading condition 3 or 4 of table I in appendix A of the specification.

N.T.S.



HYDROSTATIC LOAD - CASE II

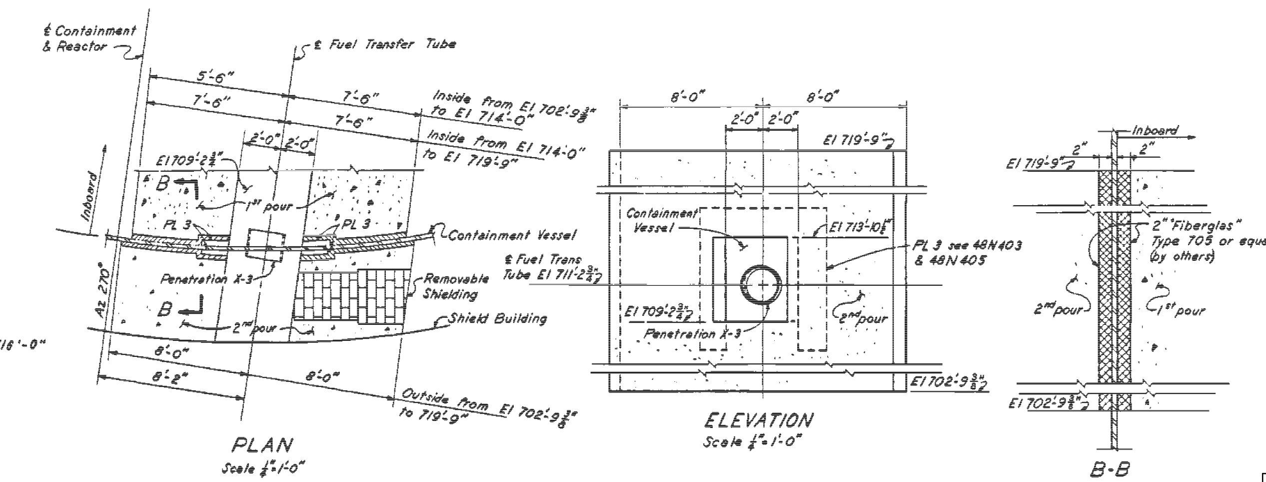
Hydrostatic load occurs when pressure and temp in containment vessel and annulus are at normal operating conditions. Earthquake not considered for this case.

N.T.S.

NOTES CONTINUED:

SEE DRAWING 48W428-1 THRU 48W428-9 FOR ADDITIONAL INFORMATION.

FOR COMPOSITE DWGS OF STIFFENERS PENETRATIONS AND PAD PLATES SEE DWG 48W428-1 THRU 9.



CONCRETE PLACEMENT LOAD AT FUEL TRANSFER TUBE

Allowable stress shall conform to table I appendix A of the specification

N.T.S.

Scale 1" = 20'

Except as noted

NOTES:

This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N400 R4 except as noted by encirclements on the R0 issue. Elevations not encircled on R0 issue are Sequoyah Reactor Building elevations raised by plus 23'-0". Encircled elevations on R0 issue have been changed otherwise to satisfy design requirements.

For general notes see 48N401.

Wind load applied only over the dome shape above the tension ring of the shield structure.

For personnel access lock live loads see 44N240

For seismic loads and pressure transients see specifications.

Walkway live load of 50 psf and cable tray weight of 225# per linear feet applied on walkways. For location of cable trays see 48N407 Walkways at El 724'-6" thru El 814'-6" are to be designed as circumferential stiffeners.

Vessel to be designed for external pressure of 2.0 psig

Temp shown inside the containment vessel are design based accident temp encountered by containment vessel walls.

Shell temperature transients are shown in specifications.

Cold shut down temperature ranges from 30° to 120°F.

Normal operation temperature ranges from 60° to 120°F.

No live load will be applied to the equipment access hatch.

Dead load of ice condenser duct is supported by the internal concrete structure.

For seismic and pressure duct loads see dwgs & specs

Penetrations shall be designed and reinforced such that under maximum combination of loads, no part of the penetration or shell will exceed the allowable stress indicated in the specifications.

All welds to be examined in accordance with Section III of ASME Boiler and Pressure Vessel Code, paragraph NE-5120.

NORMAL OPERATING CONDITIONS:

INSIDE THE CONTAINMENT VESSEL
- Max. pressure 0.3 psig.
- Upper compartment shell temperature 110°F.
- Lower compartment shell temperature 120°F.
- Ice compartment shell temperature 60°F.

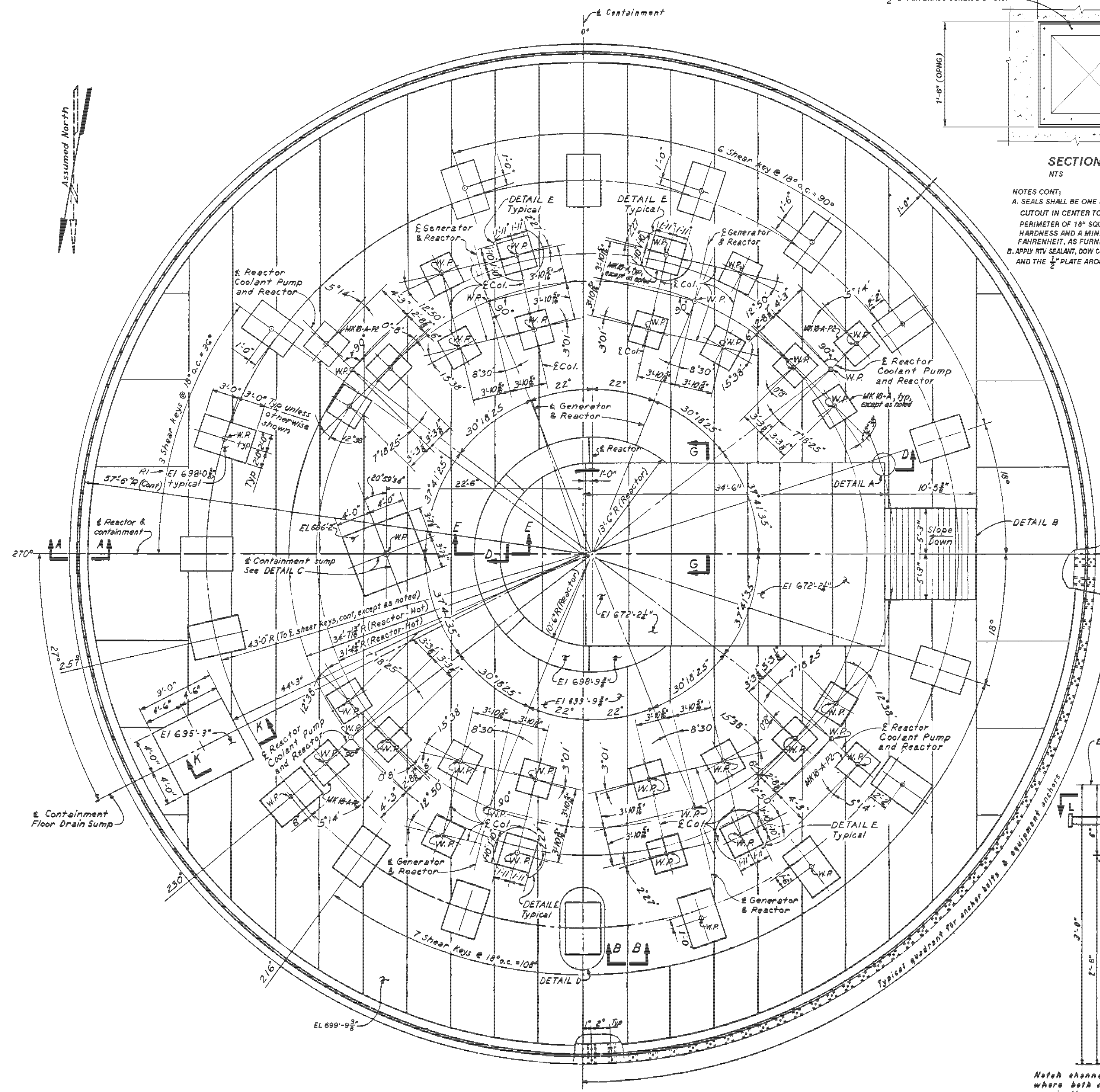
ANNULUS
- Pressure 0 psig.
- Temperature 60° to 120°F.

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
STRUCTURAL STEEL
CONTAINMENT VESSEL
DESIGN DATA SHEET
TVA DWG NO. 48N400 RD
FIGURE 3.8.2-1

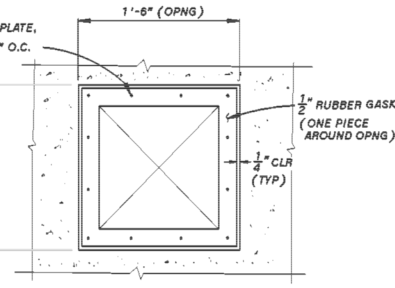
CAD MAINTAINED DRAWING



PLAN-BOTTOM LINER PLATE

UNIT 1 AS SHOWN
UNIT 2 OPP HAND

For leakage Lo & Co see sections & details



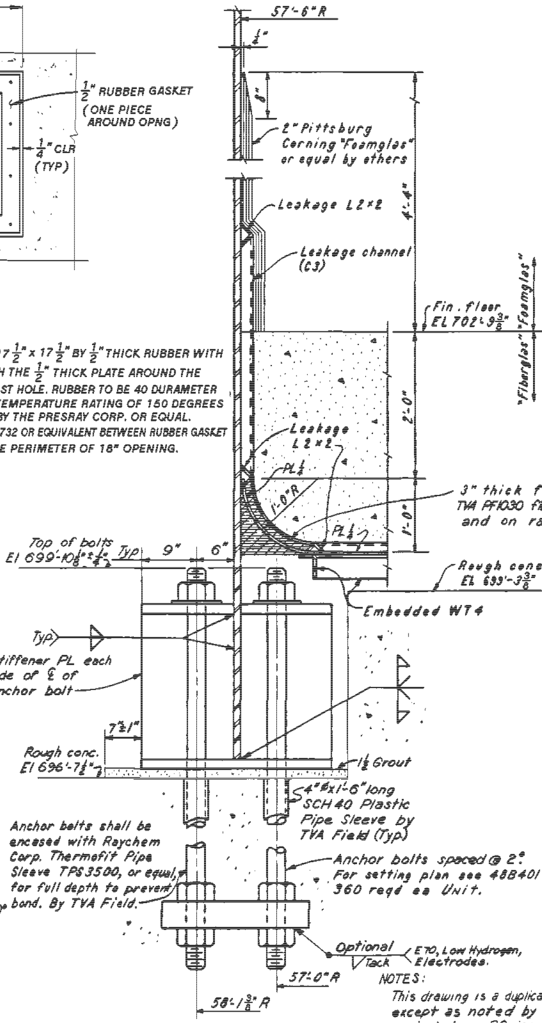
SECTION L-L

NTS

NOTES CONT;

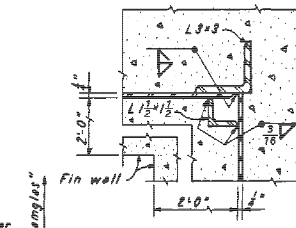
A. SEALS SHALL BE ONE PIECE 17 ½" x 17 ½" BY ½" THICK RUBBER WITH CUTOUT IN CENTER TO MATCH THE ½" THICK PLATE AROUND THE PERIMETER OF 18" SQUARE TEST HOLE. RUBBER TO BE 40 DURAMETER HARDNESS AND A MINIMUM TEMPERATURE RATING OF 150 DEGREES FAHRENHEIT, AS FURNISHED BY THE PRESRAY CORP. OR EQUAL.

B. APPLY RTV SEALANT, DOW CORNING 732 OR EQUIVALENT BETWEEN RUBBER GASKET AND THE ½" PLATE AROUND THE PERIMETER OF 18" OPENING.



SECTION A-A

Scale 1" = 1'-0"



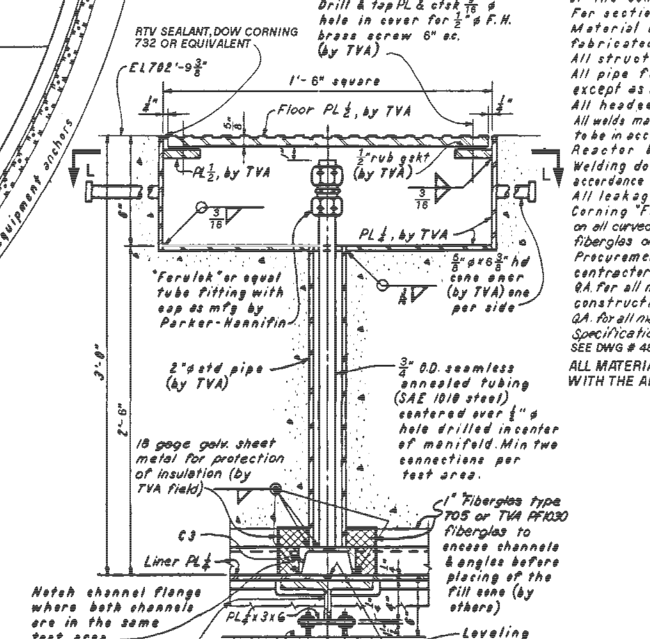
DETAIL A

TYPICAL CORNER

Scale 3" = 1'-0"

NOTES:

This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N401 R5 except as noted by encirclements on R0 issue. Elevations not encircled on R0 issue are Sequoyah Reactor Building elevations raised by plus 23'-0". Encircled elevations on R0 issue have been changed otherwise to satisfy design requirements.
W.P. denotes work point.
Contractor shall divide floor liner plate into four independent test areas.
All elevations and dimensions are computed to the inside of the containment shell, unless noted.
For sections and details not shown see 48N402.
Material by TVA denotes material to be furnished and fabricated by TVA field.
All structural steel furnished by TVA field to be ASTM A-36.
All pipe furnished by TVA field to be ASTM A120 or A53, except as noted.
All headed concrete anchors to be ASTM A108.
All welds made by TVA on items other than pressure boundary parts to be in accordance with general construction specification G-29.
Reactor building is a class I structure
Welding documentation and inspection by TVA field to be in accordance with construction specification N3G-881.
All leakage channels and angles to be encased with Owens Corning "Fiberglas" or equal. Use type 703 or TVA PF1030 fiberglas on all curved or irregular shapes and use type 705 or TVA PF1030 fiberglas on all flat surfaces.
Procurement, fabrication and documentation of material by contractor shall conform to the TVA QA plan.
QA for all material by TVA field to be quality level I per WBNP construction specification N3G-881 EAN.
QA for all material by TVA field to be quality level II per WBNP construction Specification N3G-881 (48N401 only).
SEE DWG # 48W428-1 THRU 48W428-9 FOR ADDITIONAL INFORMATION.
ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.



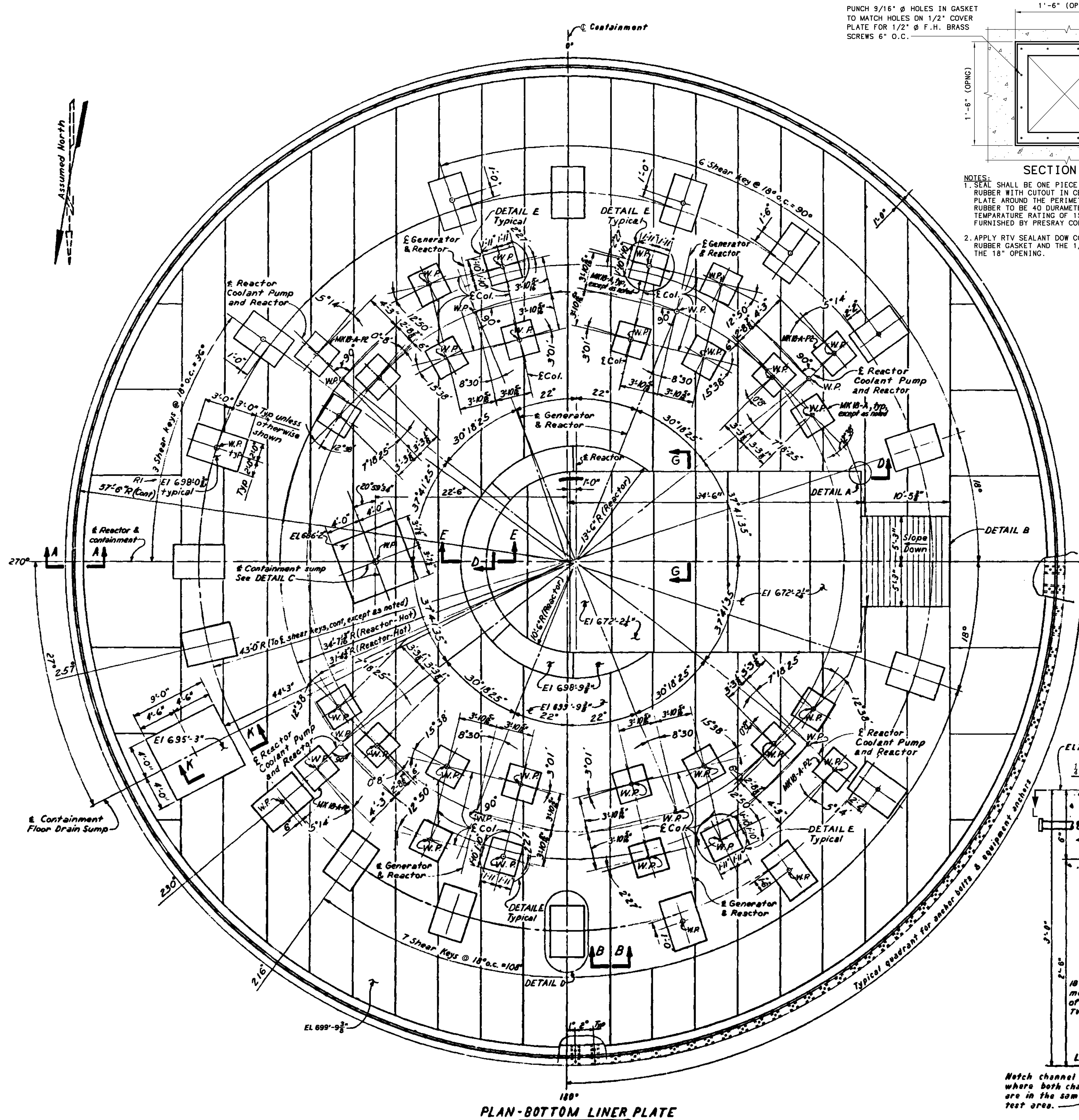
TYPICAL PLATE JOINT AND TEST CONNECTION Scale 1 3 = 1'-0"

REFERENCE DWG: 48B401

Scale: ⅛" = 1'-0" UON

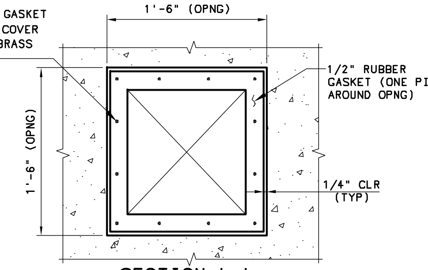
WATTS BAR FINAL SAFETY ANALYSIS REPORT
REACTOR BUILDING UNIT 1 STRUCTURAL STEEL CONTAINMENT VESSEL ANCHOR BOLT PLAN & BASE DETS SH-1 TVA DWG NO. 48N401 RG FIGURE 3.8.2-2

CAD MAINTAINED DRAWING



PLAN-BOTTOM LINER PLATE
UNIT 2 OPP HAND
For leakage Ls & Cs see sections & details

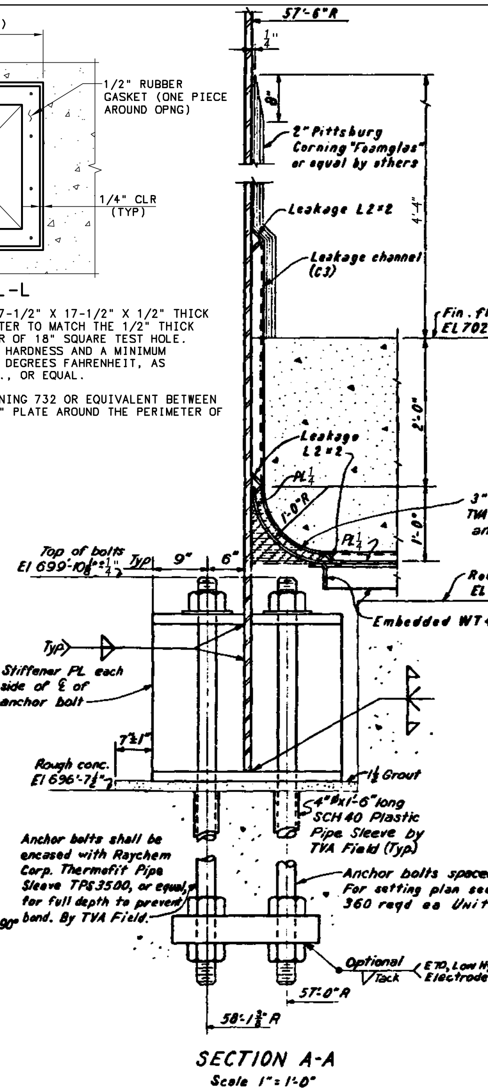
PUNCH 9/16" Ø HOLES IN GASKET TO MATCH HOLES ON 1/2" COVER PLATE FOR 1/2" Ø F.H. BRASS SCREWS 6" O.C.



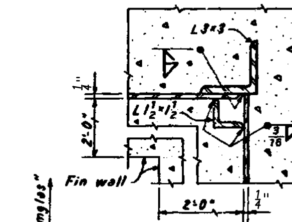
SECTION L-L

NOTES:
1. SEAL SHALL BE ONE PIECE 17-1/2" X 17-1/2" X 1/2" THICK RUBBER WITH CUTOUT IN CENTER TO MATCH THE 1/2" THICK PLATE AROUND THE PERIMETER OF 18" SQUARE TEST HOLE. RUBBER TO BE 40 DURAMETER HARDNESS AND A MINIMUM TEMPARATURE RATING OF 150 DEGREES FAHRENHEIT, AS FURNISHED BY PRESRAY CORP., OR EQUAL.

2. APPLY RTV SEALANT DOW CORNING 732 OR EQUIVALENT BETWEEN RUBBER GASKET AND THE 1/2" PLATE AROUND THE PERIMETER OF THE 18" OPENING.



SECTION A-A
Scale 1" = 1'-0"



DETAIL A
TYPICAL CORNER
Scale 3" = 1'-0"

NOTES:
ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N401 R5 except as noted by encirclements on R0 issue. Elevations not encircled on R0 issue are Sequoyah Reactor Building elevations raised by plus 23'-0". Encircled elevations on R0 issue have been changed otherwise to satisfy design requirements.
W.P. denotes work point.
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All elevations and dimensions are computed to the inside of the containment shell, unless noted
For sections and details not shown see 48N402
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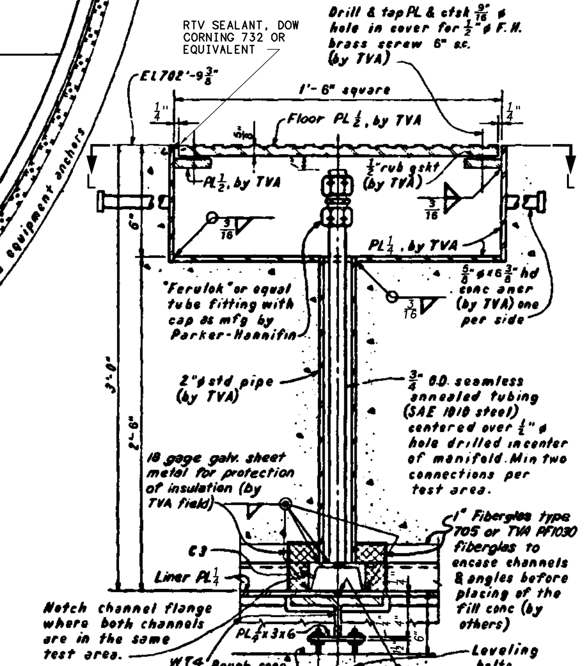
All structural steel furnished by TVA field to be ASTM A-36.
All pipe furnished by TVA field to be ASTM A120 or A53, except as noted.
All headed concrete anchors to be ASTM A108
All welds made by TVA on items other than pressure boundary parts to be in accordance with general construction specification G-29.
Reactor building is a class I structure
Welding documentation and inspection by TVA field to be in accordance with construction specification N3G-881.
All leakage channels and angles to be encased with Owens Corning "Fiberglas" or equal. Use type 703 or TVA PF103C fiberglas on all curved or irregular shapes and use type 705 or TVA PF1030 fiberglas on all flat surfaces.
Procurement, fabrication and documentation of material by contractor shall conform to the TVA QA plan.
QA for all material by TVA field to be quality level I per WBNP construction specification N3G-881 EAN.
QA for all material by TVA field to be quality level II per WBNP construction Specification N3G-881 (48N401 only).



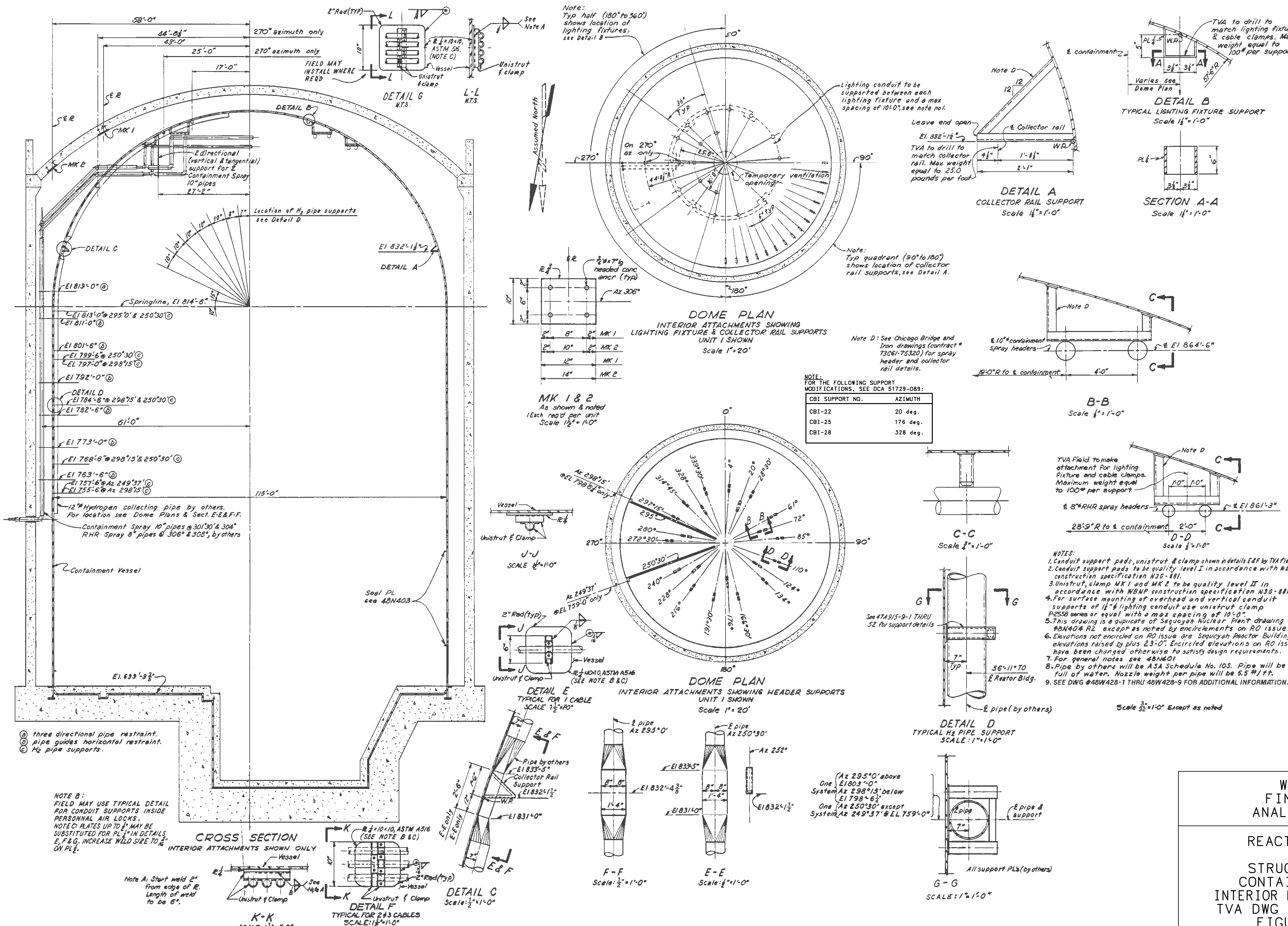
B-B
TYPICAL PLATE JOINT AND TEST CONNECTION Scale: 3" = 1'-0"

REFERENCE DWG: 48B401

FOR HATCHED AREAS SEE UNIT 1 AC DRAWING 48N401

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
STRUCTURAL STEEL
CONTAINMENT VESSEL
ANCHOR BOLT PLAN & BASE DETS SH-1
TVA DWG NO. 2-48N401 R1
FIGURE 3.8.2-2(U2)



NOTE:
FOR THE FOLLOWING SUPPORT MODIFICATIONS, SEE DCA 51729-089:

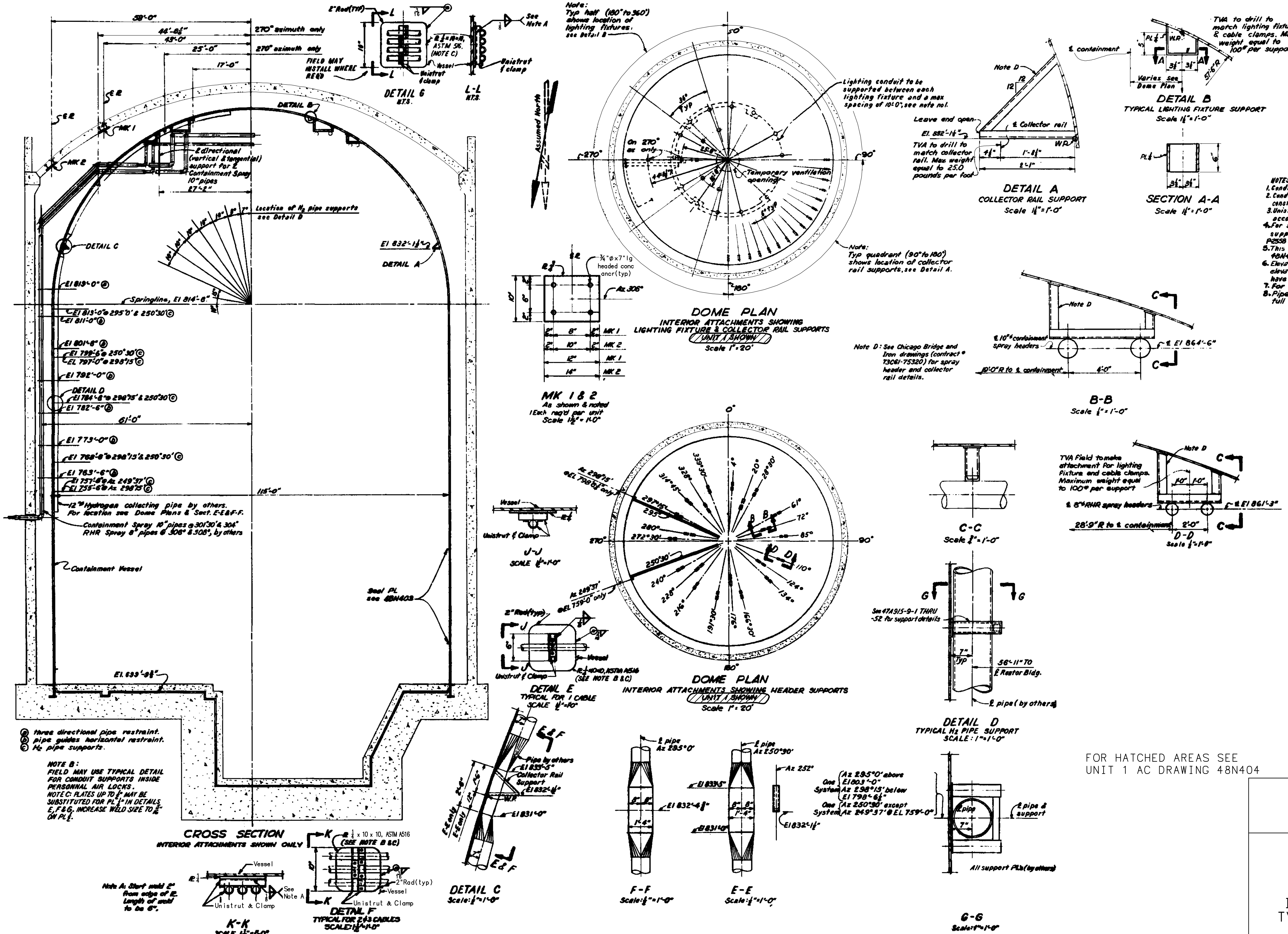
CBI SUPPORT NO.	AZIMUTH
CBI-22	20 deg.
CBI-25	176 deg.
CBI-28	328 deg.

NOTES:
1. Conduit support pads, unistrut & clamp shown in details E & F by TVA field.
2. Conduit support pads to be quality level I in accordance with WBNP construction specification N3G-881.
3. Unistrut, clamp MK 1 and MK 2 to be quality level II in accordance with WBNP construction specification N3G-881.
4. For surface mounting of overhead and vertical conduit supports of 1 1/2"Ø lighting conduit use unistrut clamp P-2558 series or equal with a max spacing of 10'-0".
5. This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N404 R2 except as noted by encirclements on R0 issue.
6. Elevations not encircled on R0 issue are Sequoyah Reactor Building elevations raised by plus 23'-0". Encircled elevations on R0 issue have been changed otherwise to satisfy design requirements.
7. For general notes see 48N401
8. Pipe by others will be ASA Schedule No. 10S. Pipe will be full of water. Nozzle weight per pipe will be 5.5#/ft.
9. SEE DWG #48W428-1 THRU 48W428-9 FOR ADDITIONAL INFORMATION.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
STRUCTURAL STEEL
CONTAINMENT VESSEL
INTERIOR ELEVATION SHEET 2
TVA DWG NO. 48N404 RE
FIGURE 3.8.2-3

CAD MAINTAINED DRAWING



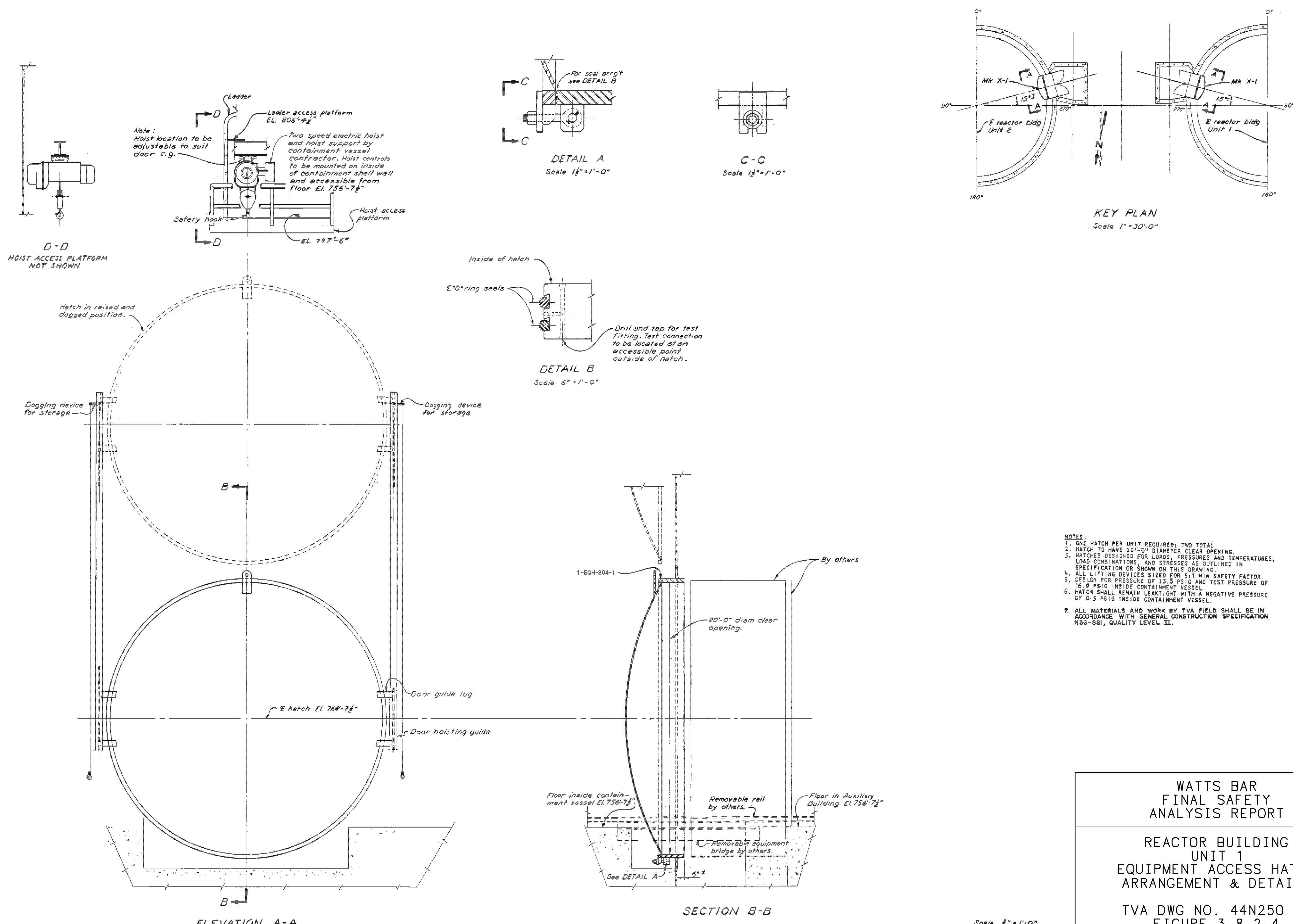
NOTES:
1. Conduit support pads, unistrut & clamp shown in details E&F by TVA Field.
2. Conduit support pads to be quality level I in accordance with WBNP construction specification N3G-881.
3. Unistrut, clamp MK 1 and MK 2 to be quality level II in accordance with WBNP construction specification N3G-881.
4. For surface mounting of overhead and vertical conduit supports of 1½"ϕ lighting conduit use unistrut clamp P2558 series or equal with a max spacing of 10'-0".
5. This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N404 R2 except as noted by encirclements on R0 issue.
6. Elevations not encircled on R0 issue are Sequoyah Reactor Building elevations raised by plus 23'-0". Encircled elevations on R0 issue have been changed otherwise to satisfy design requirements.
7. For general notes see 48N401
8. Pipe by others will be ASA Schedule No. 10S. Pipe will be full of water. Nozzle weight per pipe will be 5.5 #/ft.

FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 48N404

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
STRUCTURAL STEEL
CONTAINMENT VESSEL
INTERIOR ELEVATION SHEET 2
TVA DWG NO. 2-48N404 R0
FIGURE 3.8.2-3(U2)

CAD MAINTAINED DRAWING



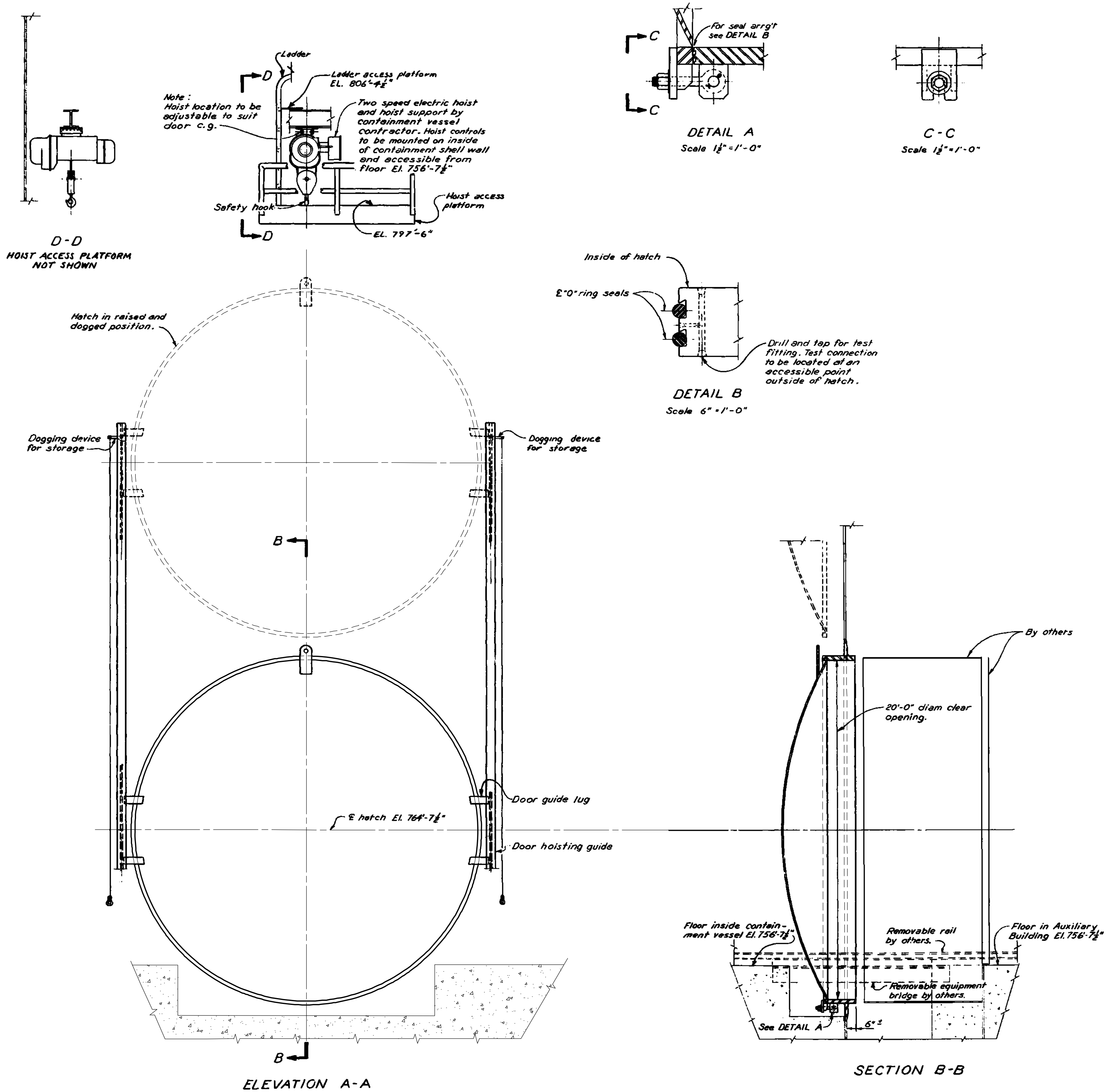
NOTES:
1. ONE HATCH PER UNIT REQUIRED; TWO TOTAL
2. HATCH TO HAVE 20'-0" DIAMETER CLEAR OPENING.
3. HATCHES DESIGNED FOR LOADS, PRESSURES AND TEMPERATURES, LOAD COMBINATIONS, AND STRESSES AS OUTLINED IN SPECIFICATION OR SHOWN ON THIS DRAWING.
4. ALL LIFTING DEVICES SIZED FOR 5:1 MIN SAFETY FACTOR
5. DESIGN FOR PRESSURE OF 13.5 PSIG AND TEST PRESSURE OF 16.9 PSIG INSIDE CONTAINMENT VESSEL.
6. HATCH SHALL REMAIN LEAKTIGHT WITH A NEGATIVE PRESSURE OF 0.5 PSIG INSIDE CONTAINMENT VESSEL.
7. ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

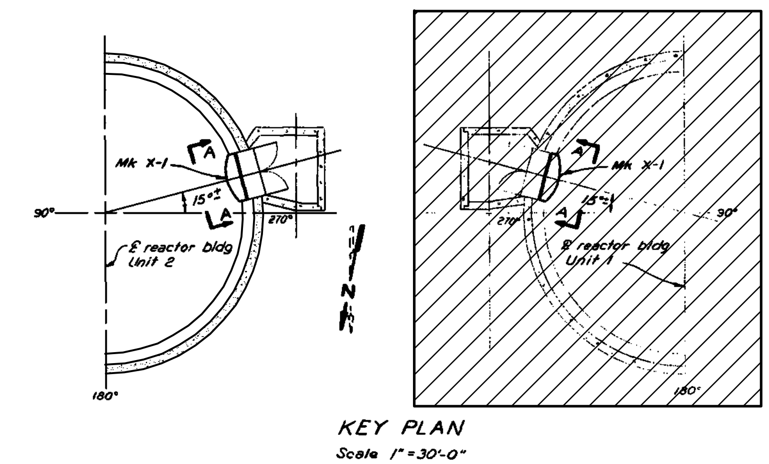
Scale ⅜" = 1'-0"
Except as noted

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
EQUIPMENT ACCESS HATCH
ARRANGEMENT & DETAILS

TVA DWG NO. 44N250 RB
FIGURE 3.8.2-4





NOTES:
1. ONE HATCH PER UNIT REQUIRED; TWO TOTAL
2. HATCH TO HAVE 20'-0" DIAMETER CLEAR OPENING.
3. HATCHES DESIGNED FOR LOADS, PRESSURES AND TEMPERATURES, LOAD COMBINATIONS, AND STRESSES AS OUTLINED IN SPECIFICATION OR SHOWN ON THIS DRAWING.
4. ALL LIFTING DEVICES SIZED FOR 5:1 MIN SAFETY FACTOR.
5. DESIGN FOR PRESSURE OF 13.5 PSIG AND TEST PRESSURE OF 16.9 PSIG INSIDE CONTAINMENT VESSEL.
6. HATCH SHALL REMAIN LEAKTIGHT WITH A NEGATIVE PRESSURE OF 0.5 PSIG INSIDE CONTAINMENT VESSEL.
7. ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

Scale ⅛" = 1'-0"
Except as noted

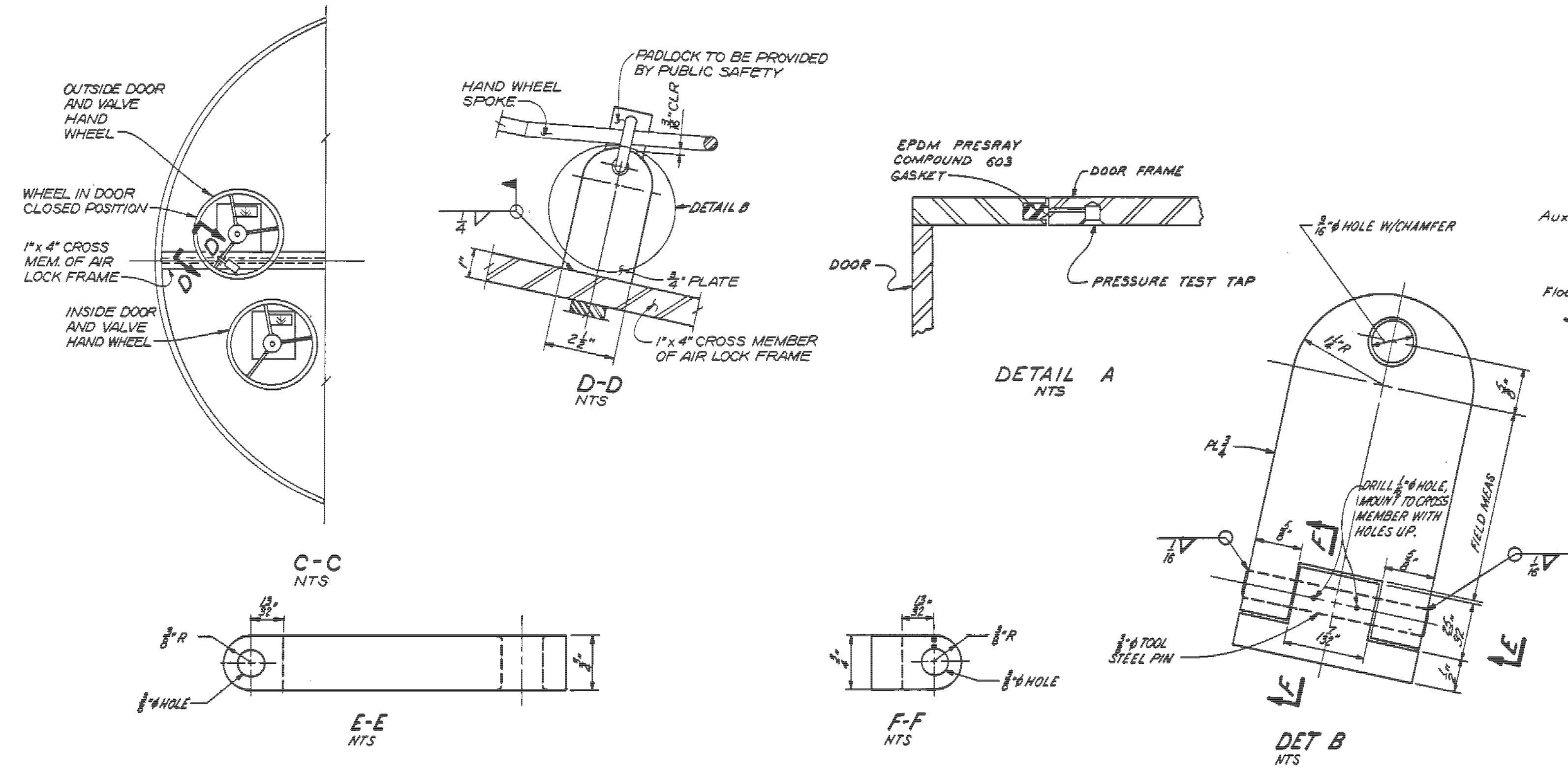
FOR HATCHED AREAS SEE UNIT 1 AC DWG 44N250

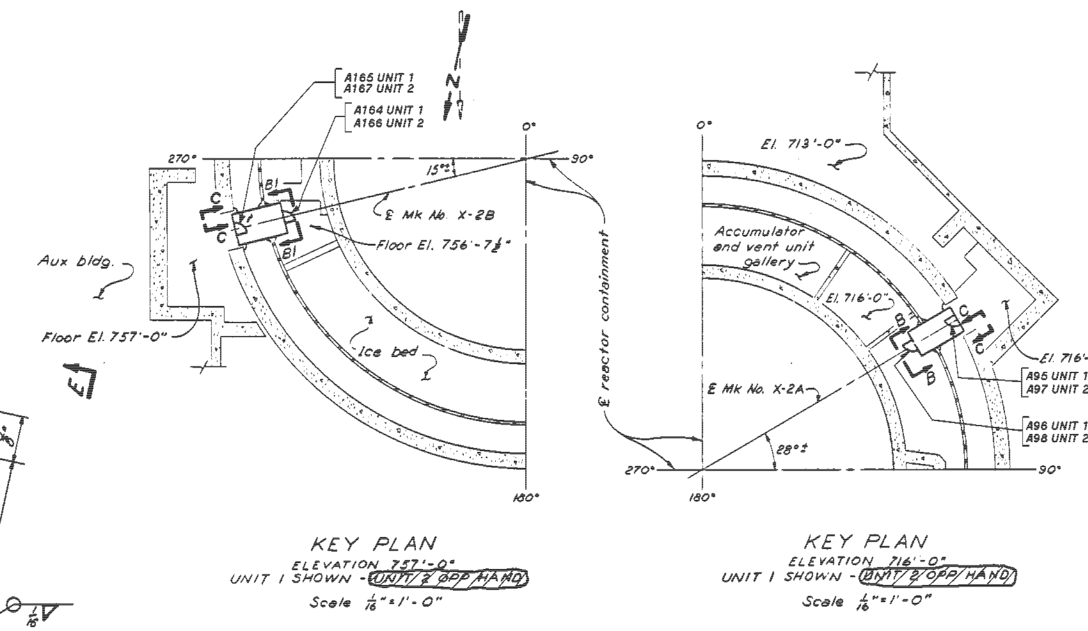
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
EQUIPMENT ACCESS HATCH
ARRANGEMENT & DETAILS

TVA DWG NO. 2-44N250 R0
FIGURE 3.8.2-4

CAD MAINTAINED DRAWING



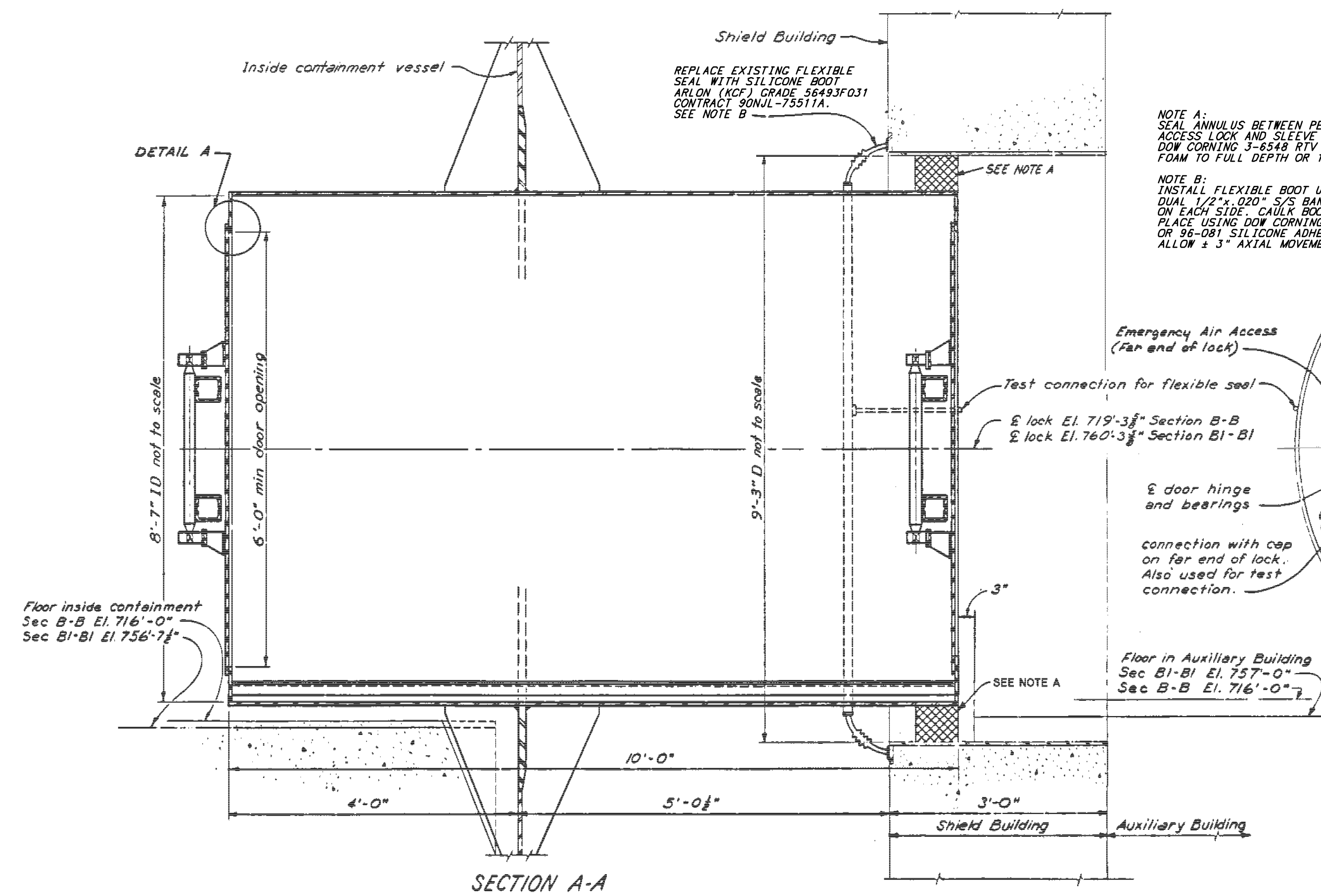


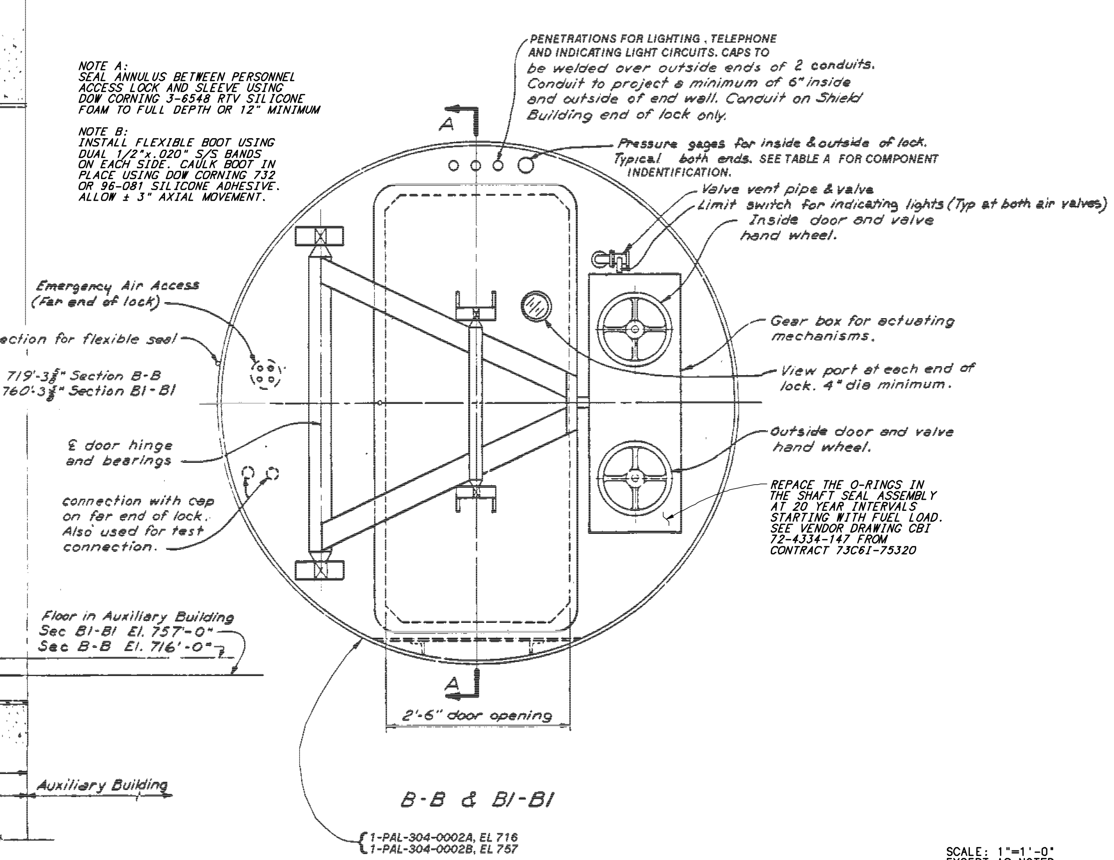
COMPONENT ID	DESCRIPTION	LOCATION
1-PI-304-0002A/1	PRESS IND OUTSIDE PAL AB END EL 716	EL 716, AZ 62
1-PI-304-0002A/2	PRESS IND INSIDE PAL AB END EL 716	EL 716, AZ 62
1-PI-304-0002A/3	PRESS IND INSIDE PAL RB END EL 716	EL 716, AZ 62
1-PI-304-0002A/4	PRESS IND OUTSIDE PAL RB END EL 716	EL 716, AZ 62
1-PI-304-0002B/1	PRESS IND OUTSIDE PAL AB END EL 757	EL 757, AZ 255
1-PI-304-0002B/2	PRESS IND INSIDE PAL AB END EL 757	EL 757, AZ 255
1-PI-304-0002B/3	PRESS IND INSIDE PAL RB END EL 757	EL 757, AZ 255
1-PI-304-0002B/4	PRESS IND OUTSIDE PAL RB END EL 757	EL 757, AZ 255

TABLE A

NOTES:
1. TWO LOCKS REQUIRED PER UNIT. FOUR LOCKS TOTAL REQUIRED.
2. LOCK SIZE - 8'-7" ID X ABOUT 10'-0" LONG.
3. DOOR OPENING SIZE - 2'-6" X 6'-0".
4. LOCKS DESIGNED FOR LOADS, PRESSURES AND TEMPERATURES, LOAD COMBINATIONS, AND STRESSES OUTLINED IN SPECIFICATION OR SHOWN ON THIS DRAWING.
5. LOCKS SEALED AT BOTH DOORS AND AT SHIELD BUILDING AS SHOWN.
6. DESIGNED FOR INTERNAL PRESSURE OF 13.5 PSIG AND TEST PRESSURE OF 16.9 PSIG.
7. EACH DOOR SHALL BE DESIGNED TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE CONTAINMENT VESSEL WITH OTHER DOOR OPEN.
8. EACH DOOR SHALL BE CAPABLE (WITH SECURING DEVICES ON INSIDE DOOR IF REQUIRED) TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE ACCESS LOCK WITHOUT LEAKING.
9. EACH DOOR SHALL REMAIN LEAKTIGHT WITH THE OTHER DOOR OPEN AND A NEGATIVE PRESSURE OF 0.5 PSIG INSIDE CONTAINMENT VESSEL.
10. LIVE LOAD OF 100 PSF OR 1000 LB MOVING CONCENTRATED LOAD AT ANY POINT.
11. LOCK OPERATION COMPLETELY MECHANICAL.
12. BOTH DOORS AND BOTH PRESSURE EQUALIZING VALVES ON EACH LOCK SHALL BE OPERABLE FROM OUTSIDE OF BOTH ENDS OF LOCK AND INSIDE LOCK.
13. OPERATION OF DOORS TO REQUIRE A MAXIMUM FORCE OF 40 LB.
14. DOORS AND VALVES INTERLOCKED SUCH THAT A DOOR AND/OR VALVE WILL NOT OPEN UNLESS THE DOOR AND VALVE AT THE OPPOSITE END OF THE LOCK ARE CLOSED. INTERLOCK TO HAVE MANUAL OVERRIDE BY SPECIAL TOOLS AND PROCEDURES.
15. DOOR HINGES TO ALLOW ADJUSTMENT IN ALL DIRECTIONS WITH EQUAL PRESSURE ON ALL SEALS.
16. INDICATORS PROVIDED INSIDE LOCK AND OUTSIDE OF BOTH ENDS OF LOCK TO SHOW THE POSITION OF BOTH DOORS AND BOTH VALVES.
17. LIMIT SWITCHES TO BE PROVIDED FOR REMOTE INDICATION ON THE UNIT CONTROL BOARD AS TO THE POSITION OF EACH DOOR AND EQUALIZING VALVE.
18. LOCKS DESIGNED TO ALLOW FOR 1-1/2" MINIMUM VERTICAL AND 1" MINIMUM HORIZONTAL MOVEMENT OF LOCKS WITH RESPECT TO SHIELD BUILDING DUE TO PRESSURE AND THERMAL EXPANSION.

B. ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.





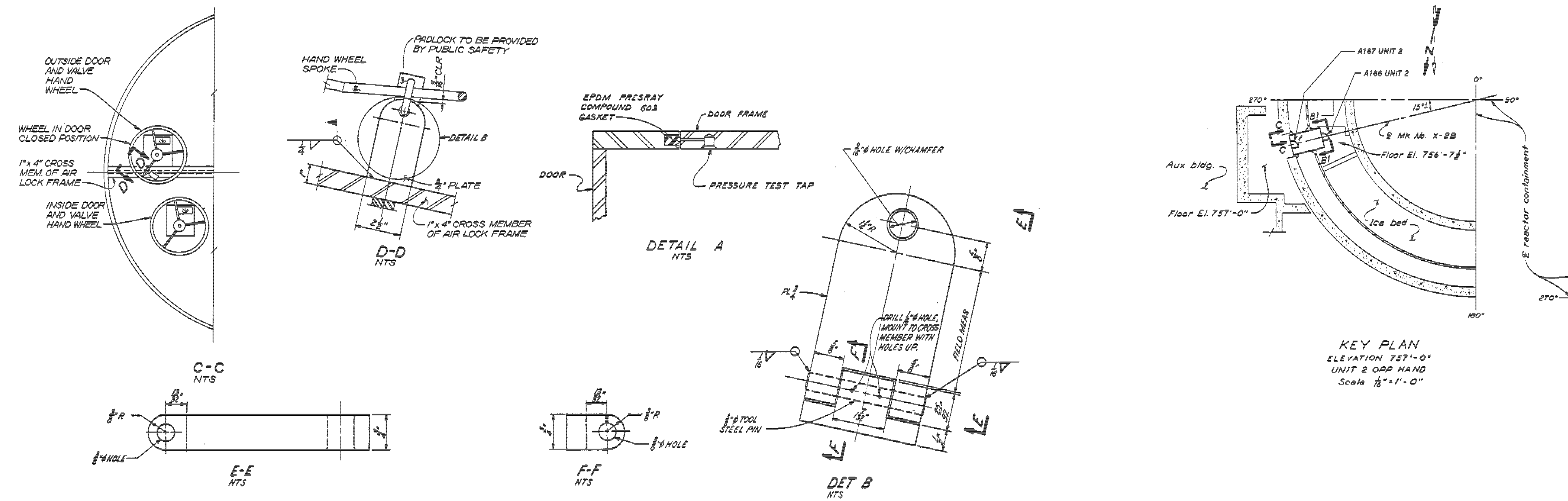
SCALE: 1"=1'-0"
EXCEPT AS NOTED

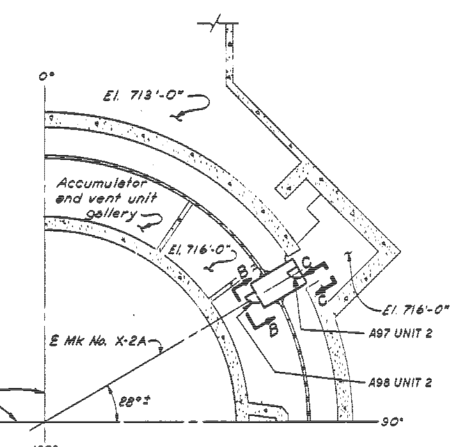
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
PERSONNEL ACCESS LOCK
ARRGT & DETAILS

TVA DWG NO. 44N240 RJ
FIGURE 3.8.2-5

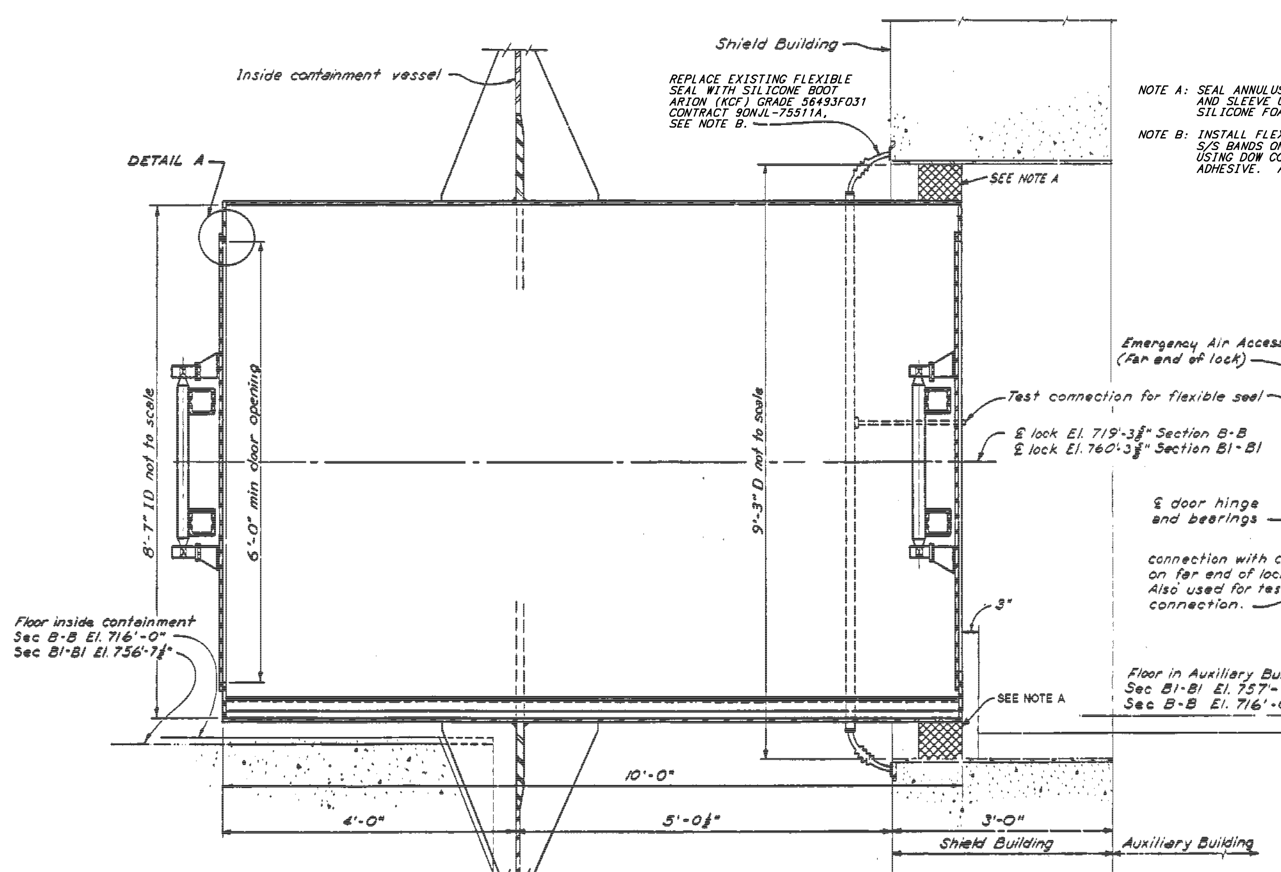
CAD MAINTAINED DRAWING





NOTES:

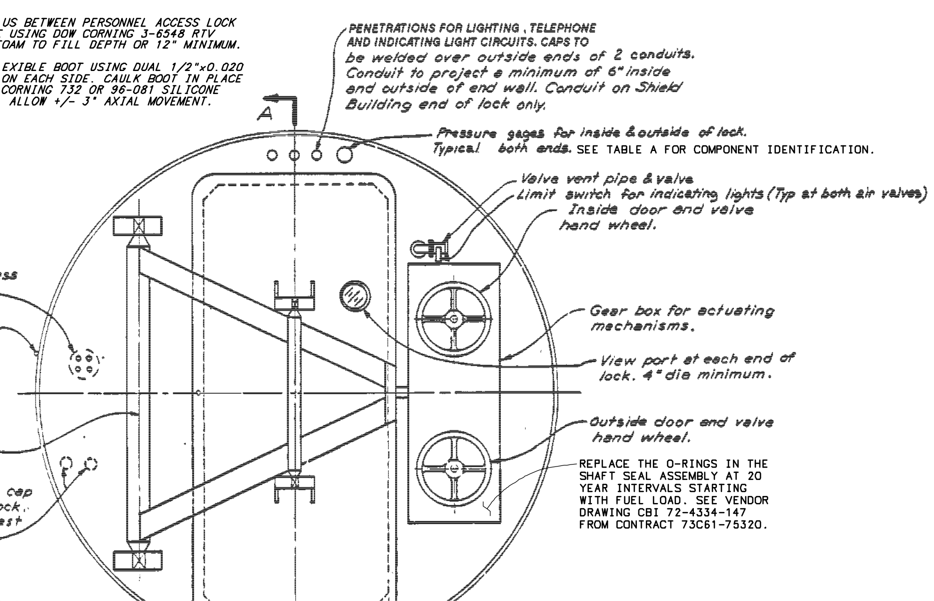
1. TWO LOCKS REQUIRED PER UNIT. FOUR LOCKS TOTAL REQUIRED.
2. LOCK SIZE - 8'-7" ID X ABOUT 10'-0" LONG.
3. DOOR OPENING SIZE - 2'-6" X 6'-0".
4. LOCKS DESIGNED FOR LOADS, PRESSURES AND TEMPERATURES, LOAD COMBINATIONS, AND STRESSES OUTLINED IN SPECIFICATION OR SHOWN ON THIS DRAWING.
5. LOCKS SEALED AT BOTH DOORS AND AT SHIELD BUILDING AS SHOWN.
6. DESIGNED FOR INTERNAL PRESSURE OF 13.5 PSIG AND TEST PRESSURE OF 16.9 PSIG.
7. EACH DOOR SHALL BE DESIGNED TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE CONTAINMENT VESSEL WITH OTHER DOOR OPEN.
8. EACH DOOR SHALL BE CAPABLE (WITH SECURING DEVICES ON INSIDE DOOR IF REQUIRED) TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE ACCESS LOCK WITHOUT LEAKING.
9. EACH DOOR SHALL REMAIN LEAKTIGHT WITH THE OTHER DOOR OPEN AND A NEGATIVE PRESSURE OF 0.5 PSIG INSIDE CONTAINMENT VESSEL.
10. LIVE LOAD OF 100 PSF OR 1000 LB MOVING CONCENTRATED LOAD AT ANY POINT.
11. LOCK OPERATION COMPLETELY MECHANICAL.
12. BOTH DOORS AND BOTH PRESSURE EQUALIZING VALVES ON EACH LOCK SHALL BE OPERABLE FROM OUTSIDE OF BOTH ENDS OF LOCK AND INSIDE LOCK.
13. OPERATION OF DOORS TO REQUIRE A MAXIMUM FORCE OF 40 LB.
14. DOORS AND VALVES INTERLOCKED SUCH THAT A DOOR AND/OR VALVE WILL NOT OPEN UNLESS THE DOOR AND VALVE AT THE OPPOSITE END OF THE LOCK ARE CLOSED. INTERLOCK TO HAVE MANUAL OVERRIDE BY SPECIAL TOOLS AND PROCEDURES.
15. DOOR HINGES TO ALLOW ADJUSTMENT IN ALL DIRECTIONS WITH EQUAL PRESSURE ON ALL SEALS.
16. INDICATORS PROVIDED INSIDE LOCK AND OUTSIDE OF BOTH ENDS OF LOCK TO SHOW THE POSITION OF BOTH DOORS AND BOTH VALVES.
17. LIMIT SWITCHES TO BE PROVIDED FOR REMOTE INDICATION ON THE UNIT CONTROL BOARD AS TO THE POSITION OF EACH DOOR AND EQUALIZING VALVE.
18. LOCKS DESIGNED TO ALLOW FOR 1-1/2" MINIMUM VERTICAL AND 1" MINIMUM HORIZONTAL MOVEMENT OF LOCKS WITH RESPECT TO SHIELD BUILDING DUE TO PRESSURE AND THERMAL EXPANSION.
19. ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.



SECTION A-A

NOTE A: SEAL ANNULUS BETWEEN PERSONNEL ACCESS LOCK AND SLEEVE USING DOW CORNING 3-6548 RTV SILICONE FOAM TO FILL DEPTH OR 12" MINIMUM.

NOTE B: INSTALL FLEXIBLE BOOT USING DUAL 1/2"x0.020 S/S BANDS ON EACH SIDE. CAULK BOOT IN PLACE USING DOW CORNING 732 OR 96-081 SILICONE ADHESIVE. ALLOW +/- 3" AXIAL MOVEMENT.



B-B & B1-B1

2-PAL-304-0002A - EL. 716'
2-PAL-304-0002B - EL. 757'

COMPONENT ID	DESCRIPTION	LOCATION
2-PI-304-0002A/1	PRESS IND OUTSIDE PAL AB END EL 716	EL 716', AZ 62°
2-PI-304-0002A/2	PRESS IND INSIDE PAL AB END EL 716	EL 716', AZ 62°
2-PI-304-0002A/3	PRESS IND INSIDE PAL RB END EL 716	EL 716', AZ 62°
2-PI-304-0002A/4	PRESS IND OUTSIDE PAL RB END EL 716	EL 716', AZ 62°
2-PI-304-0002B/1	PRESS IND OUTSIDE PAL AB END EL 757	EL 757', AZ 255°
2-PI-304-0002B/2	PRESS IND INSIDE PAL AB END EL 757	EL 757', AZ 255°
2-PI-304-0002B/3	PRESS IND INSIDE PAL RB END EL 757	EL 757', AZ 255°
2-PI-304-0002B/4	PRESS IND OUTSIDE PAL RB END EL 757	EL 757', AZ 255°

TABLE A

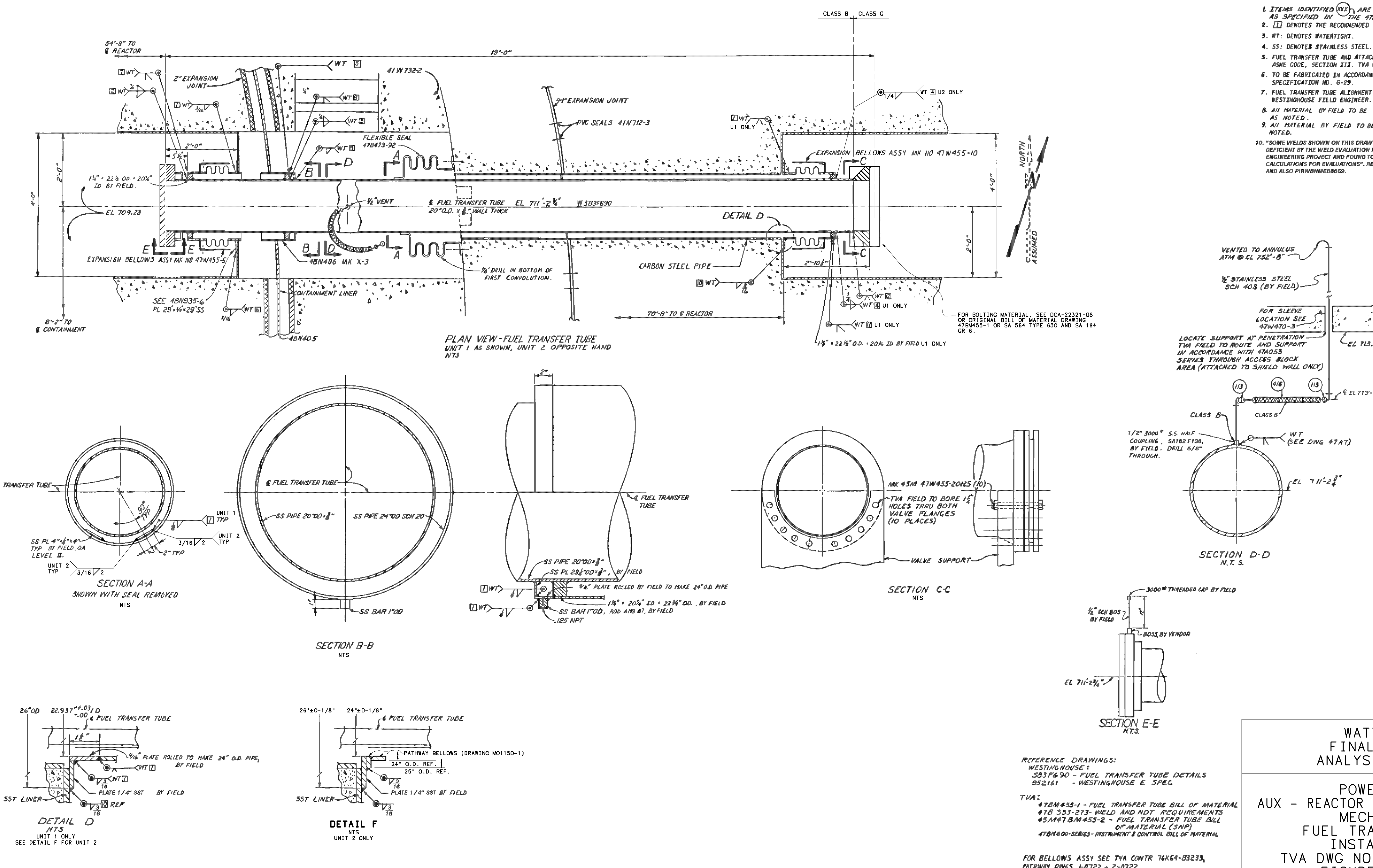
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
PERSONNEL ACCESS LOCK
ARRGT & DETAILS

TVA DWG NO. 2-44N240 R4
FIGURE 3.8.2-5(U2)

CAD MAINTAINED DRAWING



NOTES:

1. ITEMS IDENTIFIED (XXX) ARE TVA MARK NO. 47W600-XXX AS SPECIFIED IN THE 47BM600 SERIES.
2. [1] DENOTES THE RECOMMENDED SEQUENCE FOR FIELD WELDS.
3. WT: DENOTES WATERTIGHT.
4. SS: DENOTES STAINLESS STEEL.
5. FUEL TRANSFER TUBE AND ATTACHMENT WELDS ARE CLASS 2, ASME CODE, SECTION III. TVA CLASS B.
6. TO BE FABRICATED IN ACCORDANCE WITH TVA CONSTRUCTION SPECIFICATION NO. G-29.
7. FUEL TRANSFER TUBE ALIGNMENT TO BE AS SPECIFIED BY WESTINGHOUSE FIELD ENGINEER.
8. All MATERIAL BY FIELD TO BE A240 TP 304 S.S. EXCEPT AS NOTED.
9. All MATERIAL BY FIELD TO BE QA LEVEL I, EXCEPT AS NOTED.
10. "SOME WELDS SHOWN ON THIS DRAWING (SERIES) WERE FOUND TO BE DEFICIENT BY THE WELD EVALUATION PROJECT, EVALUATED BY WATTS BAR ENGINEERING PROJECT AND FOUND TO BE SUITABLE FOR SERVICE. SEE DESIGN CALCULATIONS FOR EVALUATIONS". REFER TO RIMS NO. B26890901319 AND ALSO PIRWBNMEB8669.

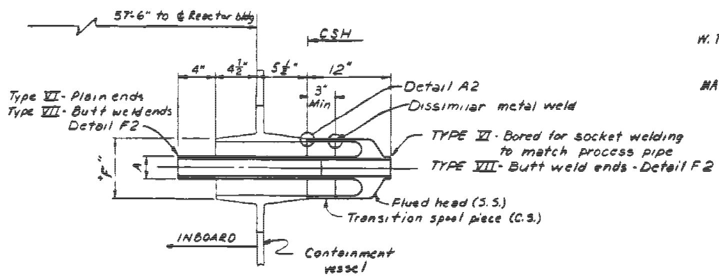
REFERENCE DRAWINGS:
WESTINGHOUSE:
5B3F690 - FUEL TRANSFER TUBE DETAILS
952161 - WESTINGHOUSE E SPEC
TVA:
47BM455-1 - FUEL TRANSFER TUBE BILL OF MATERIAL
47B 333-273- WELD AND NDT REQUIREMENTS
45M47BM455-2 - FUEL TRANSFER TUBE BILL OF MATERIAL (SNP)
47BM600-SERIES - INSTRUMENT & CONTROL BILL OF MATERIAL

FOR BELLOWS ASSY SEE TVA CONTR 76K64-83233, PATHWAY DWGS 1-0722 + 2-0722.

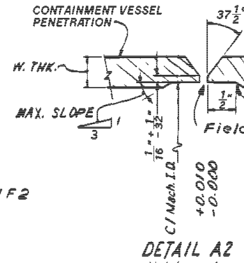
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
AUX - REACTOR BLDGS-UNITS 1 & 2
MECHANICAL
FUEL TRANSFER TUBE
INSTALLATION
TVA DWG NO. 47W455-1 RL
FIGURE 3.8.2-6

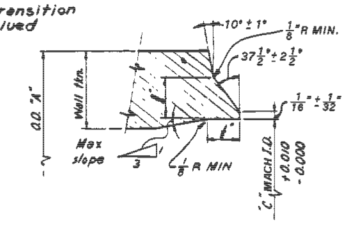
CAD MAINTAINED DRAWING



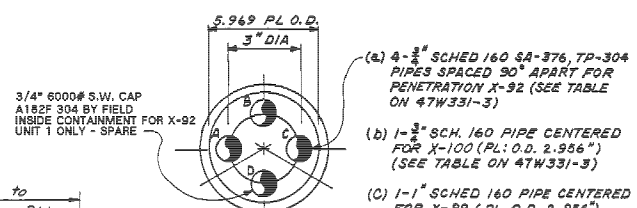
TYPE VI & VII
Type VI for socket weld SS process lines
Type VII for butt weld SS process lines
Not to scale

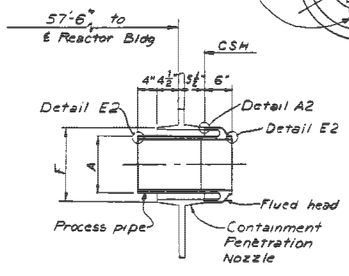


DETAIL A2
Not to scale

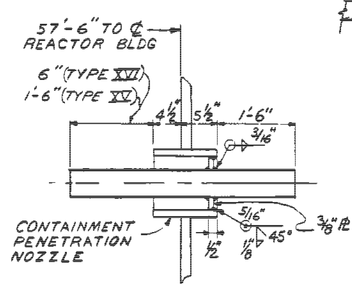


DETAIL B2
For wall thickness over 3/4"



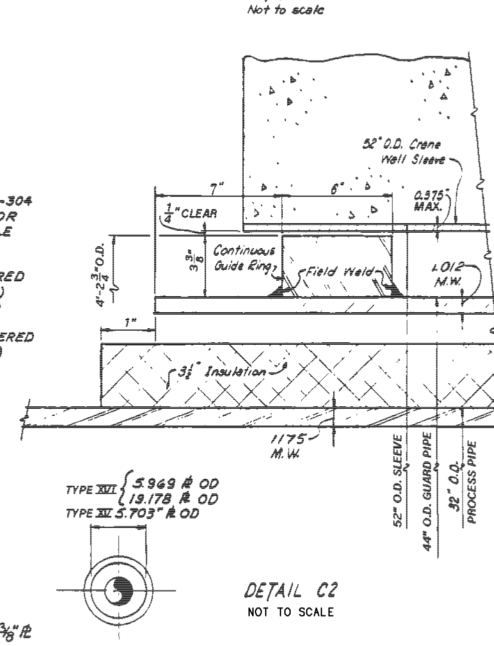


TYPE VIII
For butt weld C.S. process lines
Not to scale

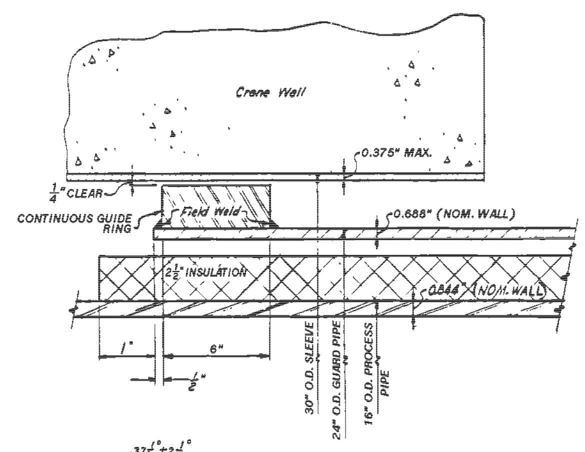


TYPE XV
TYPE XVI

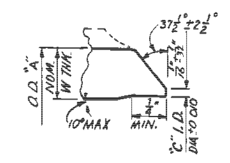
NOT TO SCALE



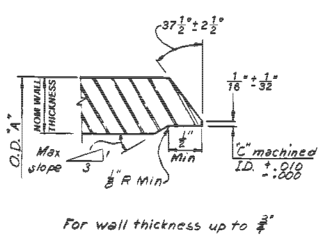
DETAIL C2
NOT TO SCALE



DETAIL D2
NOT TO SCALE

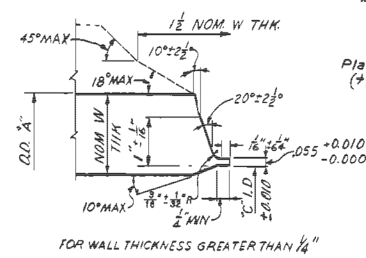


FOR WALL THICKNESS 1/8" TO 1/4" INCLUSIVE



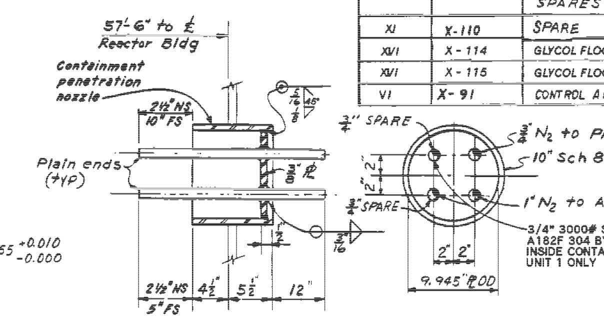
For wall thickness up to 3/4"

DETAIL E2
Not to scale

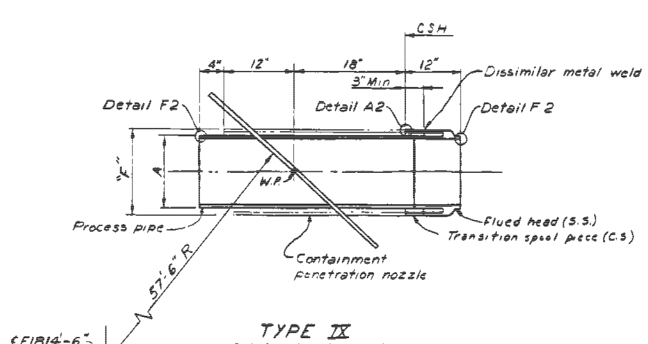


FOR WALL THICKNESS GREATER THAN 1/4"

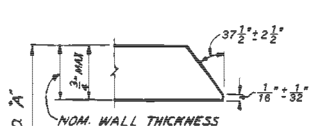
DETAIL F2
NOT TO SCALE



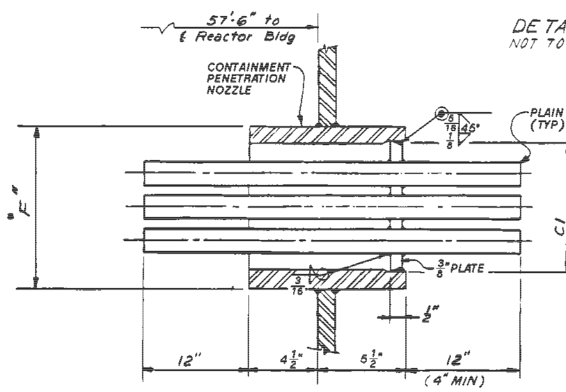
TYPE XIII
Multi line penetration X-39
Not to scale



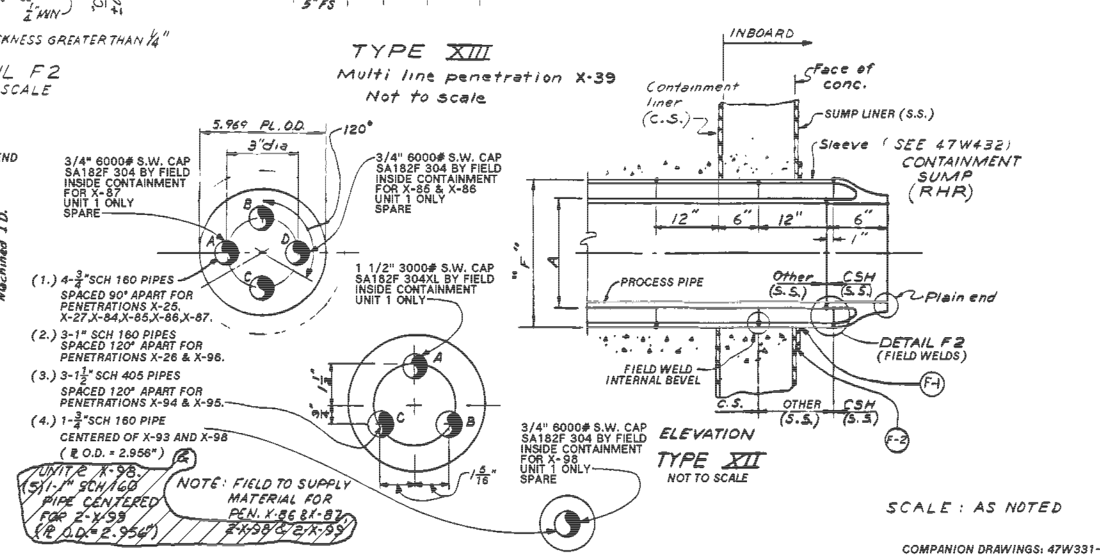
TYPE IX
FOR S.S PROCESS LINES
Not to scale



DETAIL G2
NOT TO SCALE



TYPE X
Instrument Penetrations
Not to scale



ELEVATION
TYPE XII
NOT TO SCALE

PENETRATION WITHOUT BELLOWS																
TYPE	PENETRATION NO.	DESCRIPTION	PROCESS PIPE						CONT. VESSEL PENETRATION				FLUED HEAD			LOCATED ON T.V.A. DWG NO.
			O.D.A.	"C" I.D.	W.THK OR SCH	ASME MAT'L	DESIGN TEMP °F	DESIGN PRESS PSI	O.D. "F"	CI MACH I.D.	W.THK OR SCH	ASME MAT'L	SIZE	ASME MAT'L		
VII	X - 16	NORMAL CHARGING	3.50"	2.692"	160	SA-376 TP-304	150°	2800	6.625"	6.094"	80	SA-333 GR 6	6 X 3	SA-182 GR F304		47W406
X	X - 25	INST LINES	4-1.05"	—	160	↓	650°	2485	6.625"	6.094"	80	SA-333 GR 6	6 X 3/8 ℞	SA-516 GR 60		47W600
XVI	X - 64	CHILLED WATER	2.375"	—	40	SA-106 GR B	45°	25	6.625"	6.094"	80	SA-333 GR-6	6 x 3/8 ℞	SA-516 GR-60		47W915
VI	X - 41	FLOOR PUMP SUMP DISCH	2.375"	—	40S	SA-312 TYP302	200°	150	6.625"	6.094"	80	SA-333 GR 6	6 X 2	SA-182 GRF304		47W482
VII	X - 42	PRT MAKE - UP	3.50"	3.081	40S	SA-312 TYP304	300°	300					6 X 3			47W492
VI	X - 43A	SEAL WATER INJECTION	2.375"	—	160	SA-376 TP304	150°	2800					6 X 2			47W406
	X - 43B			—												
	X - 43C			—												
	X - 43D	↓		—	↓	↓	↓	↓								↓
↓	X - 77	DEMINERALIZED WATER	↓	—	40S	SA-312 TYP304	150°	100	↓	↓			↓			47W492
VII	X - 82	REFUEL W PURIF PUMP SUCT	6.625"	6.094	40S	SA-312 TYP304	150°	150	10.75"	10.070"			10 X 6			47W454
VII	X - 83	REFUEL W PURIF PUMP DISCH	4.50"	4.044	40S	SA-312 TYP304	150°	150	8.625"	8.020"			8 X 4	↓		47W454
X	X - 84	INST LINES	4-1.05"	—	160	SA-376 TYP304	150°	25	6.625"	6.094"			6 X 3/8 ℞	SA-516 GR 60		47W600
↓	X - 35	INST LINES	4-1.05"	—	160	SA-376 TYP304	650°	2485	6.625"	6.094"	↓	↓	↓	SA-516 GR 60		47W600
XVI	X - 65	CHILLED WATER	2.375"	—	40S	SA-106 GR B	45°	25	6.625"	6.094"	80	SA-333 GR-6	6 x 3/8 ℞	SA-516 GR-60		47W915
XVI	X - 66	↓	↓	—	↓	↓	↓	↓	↓	↓	↓	↓	↓	↓		↓
VII	X - 44	RC PUMP SEAL WATER OUT	4.50"	4.044"	40S	SA-312 TYP304	150°	200	8.625"	8.020"	80	SA-333 GR-6	8 X 4	SA-182 GRF304		47W406
VI	X - 90	CONTROL AIR	2.375"	—	40S	↓	150°	120	4.50"	4.044"	80	SA-333 GR-6	4 X 2			47W600
X	X - 93	INST LINES	1.05"	—	160	SA-376 TP 304	300°	2485	3.50"	3.081"	XX-STG		3 X 3/8 ℞	SA-516 GR 60		
	X - 94		3-1.9"	—	40S	SA-312 TP 304	150°	30	6.625"	6.094"	80		6 X 3/8 ℞			
	X - 95		↓	—	↓	↓	↓	↓								
	X-96 & X-26		3-1.315"	—	160	SA-376 TP304	150°	30								
↓	X - 27	↓	4-1.05"	—	↓	↓	650°	2485	↓	↓		↓	↓	↓		↓
VI	X - 34	CONTROL AIR	2.375"	—	40S	SA-312 TP304	150	120	8.625"	8.020"		SA-333 GR-6	8 X 2	SA-182 GRF304		47W600
VIII	X - 40A	AUXILIARY FEEDWATER	4.50"	3.869"	80	SA-333 GR6	600°	1185	6.625"	6.094"		SA-333 GR-6	6 X 4	SA-350 GR LF2		47W427
VIII	X - 40B	AUXILIARY FEEDWATER	4.50"	3.869"	80	SA-333 GR6	600°	↓	6.625"	6.094"	↓		6 X 4	↓		47W427
IX	X - 48A	CONTAINMENT SPRAY	10.75"	10.070"	40S	SA-312 TYP304	115°	150	14.0"	13.303"	100		14 X 10	SA-182 GRF304		47W437
	X - 48B	CONTAINMENT SPRAY	10.75"	10.070"	40S	↓	↓	↓	14.0"	13.303"	100		14 X 10			
	X - 49A	RHR SPRAY	8.625"	8.020"	40S	SA-312 TP304	250°	150	12.75"	12.053"	80		12 X 8			
↓	X - 49B	RHR SPRAY	8.625"	8.020"	40S	↓	↓	↓	12.75"	12.053"	80		12 X 8			↓
VI	X - 97	INST LINES	1.05"	—	160	SA-376 TP304	653°	2235	3.5"	3.081"	XX-STG		3 X 3/4			47W600
X	X - 98	(SPARE)	↓	—									3 X 3/4			↓
VI	X - 99	INST LINES	1.315"	—	↓	↓	↓	↓	↓	↓	↓	↓	3 X 1	↓		47W625-11
XVI	X - 67	CHILLED WATER	2.375"	—	40	SA-106 GR B	45°	25	6.625"	6.094"	80	SA-333 GR 6	6 x 3/8 ℞	SA-516 GR 60		47W915
XII	X - 19A	REACTOR RHR CONT. SUMP	18"	17.303"	.375"	SA358-1 TP304	190°	50	24" *	23.303"	0.375"	SA-516 GR 70	24 x 18	SA-182 GRF304		47W432
XII	X - 19B	↓	18"	17.303"	.375"	SA358-1 TP304	190°	50	24" *	23.303"	0.375"	↓	24 x 18	↓		47W432
XIII	X - 39	N2 TO ACC.		—	160	SA-106 GR B	150°	900	10.750"	10.070"	80	SA-333 GR-6	10 x 3/8 TK	SA-516 GR 60		47W500
		N2 TO PRI	1.05"	—	40S	SA-312 TP304	150°	150								
		SPARES (2)	1.05"	—	40S	SA-312 TP304	—	—								
XI	X - 110	SPARE	2.375"	—	160	SA-376 TP304	150°	2735	6.625"	5.828"		SA-333 GR-6	6 x 3/8 TK	SA-516 GR 60		47W435-22
XVI	X - 114	GLYCOL FLOOR COOLING	2.375"	—	40	SA-333 GP-6	-10°	150	20"	19.303"	↓		20 x 3/8 TK	↓		47W462
XVI	X - 115	GLYCOL FLOOR COOLING	2.375"	—	40	SA-333 GR-6	-10°	150	20"	19.303"	80	↓	20 x 3/8 TK	SA-516 GR 60		47W462
VI	X - 91	CONTROL AIR	2.375"	—	40S	SA-312 TYP304	150°	120	4.50"	4.044"	80	SA-333 GR-6	4 X 2	SA-182 GRF304		47W600

* SLEEVE

NOTES:
1. For notes and reference dwg's see 47W331-1.

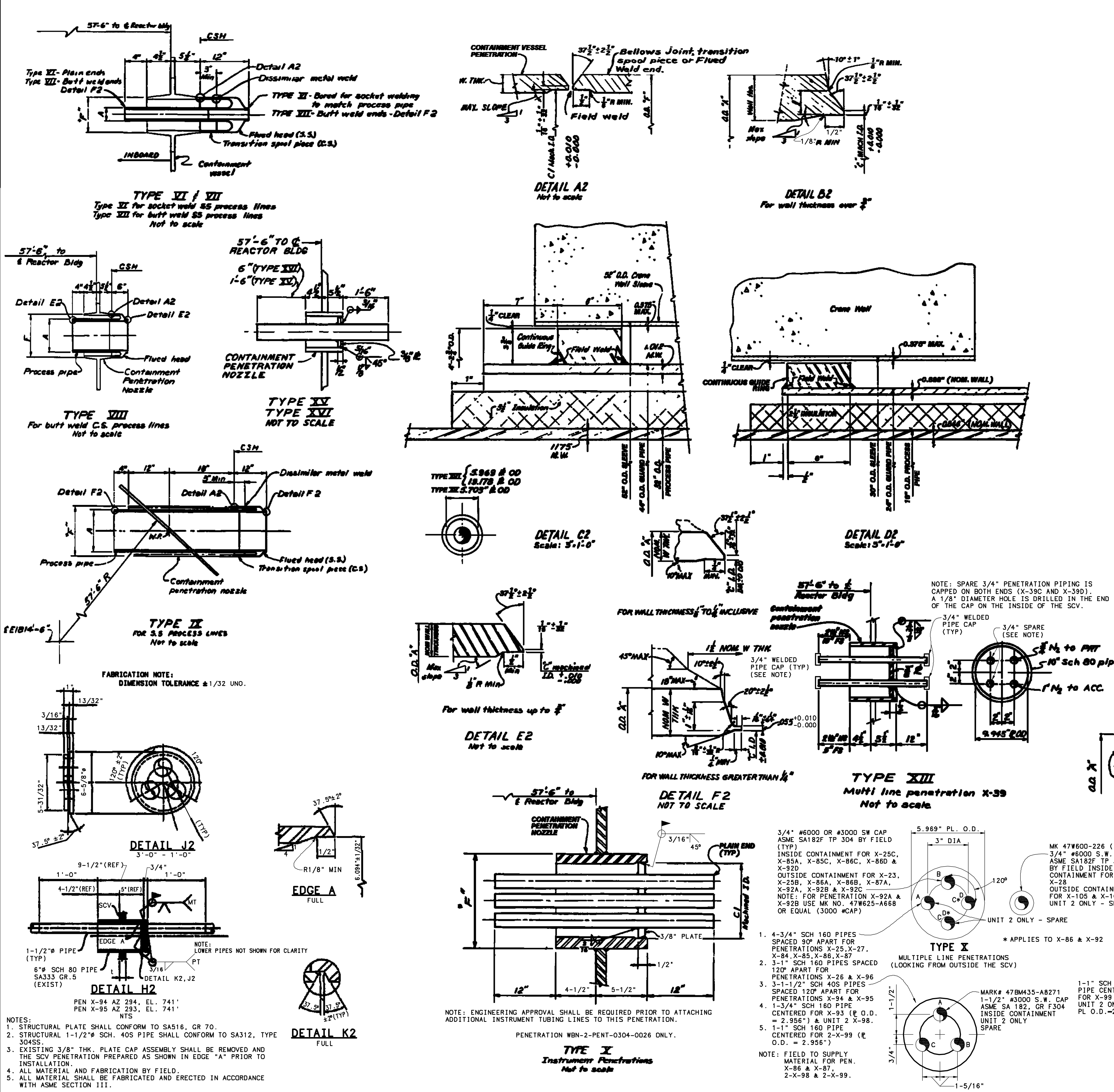
SCALE: AS NOTED

COMPANION DRAWINGS: 47W331-1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING
UNIT 1
MECHANICAL
CONTAINMENT PENETRATION
TVA DWG NO. 47W331-2 RF
FIGURE 3.8.2-7

CAD MAINTAINED DRAWING

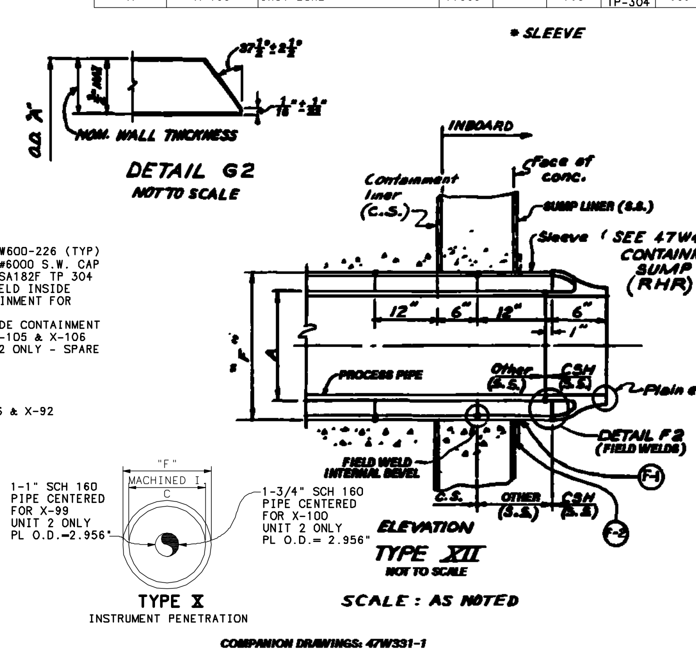


PENETRATION WITHOUT BELLOWS

TYPE	PENETRATION NO.	DESCRIPTION	PROCESS PIPE O.D.A.	"C" I.D.	W.THK OR SCH	ASME MAT'L	DESIGN TEMP °F	DESIGN PRESS PSI	CONT. VESSEL PENETRATION O.D. "F"	CI MACH I.D.	W.THK OR SCH	ASME MAT'L	FLUED HEAD SIZE	ASME MAT'L		LOCATED ON T.V.A. DWG NO.
VII	X - 16	NORMAL CHARGING	3.50"	2.692"	160	SA-376 TP-304	150°	2800	6.625"	6.094"	80	SA-333 GR 6	6 X 3	SA-182 GR F304		47W406
X	X - 25	INST LINES	4-1.05"	—	160	↓	650°	2485	6.625"	6.094"	80	SA-333 GR 6	6 X 3/8 ℄	SA-516 GR 60		47W600
XVI	X - 64	CHILLED WATER	2.375"	—	40	SA-106 GR B	45°	25	6.625"	6.094"	80	SA-333 GR-6	6 x 3/8 ℄	SA-516 GR-60		47W915
VI	X - 41	FLOOR PUMP SUMP DISCH	2.375"	—	40S	SA-312 TYP302	200°	150	6.625"	6.094"	80	SA-333 GR 6	6 X 2	SA-182 GRF304		47W482
VII	X - 42	PRT MAKE - UP	3.50"	3.081	40S	SA-312 TYP304	300°	300					6 X 3			47W492
VI	X - 43A	SEAL WATER INJECTION	2.375"	—	160	SA-376 TP304	150°	2800					6 X 2			47W406
	X - 43B			—												
	X - 43C			—												
	X - 43D	↓		—	↓	↓	↓	↓								↓
↓	X - 77	DEMINERALIZED WATER	↓	—	40S	SA-312 TYP304	150°	100	↓	↓			↓			47W492
VII	X - 82	REFUEL W PURIF PUMP SUCT	6.625"	6.094	40S	SA-312 TYP304	150°	150	10.75"	10.070"			10 X 6			47W454
VII	X - 83	REFUEL W PURIF PUMP DISCH	4.50"	4.044	40S	SA-312 TYP304	150°	150	8.625"	8.020"			8 X 4	↓		47W454
X	X - 84	I&C RVLIS	4-1.05"	—	160	SA-376 TYP304	150°	25	6.625"	6.094"			6 X 3/8 ℄	SA-516 GR 60		47W600 47W600-314
↓	X - 35	INST LINES	4-1.05"	—	160	SA-376 TYP304	650°	2485	6.625"	6.094"	↓	↓	↓	SA-516 GR 60		47W600
XVI	X - 65	CHILLED WATER	2.375"	—	40	SA-106 GR B	45°	25	6.625"	6.094"	80	SA-333 GR-6	6 x 3/8 ℄	SA-516 GR-60		47W915
XVI	X - 66	↓	↓	—	↓	↓	↓	↓	↓	↓	↓	↓	↓	↓		↓
VII	X - 44	RC PUMP SEAL WATER OUT	4.50"	4.044*	40S	SA-312 TYP304	150°	200	8.625"	8.020"	80	SA-333 GR-6	8 X 4	SA-182 GRF304		47W406
VI	X - 90	CONTROL AIR	2.375"	—	40S	↓	150°	120	4.50"	4.044"	80	SA-333 GR-6	4 X 2	↓		47W600
X	X - 93	INST LINES	1.05"	—	160	SA-376 TP 304	300°	700	3.50"	3.081"	XX-STG		3 X 3/8 ℄	SA-516 GR 60		47W600
	X - 94	RADIATION MONITORING	3-1.9"	—	40S	SA-312 TP 304	150°	30	6.625"	6.094"	80		6 X 3/8 ℄			47W600-105
	X - 95	RADIATION MONITORING	↓	—	↓	↓	↓	↓								47W600-105
	X-96 & X-26	INST LINES	3-1.315"	—	160	SA-376 TP304	150°	30								47W600
↓	X - 27	INST LINES	4-1.05"	—	↓	↓	650°	2485	↓	↓		↓	↓	↓		47W600
VI	X - 34	CONTROL AIR	2.375"	—	40S	SA-312 TP304	150	120	8.625"	8.020"		SA-333 GR-6	8 X 2	SA-182 GRF304		47W600
VIII	X - 40A	AUXILIARY FEEDWATER	4.50"	3.869"	80	SA-333 GR6	600°	1185	6.625"	6.094"		SA-333 GR-6	6 X 4	SA-350 GR LF2		47W427
VIII	X - 40B	AUXILIARY FEEDWATER	4.50"	3.869"	80	SA-333 GR6	600°	↓	6.625"	6.094"	↓		6 X 4	↓		47W427
IX	X - 48A	CONTAINMENT SPRAY	10.75"	10.070"	40S	SA-312 TYP304	118°	150	14.0"	13.303"	100		14 X 10	SA-182 GRF304		47W437
	X - 48B	CONTAINMENT SPRAY	10.75"	10.070"	40S	↓	↓	↓	14.0"	13.303"	100		14 X 10			
	X - 49A	RHR SPRAY	8.625"	8.020"	40S	SA-312 TP304	250°	150	12.75"	12.063"	80		12 X 8			
↓	X - 49B	RHR SPRAY	8.625"	8.020"	40S	↓	↓	↓	12.75"	12.063"	80		12 X 8			↓
VI	X - 97	INST LINES	1.05"	—	160	SA-376 TP304	653°	2235	3.5"	3.081"	XX-STG		3 X 3/4			2-47W600-89
X	X - 98	INST LINES	↓	—									3 X 3/4	↓		2-47W600-89
X	X - 99	H_2 ANAL SAMPLE	1.315"	—	↓	↓	↓	↓	↓	↓	↓	↓	3 X 1	SA-128 GRF304		47W625-11
XVI	X - 67	CHILLED WATER	2.375"	—	40	SA-106 GR B	45°	25	6.625"	6.094"	80	SA-333 GR 6	6 x 3/8 ℄	SA-516 GR 60		47W915
XII	X - 19A	REACTOR RHR CONT. SUMP	18"	17.303"	.375"	SA358-1 TP304	190°	50	24"*	23.303"	0.375"	SA-516 GR 70	24 x 18	SA-182 GRF304		47W432
XII	X - 19B	↓	18"	17.303"	.375"	SA358-1 TP304	190°	50	24"*	23.303"	0.375"	↓	24 x 18	↓		47W432
XIII	X - 39	N_2 TO ACC.	1.315"	—	160	SA-106 GR B	150°	900	10.750"	10.070"	80	SA-333 GR-6	10 x 3/8 TK	SA-516 GR 60		47W600
		N_2 TO PRI	1.05"	—	40S	SA-312 TP304	150°	150								
		SPARES (2)	1.05"	—	40S	SA-312 TP304	—	—								
XI	X-110	SPARE	2.375"	—	160	SA-376 TP304	150°	2735	6.625"	5.828"		SA-333 GR-6	6 x 3/8 TK	SA-516 GR 60		47W435-22
XVI	X - 114	GLYCOL FLOOR COOLING	2.375"	—	40	SA-333 GR-6	-10°	150	20"	19.303"			20 x 3/8 TK	↓		47W462
XVI	X - 115	GLYCOL FLOOR COOLING	2.375"	—	40	SA-333 GR-6	-10°	150	20"	19.303"	80	↓	20 x 3/8 TK	SA-516 GR 60		47W462
VI	X - 91	CONTROL AIR	2.375"	—	40S	SA-312 TYP304	150°	120	4.50"	4.044"	80	SA-333 GR-6	4 X 2	SA-182 GRF304		47W600
X	X-92A	SPARE	1.050"	-	160	SA 376 TP-304	-	-	3.5"	N/A	160	SA-333 GR6	3"X 3/4"	SA-128 GRF304		-
X	X-92B	SPARE	1.050"	-	160	SA 376 TP-304	-	-	3.5"	N/A	160	SA-333 GR6	3"X 3/4"	SA-128 GRF304		-
X	X-92C	SPARE	1.050"	-	160	SA 376 TP-304	-	-	3.5"	N/A	160	SA-333 GR6	3"X 3/4"	SA-128 GRF304		-
X	X-92D	SPARE	1.050"	-	160	SA 376 TP-304	-	-	3.5"	N/A	160	SA-333 GR6	3"X 3/4"	SA-128 GRF304		-
X	X-100	INST LINE	1.050"	-	160	SA 376 TP-304	653°	2235	3.5"	3.081	XX-STG	SA-333 GR6	3"X 3/4"	SA-128 GRF304		47W625-11

* SLEEVE

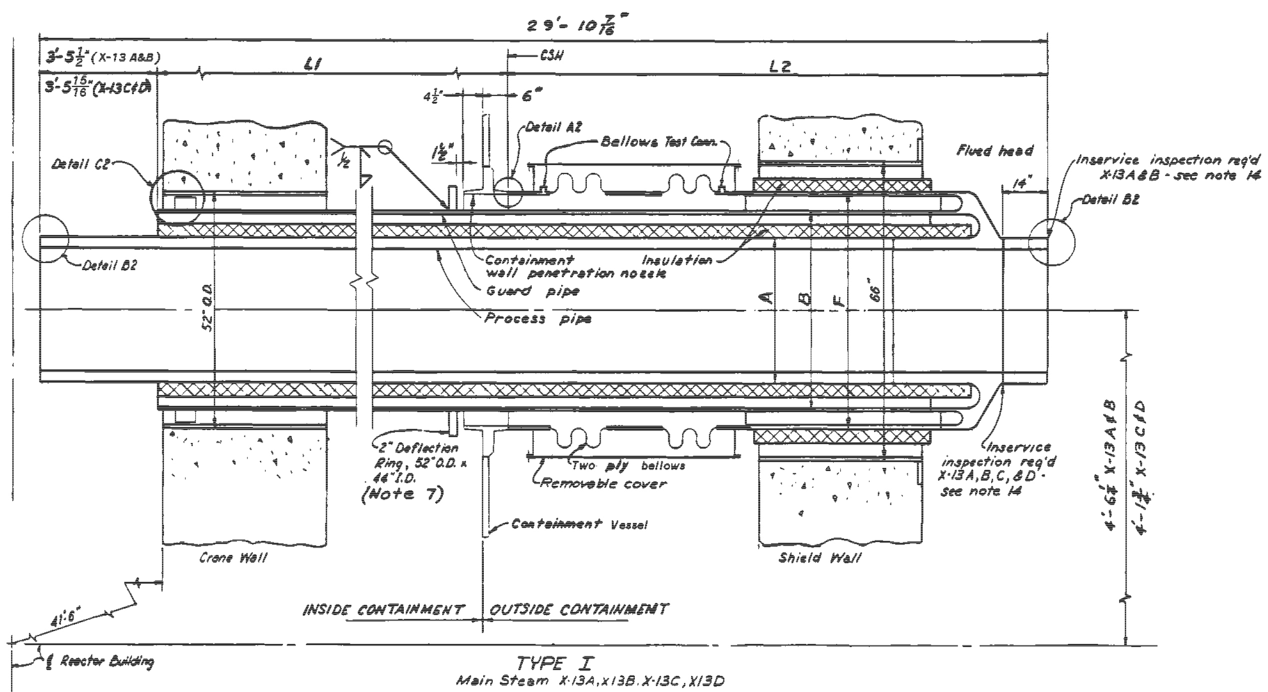
NOTES:
1. For notes and reference dwg's see 47W331-1.



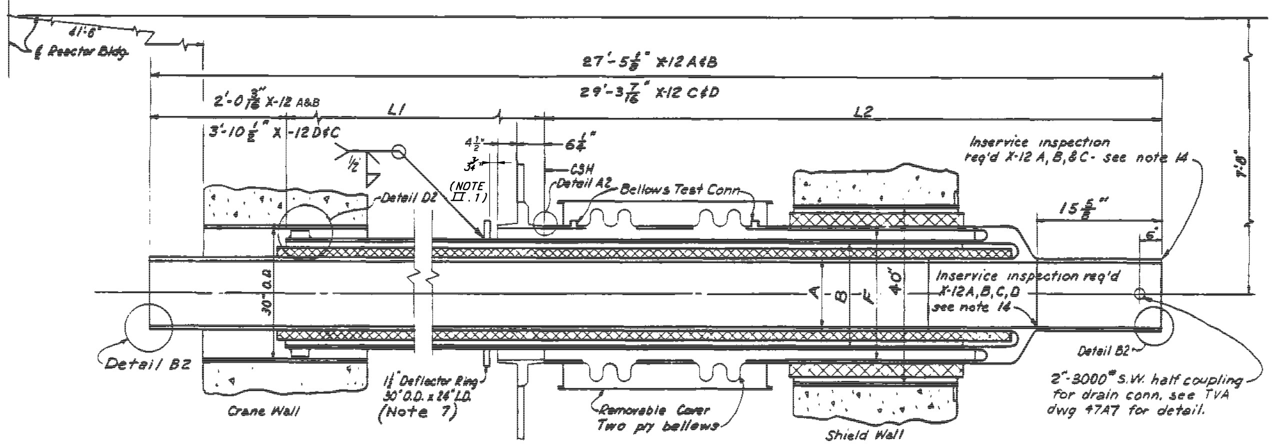
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING
UNIT 2
MECHANICAL
CONTAINMENT PENETRATION
TVA DWG NO. 2-47W331-2 R1
FIGURE 3.8.2-7(U2)

CAD MAINTAINED DRAWING



TYPE I
Main Steam X-13A, X13B, X-13C, X13D



NOTE II.1

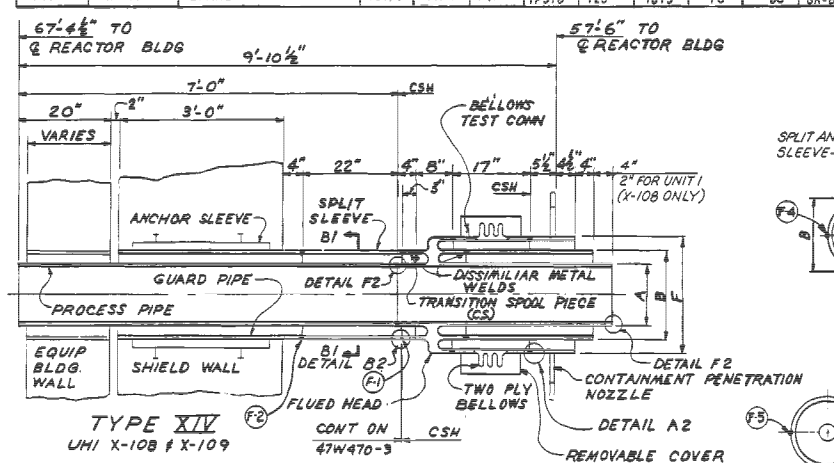
FOR PENETRATION X12C, THE OUTSIDE FACE (i.e. TOWARD THE CONTAINMENT VESSEL) OF THE DEFLECTOR RING MAY BE GROUND TO PROVIDE A MINIMUM CLEARANCE WITH THE PENETRATION SLEEVE OF 3 9/16". THE CLEARANCE IS TO BE PROVIDED FOR A MINIMUM OF 1/2" INSIDE THE I.D. OF THE PENETRATION SLEEVE.

TYPE II
Feedwater X-12A, X12B, X12C, X12D

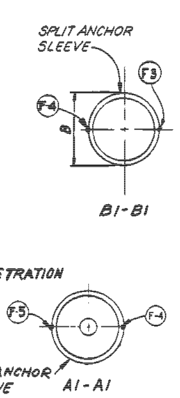
PEN. No.	UNIT 1 L1	UNIT 1 L2	UNIT 1 L3	UNIT 2 L1	UNIT 2 L2	UNIT 2 L3
X13A	16'-5 3/4"	9'-11 13/16"	NA	16'-6 3/16"	9'-10 3/8"	NA
X13B	16'-4 15/16"	10'-0"	NA	16'-4 11/16"	10'-0 1/4"	NA
X13C	16'-5 1/2"	9'-11"	NA	16'-5 1/16"	9'-11 7/16	NA
X13D	16'-6 1/4"	9'-10 7/8"	NA	16'-5 13/16"	9'-10 9/16"	NA
X12A	15'-1 3/4"	10'-3 3/16"	NA	15'-1 1/2"	10'-3 7/16"	NA
X12B	15'-0 7/8"	10'-4 1/16"	NA	15'-0 9/16"	10'-4 1/2"	NA
X12C	15'-1 9/16"	10'-3 3/8"	NA	15'-0 7/8"	10'-4 9/16"	NA
X12D	15'-1 7/8"	10'-3 1/16"	NA	15'-1"	10'-3 15/16"	NA
X17	57'-5 9/16"	5'-0 11/16"	3 9/16"	57'-6 9/16"	4'-11 7/8"	4 3/8"
X107	57'-5 3/16"	5'-1 5/16"	3 3/16"	57'-7 3/16"	4'-10 15/16"	5 9/16"

PENETRATIONS WITH BELLOWS

TYPE	PENETRATION NO.	DESCRIPTION	PROCESS PIPE O.D.A	PROCESS PIPE I.D."C"	PROCESS PIPE B THK OR SCH	PROCESS PIPE ASME MAT'L	PROCESS PIPE DESIGN TEMP °F	PROCESS PIPE DESIGN PRESS PSI	GUARD PIPE O.D.B	GUARD PIPE W THK OR SCH	GUARD PIPE ASME MAT'L	CONT. VESSEL PENETRATION O.D.F	CONT. VESSEL PENETRATION CI "MACH" I.D.	CONT. VESSEL PENETRATION W. THK OR SCH	CONT. VESSEL PENETRATION ASME MAT'L	FLUED HEAD SIZE	FLUED HEAD ASME MAT'L		BELLOWS MOVEMENTS AXIAL EXTEN.	BELLOWS MOVEMENTS AXIAL COMP.	BELLOWS MOVEMENTS LATERAL	BELLOWS MOVEMENTS TORSIONAL ROTATION	INSULATION THICKNESS INCHES	LOCATED ON TVA DWG. NO.
I	X-13A	MAIN STEAM	32"	29.609"	1.175" MIN. W	SA-155-1 KCF 70	600°	1185	44"	1.012" MIN. W	SA-155 KC 70-1	52"	51.303"	1.250"	SA-516 GR 70	52X44X32	SA-350 GR.LF2		1.278"	2.461"	2.617"	0.109°	3 1/2	47W400
	X-13B																		1.278"	2.461"	2.617"	0.109°		47W400
	X-13C																		1.278"	2.461"	2.617"	0.109°		47W400
	X-13D																		1.278"	2.461"	2.617"	0.109°		47W400
II	X-12A	FEEDWATER	16"	14.484"	80	SA-333 GR6			24"	40	SA-106 GRB	30"	29.303"			30X24X16			1.774"	1.520"	1.659"	0.083°	2 1/2	47W401
	X-12B																		1.774"	1.520"	1.659"	0.085°		47W401
	X-12C																		3.606"	1.520"	1.793"	0.074°		47W401
	X-12D																		3.606"	1.520"	1.793"	0.074°		47W401
III	X-17	SIS-RHR HL. INJ.	12.75"	10.740"	1.125"	SA-376 TP316	650°	2485	20"	100		28"	27.303"			28X20X12	SA-182 GR F304		0.963"	1.220"	0.526"	0°	1 1/2	47W432
III	X-107	RHR SUPPLY	14"	13.192"	.438"	SA-358-1 TP304	400°	600	24"	STD.WT		30"	29.303"			30X24X14	SA-182 GR F304		0.576"	0.939"	0.310"		1 1/2	47W432
IV	X-14A	STEAM GEN. BLOW DOWN	2.375"	—	160	SA-106 GR B	600°	1185	10.75"	40		16"	15.303"	80	SA-333 GR 6	16X10X2	SA-350 GRLF2		1.302"	1.621"	0.615"		2	47W400
	X-14B			—															1.302"	1.621"	0.616"			
	X-14C			—															1.101"	1.542"	0.518"			
	X-14D			—															1.101"	1.542"	0.518"			
	X-15	CHEM & VOLUME SYS LET DN	2.375"	—	40S	SA-312 TP 304	400°	600				18"	17.303"		SA-333 GR6	18X10X2	SA-182 GRF304		0.964"	0.987"	0.526"		1 1/2	47W406
	X-30	ACCUMULATORS TO HOLD UP TK	1.05"	—	.219	SA-376 TP316	650°	2580	10.75"	100		16"	15.303"		SA-333 GR6	16X10X3/4	SA-182 GRF304		0.633"	1.101"	0.359"		2 1/2	47W435
	X-45	RC DRAIN TANK & PRT TO VH	1.315"	—	.133	SA-106 GR B	267°	220	8.625"	40		16"	15.303"		SA-333 GR6	16X8X1	SA-182 GRLF2		0.690"	1.144"	0.431"		1"	47W560
V	X-20A	LOW HEAD SIS (B)	8.625"	6.996"	.906"	SA-376 TP316	650°	2580	18"	120		24"	23.303"		SA-333 GR6	24X18X8	SA-182 GRF304		0.963"	1.220"	0.526"		2 1/2	47W435
	X-20B	LOW HEAD SIS (A)	8.625"	6.998"	.906"	SA-376 TP316	650°		18"	120		24"	23.303"			24X18X8			0.963"	1.220"	0.526"			47W435
	X-21	SIS PUMP DISCH TO HOT LEG	4.50"	3.530"	.531	SA-376 TP316	650°		16"	100		20"	19.303"	1.250"	SA-516 GR 70	20X16X4			0.898"	1.196"	0.491			47W435
	X-22	BORON INJECTION TANK DISCH	3.50"	2.092"	.438	SA-376 TP316	200°	2800	12.75"	80		20"	19.303"	80	SA-333 GR6	20X12X3			0.944"	1.217"	0.525"		1	47W435
	X-24	SIS PUMP R.V. DISCHARGE	4.50"	4.044"	40S	SA-312 TP304	200°	240	14"	STD.WT		20"	19.303"			20X14X4			0.895"	1.001"	0.483"		1	47W465
	X-46	RC DRAIN TANK PUMP DISCH	3.50"	3.081"	40S	SA-312 TP304	200°	240	12.75"	40		18"	17.303"		SA-333 GR6	18X12X3			0.690"	1.144"	0.431"		1	47W560
V	X-47A	GLYCOL OUT	6.625"	6.094"	40	SA-333 GR6	10° 100°	150	14"	STD.WT		20"	19.303"			20X14X6	SA-350 GRLF2		1.643"	1.643"	1.026"		1 1/2	47W462
V	X-47B	GLYCOL IN	6.625"	6.094"	40	SA-333 GR6	10° 100°	150	14"			20"	19.303"			20X14X6	SA-350 GRLF2		1.643"	1.643"	1.026"		1 1/2	47W462
IV	X-81	RC DRAIN TANK TO GAS ANALY	1.05"	—	40S	SA-312 TP304	267°	220	6.625"			12.750"	12.053"		SA-333 GR6	12X6X3/4	SA-182 GRF304		0.576"	1.073"	0.310"		1	47W560
XI	X-14	RHR RECIRC SUCTION	14"	13.192"	.438"	SA-358-1 TYP304	400°	600	24"			30"	—	.375"	SA-155 KC60-1	30X24X14	SA-182 GRF304		0.697"	—	0.226"		1 1/2	47W432
XI	X-15	RHR RECIRC SUCTION	14"	13.192"	.438"	SA-358-1 TYP304	400°	600	24"			30"	—	.375"	SA-155 KC60-1	30X24X14	SA-182 GRF304		0.697"	—	0.226"		1 1/2	47W432
XI	X-16	CONT. SPRAY RECIRC.	12.75"	12.053"	40S	SA-312 TYP304	260°	30	20"			28"	—	.375"	SA-155 KC60-1	28X20X12	SA-182 GRF304		0.697"	—	0.226"		1 1/2	47W437
XI	X-17	CONT. SPRAY RECIRC.	12.75"	12.053"	40S	SA-312 TYP304	260°	30	20"			29"	—	.375"	SA-155 KC60-1	28X20X12	SA-182 GRF304		0.697"	—	0.226"		1 1/2	47W437
V	X-32	SIS PUMP DISCH TO HOT LEG	4.50"	3.530"	.531"	SA-376 TP316	650°	2580	14"	100		20"	19.303"	80	SA-333 GR 6	20X14X4	SA-182 GRF304		0.633"	1.101"	0.359"		2	47W435
V	X-33	SIS PUMP DISCH TO COLD LEG	4.50"	3.530"	.531"	SA-376 TP316	650°	2580	14"	100		20"	19.303"		SA-333 GR 6	20X14X4	SA-182 GRF304		0.633"	1.101"	0.359"		2	47W435
XIV	X-108	SPARE	12.75"	10.74"	140	SA-376 TP316	120°	1815	18"	80	SA-106 GR-B	24"	23.303"			24X18X12	SA-182 GRF304		0.356"	1.130"	0.178"		NONE	47W435-22
XIV	X-109	SPARE	12.75"	10.74"	140	SA-376 TP316	120°	1815	18"	80	SA-106 GR-B	24"	23.303"			24X18X12	SA-182 GRF304		0.356"	1.130"	0.178"		NONE	47W435-22

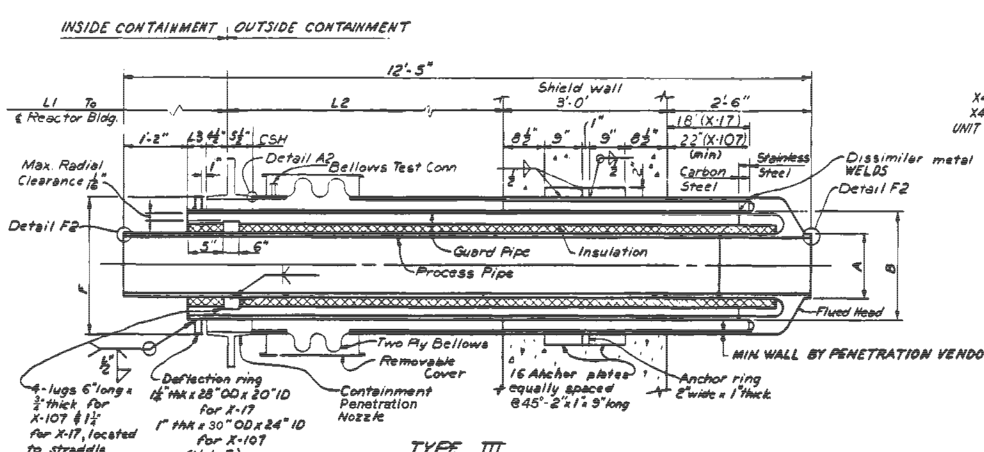


TYPE XIV
UNI X-108 & X-109

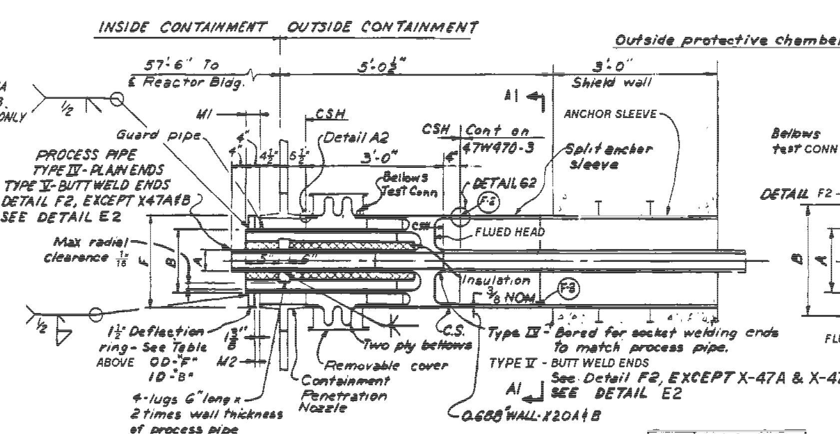


13. FIELD WELD NUMBERS (F1) ETC. DESIGNATE FIELD WELDS. SEE 48N406 FOR CONTAINMENT TO PENITRATION ASSY FIELD WELDS. SEE APPLICABLE DRAVO DWGS FOR PROCESS PIPE FIELD WELDS.
14. FLUED HEAD EXTENSION SPOOLS MUST BE COUNTERBORED A MINIMUM DISTANCE OF 2T WHERE IDENTIFIED AS REQUIRING INSERVICE INSPECTION.
15. DEFLECTION RINGS MAY BE SPLIT IN HALF TO FACILITATE INSTALLATION. THE 2 HALVES ARE NOT REQ'D TO BE WELDED TOGETHER.
16. SEE DRAWING 48N406 FOR PENETRATION LOCATIONS.
17. SOME WELDS SHOWN ON THIS DRAWING (SERIES) WERE FOUND TO BE DEFICIENT BY THE WELD EVALUATION PROJECT, EVALUATED BY WATTS BAR ENGINEERING PROJECT AND FOUND TO BE SUITABLE FOR SERVICE. SEE DESIGN CALCULATIONS FOR EVALUATION. REFER TO RIMS NO. B26890901319 AND ALSO PIR WBN MEB 8669, FOR DOCUMENTATION DEFICIENCY ON 1X-97.
18. CAQR WBP88310 OBTAINED ALTERNATE ACCEPTANCE TO THE CODE OF RECORD PRESSURE TEST FOR UNIT 1 PENETRATION. PENETRATIONS 1-X-8A THRU 1-X-8D, 1-X-12A THRU 1-X-12D, AND 1-X-13A THRU 1-X-13D AND 1-X-99 REQUIRED VISUAL INSPECTION DURING PRESSURE TEST. PENETRATION 1-X-98 IS A SPARE REQUIRING EXAMINATION OF VENDOR WELDS DURING HYDRO IF USED. PENETRATIONS 1-X-34, 1-X-90, 1-X-91, AND 1-X-97 WERE PNEUMATICALLY TESTED TO 1.25 TIMES THE SYSTEM PRESSURE AND ARE ACCEPTABLE. PENETRATIONS 1-X-108 AND 1-X-109 HAVE BEEN CONVERTED TO SPARES AND RELIEF WAS GRANTED ON VISUAL EXAMINATION IN THE EVENT THAT THEY ARE NEEDED IN THE FUTURE.

NOTES:
1. FOR SPECIFICATIONS SEE SPECIFICATION FOR PENETRATION ASSEMBLIES, TVA SPECIFICATION 2529 CONTR. 76K61 83180
2. FOR GENERAL SPECIFICATION SEE TVA SPECIFICATION 1521, Contr 74C38-83015.
3. CONTAINMENT VESSEL PENETRATION NOZZLES BY OTHERS.
4. PENETRATION ASSEMBLIES SHOWN FOR UNIT 1, UNIT 2 ARE IDENTICAL.
5. PENETRATIONS WITH BELLOWS REQUIRE THE PROCESS PIPE TO BE INSULATED BY THE FABRICATOR DURING ASSEMBLY WITH MATERIAL AS PER SPECIFICATION. THE INSULATION SHALL INCLUDE ONLY THAT PORTION OF THE PROCESS PIPE WITHIN GUARD PIPE, AND SHALL EXTEND 1" BEYOND THE END OF GUARD PIPING.
6. BELLOWS OUTLINE IS SHOWN FOR ILLUSTRATIVE PURPOSES. CONTRACTOR TO DETERMINE THE LENGTH AND NUMBER OF CORRUGATIONS.
7. GUIDE RINGS ON OD OF GUARD PIPE FOR PENETRATIONS X12A, B, C, D, X13A, B, C, AND D AND DEFLECTION RINGS ON ALL PENETRATIONS TO BE FIELD WELDED IN PLACE AFTER GUARD PIPE IS INSERTED THROUGH VESSEL PENETRATION NOZZLE.
8. TYPES III, IV, AND V ANCHORED IN THE SHIELD BUILDING WALL.
9. PENETRATION ASSEMBLIES WILL HAVE A PREFIX INDICATING UNIT DESIGNATION, I.E., UNIT 1, 1-X-16, UNIT 2, 2-X-16.
10. CSH DENOTES, "CONTRACT STARTS HERE."
11. BELLOWS FOR K-14, K-15, K-16, AND K-17 HAVE THE FOLLOWING DESIGN CONDITIONS:
    MAXIMUM INTERNAL PRESSURE - 27 PSIG
    MAXIMUM EXTERNAL PRESSURE - 25 PSIG
12. ALL WELDS TO BE SHOP WELDS, UNLESS OTHERWISE NOTED.

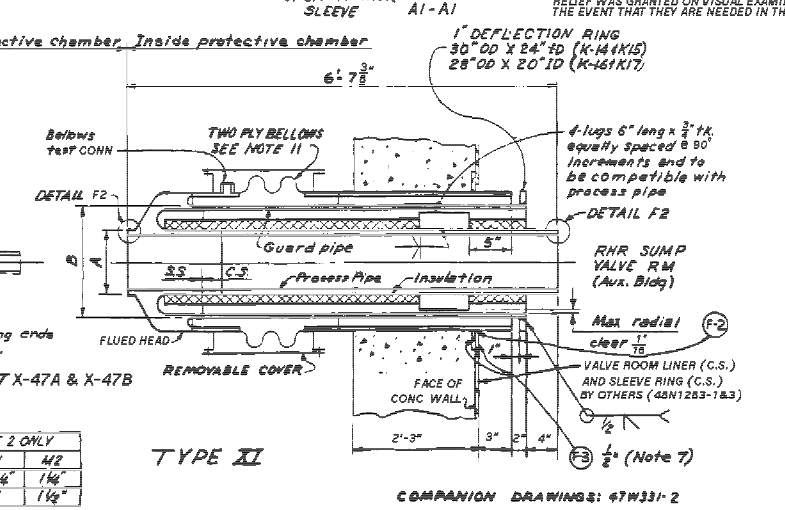


TYPE III
RESIDUAL HEAT REMOVAL
PUMP RETURN X-17
PUMP SUPPLY X-107



TYPE IV & V
TYPE IV SOCKET WELD ENDS
TYPE V BUTT WELD ENDS
FOR PEN. X-47A & X-47B
SEE DWG 47W462

PEN. NO.	UNIT 2 ONLY M1	UNIT 2 ONLY M2
X47A	2 3/4"	1 1/4"
X47B	3"	1 1/2"



TYPE XI

COMPANION DRAWINGS: 47W331-2

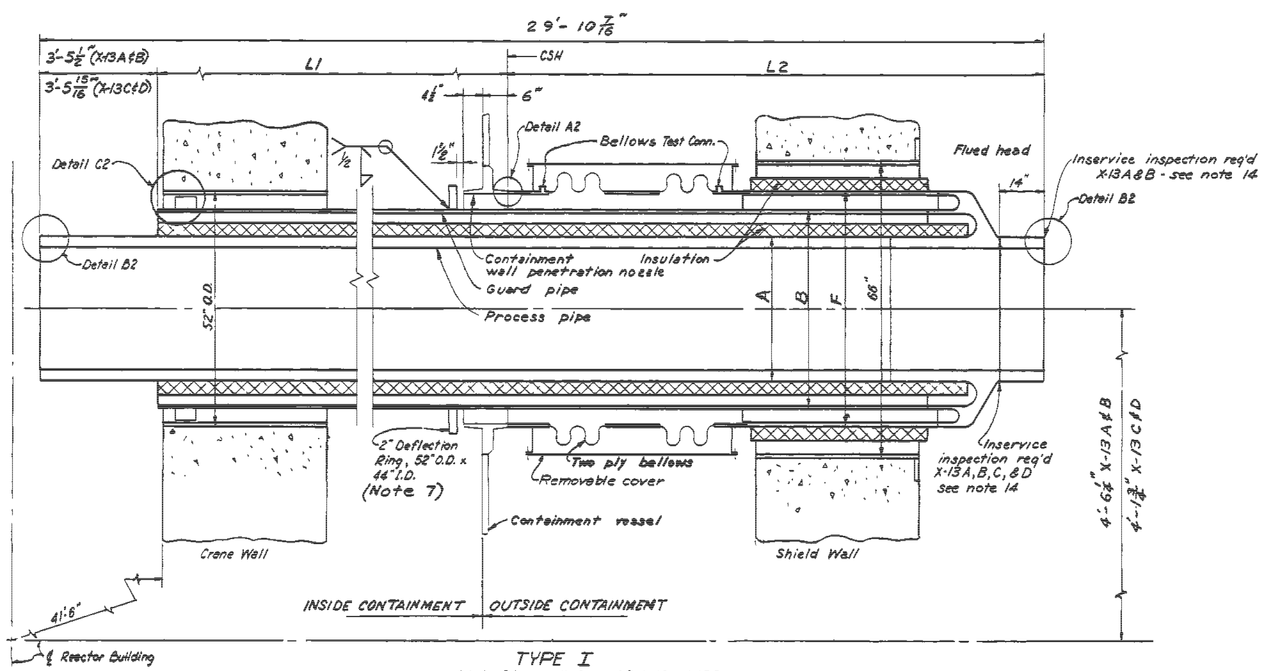
SCALE: NONE

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

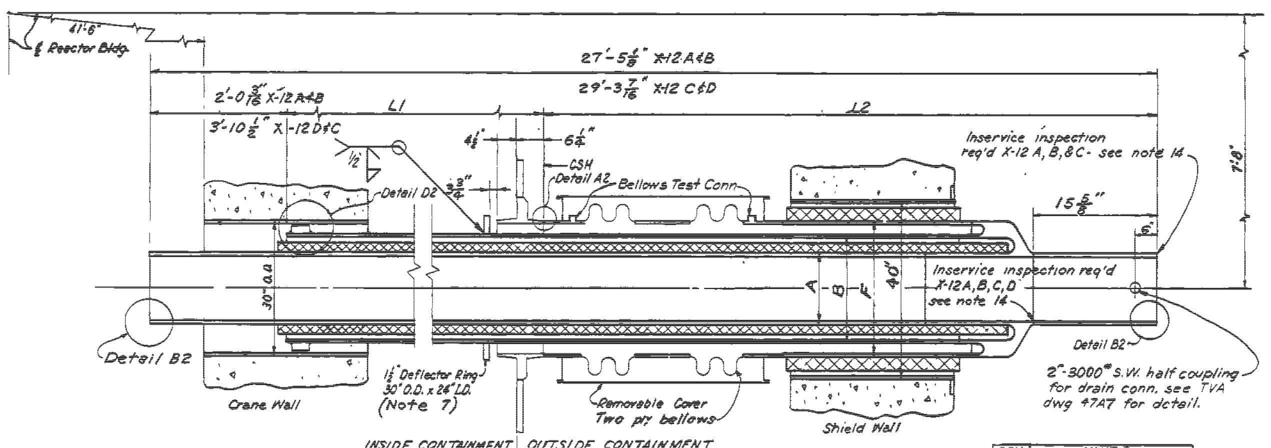
POWERHOUSE
REACTOR BUILDING-UNIT 1
MECHANICAL
CONTAINMENT PENETRATIONS

TVA DWG NO. 47W331-1 RG
FIGURE 3.8.2-8

CAD MAINTAINED DRAWING



TYPE I
Main Steam X-13A, X13B, X-13C, X13D

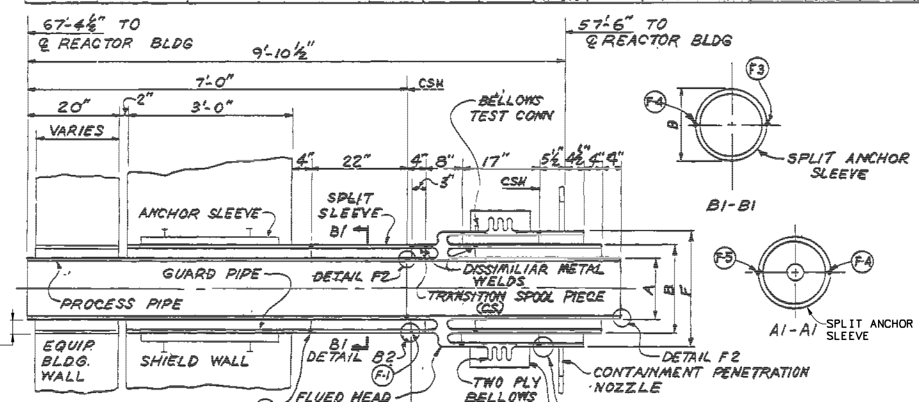


TYPE II
Feedwater X-12A, X12B, X12C, X12D

PEN. No.	UNIT 2 L1	UNIT 2 L2	UNIT 2 L3
X13A	16'-6 3/16"	9'-10 3/4"	NA
X13B	16'-4 11/16"	10'-0 1/4"	NA
X13C	16'-5 1/16"	9'-11 7/16	NA
X13D	16'-5 5/16"	9'-10 9/16"	NA
X12A	15'-1 1/2"	10' 3 5/16"	NA
X12B	15'-0 9/16"	10'-4 1/8"	NA
X12C	15'-0 3/8	10-4 5/16"	NA
X12D	15'-1"	10'-3 3/16"	NA
X17	57'-6 9/16"	4'-11 1/8"	4 5/8"
X107	57'-7 9/16"	4'-10 5/16	5 9/16"

PENETRATIONS WITH BELLOWS

TYPE	PENETRATION NO.	DESCRIPTION	PROCESS PIPE O.D.A	PROCESS PIPE I.D."C"	PROCESS PIPE W. THK OR SCH	PROCESS PIPE ASME MAT'L	PROCESS PIPE DESIGN TEMP F	PROCESS PIPE DESIGN PRESS PSI	GUARD PIPE O.D.B	GUARD PIPE W. THK OR SCH	GUARD PIPE ASME MAT'L	CONT. VESSEL O.D.F	CONT. VESSEL CI "MACH" I.D.	PENETRATION W. THK OR SCH	PENETRATION ASME MAT'L	FLUED HEAD SIZE	FLUED HEAD ASME MAT'L	BELLOWS MOVEMENTS AXIAL EXTEN.	BELLOWS MOVEMENTS AXIAL COMP.	BELLOWS MOVEMENTS LATERAL	BELLOWS MOVEMENTS TORSIONAL ROTATION	INSULATION THICKNESS INCHES	LOCATED ON TVA DWG. NO.
I	X-13A	MAIN STEAM	32"	29.609"	1.175" MIN. W	SA-155-1 KCF 70	600°	1185	44"	1.012" MIN. W	SA-155 KC 70-1	52"	51.303"	1.250"	SA-516 GR 70	52X44X32	SA-350 GRLF2	1.278"	2.461"	2.617"	0.109°	3 1/2	47W400
	X-13B																	1.278"	2.461"	2.617"	0.109°		47W400
	X-13C																	1.278"	2.461"	2.617"	0.109°		47W400
	X-13D																	1.278"	2.461"	2.617"	0.109°		47W400
II	X-12A	FEEDWATER	16"	14.484"	80	SA-333 GR6			24"	40	SA-106 GRB	30"	29.303"			30X24X16		1.774"	1.520"	1.659"	0.083°	2 1/2	47W401
	X-12B																	1.774"	1.520"	1.659"	0.083°		47W401
	X-12C																	3.606"	1.520"	1.793"	0.074°		47W401
	X-12D																	3.606"	1.520"	1.793"	0.074°		47W401
III	X-17	SIS-RHR HL. INJ.	12.75"	10.740"	1.125"	SA-376 TP-316	650°	2485	20"	100		28"	27.303"			28X20X 12	SA-182 GRF304	0.963"	1.220"	0.526"	0°	1 1/2	47W432
III	X-107	RHR SUPPLY	14"	13.192"	.438"	SA-358-1 TP-304	400°	600	24"	STD.WT		30"	29.303"			30X24X 14	SA-182 GRF304	0.576"	0.939"	0.310"		1 1/2	47W432
IV	X-14A	STEAM GEN. BLOW DOWN	2.375"	—	160	SA-106 GR B	600°	1185	10.75"	40		16"	15.303"	80	SA-333 GR 6	16X10X2	SA-350 GRLF2	1.302"	1.621"	0.616"		2	47W400
	X-14B			—														1.302"	1.621"	0.616"			
	X-14C			—														1.101"	1.542"	0.518"			
	X-14D			—														1.101"	1.542"	0.518"			
	X-15	CHEM & VOLUME SYS LET DN	2.375"	—	40S	SA-312 TP 304	400°	600				18"	17.303"		SA-333 GR 6	18X10X2	SA-182 GRF304	0.964"	0.987"	0.526"		1 1/2	47W406
	X-30	ACCUMULATORS TO HOLD UP TK	1.05"	—	.219	SA-376 TP 316	650°	2580	10.75"	100		16"	15.303"		SA-333 GR 6	16X10X3/4	SA-182 GRF304	0.633"	1.101"	0.359"		2 1/2	47W435
	X-45	RC DRAIN TANK & PRT TO VH	1.315"	—	.133	SA-106 GR B	267°	220	8.625"	40		16"	15.303"		SA-333 GR 6	16X8X1	SA-350 GRLF2	0.690"	1.144"	0.431"		1"	47W560
V	X-20A	LOW HEAD SIS (B)	8.625"	6.998"	.906"	SA-376 TP 316	650°	2580	18"	120		24"	23.303"		SA-333 GR 6	24X18X8	SA-182 GRF304	0.963"	1.220"	0.526"		2 1/2	47W435
	X-20B	LOW HEAD SIS (A)	8.625"	6.998"	.906"	SA-376 TP 316	650°		18"	120		24"	23.303"			24X18X8		0.963"	1.220"	0.526"			47W435
	X-21	SIS PUMP DISCH TO HOT LEG	4.50"	3.530"	.531	SA-376 TP 316	650°		16"	100		20"	19.303"	1.250"	SA-516 GR 70	20X16X4		0.898"	1.196"	0.491			47W435
	X-22	BORON INJECTION TANK DISCH	3.50"	2.092"	.438	SA-376 TP 316	200°	2800	12.75"	80		20"	19.303"	80	SA-333 GR 6	20X12X3		0.944"	1.217"	0.525"		1	47W435
	X-24	SIS PUMP R.V. DISCHARGE	4.50"	4.044"	40S	SA-312 TP 304	200°	240	14"	STD.WT		20"	19.303"			20X14X4		0.895"	1.001"	0.483"		1	47W465
	X-46	RC DRAIN TANK PUMP DISCH	3.50"	3.081"	40S	SA-312 TP 304	200°	240	12.75	40		18"	17.303"		SA-333 GR 6	18X12X3		0.690"	1.144"	0.431"		1	47W560
V	X-47A	GLYCOL OUT	6.625"	6.094"	40	SA-333 GR6	-10° 100°	150	14"	STD.WT		20"	19.303"			20X14X6	SA-350 GRLF2	1.643"	1.643"	1.026"		1 1/2	47W462
V	X-47B	GLYCOL IN	6.625"	6.094"	40	SA-333 GR6	-10° 100°	150	14"			20"	19.303"			20X14X6	SA-350 GRLF2	1.643"	1.643"	1.026"		1 1/2	47W462
IV	X-81	RC DRAIN TANK TO GAS ANALY	1.05"	—	40S	SA-312 TP 304	267°	220	6.625"			12.750"	12.053"		SA-333 GR 6	12 X 6 X 3/4	SA-182 GRF304	0.576"	1.073"	0.310"		1	47W560
XI	K-14	RHR RECIRC SUCTION	14"	13.192"	.438"	SA-358-1 TYP 304	400°	600	24"			30"	—	.375"	SA-155 KC60-1	30 X 24 X 14	SA-182 GRF304	0.697"	—	0.226"		1 1/2	47W432
XI	K-15	RHR RECIRC SUCTION	14"	13.192"	.438"	SA-358-1 TYP 304	400°	600	24"			30"	—	.375"	SA-155 KC60-1	30 X 24 X 14	SA-182 GRF304	0.697"	—	0.226"		1 1/2	47W432
XI	K-16	CONT. SPRAY RECIRC	12.75"	12.053"	40 S	SA-312 TYP 304	260°	30	20"			28"	—	.375"	SA-155 KC60-1	28 X 20 X 12	SA-182 GRF304	0.697"	—	0.226"		1 1/2	47W437
XI	K-17	CONT. SPRAY RECIRC	12.75"	12.053"	40 S	SA-312 TYP 304	260°	30	20"			28"	—	.375"	SA-155 KC60-1	28 X 20 X 12	SA-182 GRF304	0.697"	—	0.226"		1 1/2	47W437
V	X-32	SIS PUMP DISCH TO HOT LEG	4.50"	3.530"	.531"	SA-376 TP 316	650°	2580	14"	100		20"	19.303"	80	SA-333 GR 6	20 X 14 X 4	SA-182 GRF304	0.633"	1.101"	0.359"		2	47W435
V	X-33	SIS PUMP DISCH TO COLD LEG	4.50"	3.530"	.531"	SA-376 TP 316	650°	2580	14"	100		20"	19.303"		SA-333 GR 6	20 X 14 X 4	SA-182 GRF304	0.633"	1.101"	0.359"		2	47W435
XIV	X-108	SPARE	12.75"	10.74"	140	SA-376 TP 316	120°	1815	18"	80	SA-106 GR-B	24"	23.303"			24 X 18 X 12	SA-182 GRF304	0.356"	1.130"	0.178"		NONE	47W435-22
XIV	X-109	SPARE	12.75"	10.74"	140	SA-376 TP 316	120°	1815	18"	80	SA-106 GR-B	24"	23.303"			24 X 18 X 12	SA-182 GRF304	0.356"	1.130"	0.178"		NONE	47W435-22

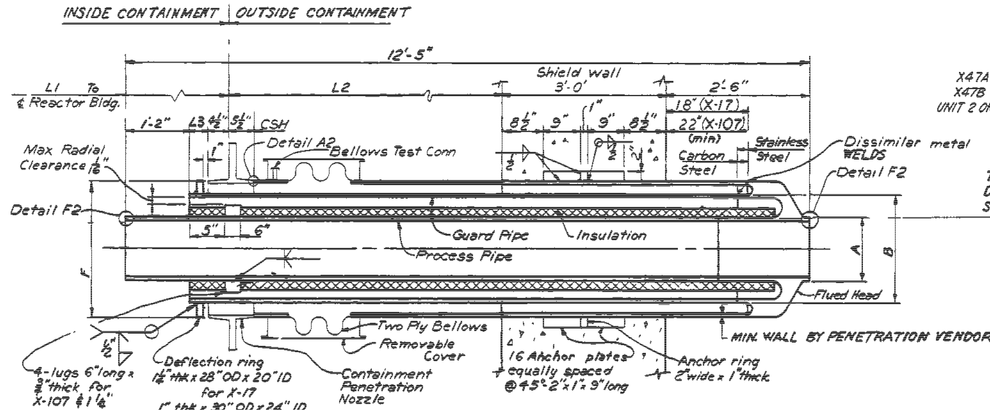


5/8" (MIN)
X-108 ONLY
REF. CR 1059159

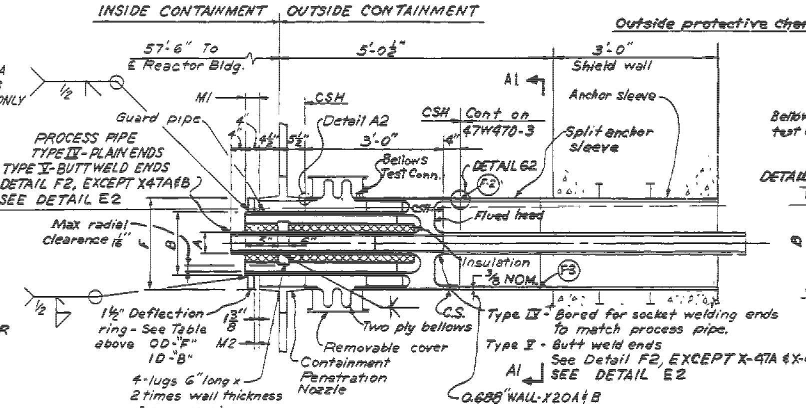
TYPE XIV
UHI X-108 & X-109

16. SEE DRAWING 48N406 FOR PENETRATION LOCATIONS
17. PER 165251 DOCUMENTS HISTORICAL NONCONFORMING CONDITION REPORTS (NCRs) AND SUBSEQUENT INSPECTION REPORTS ON THE BELLOW TYPE PENETRATIONS (SCRATCHES, DENTS, AND/OR GOUGES): 2X-7, 2X-8A, 2X-8B, 2X-8D, 2X-12A, 2X-12B, 2X-12C, 2X-12D, 2X-13A, 2X-13B, 2X-13C, 2X-13D, 2X-14A, 2X-14B, 2X-14C, 2X-14D, 2X-17, 2X-21, 2X-22, 2X-24, 2X-30, 2X-32, 2X-33, 2X-45, 2X-46, 2X-47A, 2X-47B, 2X-81, 2X-107. THESE DAMAGES ARE BEING DOCUMENTED AS A NOTE FOR FUTURE REFERENCE.
18. PER 462739-001 DOCUMENTS A NONCONFORMING CONDITION WITH SLEEVE R2C206. THIS SLEEVE HAS BEEN EVALUATED AS ACCEPTABLE AS IS AND NO FURTHER ACTION IS NECESSARY. THIS CONDITION IS BEING DOCUMENTED AS A NOTE FOR FUTURE REFERENCE.
19. THE FOLLOWING PENETRATION MARK NUMBERS HAVE HISTORICAL BELLOW MISALIGNMENTS: 2X-20A; 2X-20B; 2X-21; 2X-22; 2X-24; 2X-30; 2X-32; 2X-33; 2X-13A; 2X-13B; 2X-13C; 2X-13D; 2X-14A; 2X-14B; 2X-14C; 2X-12B; 2X-12C; 2X-12D; 2X-8D. ACCEPTABILITY OF THESE NONCONFORMANCES WAS DOCUMENTED WITH PERS 143565, 143533 (SUPPLEMENTED WITH PER 413617) AND 143566.
20. SECTIONS OF GUARD PIPE ORIGINALLY CONSTRUCTED FROM SA-155 KC 70-1 CAN BE REPLACED WITH ROLLED PLATE CONSTRUCTED FROM SA 515 GR. 70 OR SA 516 GR. 70.
21. GUARD PIPE IS AN INTEGRAL ATTACHMENT TO THE PRESSURE BOUNDARY, BUT NOT PART OF THE PRESSURE BOUNDARY.

NOTES:
1. FOR SPECIFICATIONS SEE SPECIFICATION FOR PENETRATION ASSEMBLIES, TVA SPECIFICATION 2529 CONTR. 76K61 83180
2. FOR GENERAL SPECIFICATION SEE TVA SPECIFICATION 1521 CONTR. 74C38-83015.
3. CONTAINMENT VESSEL PENETRATION NOZZLES BY OTHERS.
4. PENETRATION ASSEMBLIES SHOWN FOR UNIT 1, UNIT 2 ARE IDENTICAL.
5. PENETRATION WITH BELLOWS REQUIRE THE PROCESS PIPE TO BE INSULATED BY FABRICATOR DURING ASSEMBLY WITH MATERIAL AS PER SPECIFICATION. THE INSULATION SHALL INCLUDE ONLY THAT PORTION OF THE PROCESS PIPE WITH IN GUARD PIPE AND SHALL EXTEND 1" BEYOND THE END OF THE GUARD PIPING.
6. BELLOWS OUTLINE IS SHOWN FOR ILLUSTRATIVE PURPOSES. CONTRACTOR TO DETERMINE THE LENGTH AND NUMBER OF CORRUGATIONS.
7. GUIDE RINGS ON OD OF GUARD PIPE FOR PENETRATIOBNS X12A, B, C, D, X13A, B, C, AND D AND DEFLECTION RINGS ON ALL PENETRATIONS TO BE FIELD WELDED IN PLACE AFTER GUARD PIPE IN INSERTED THROUGH VESSEL PENETRATION NOZZLE.
8. TYPES III, IV, AND V ANCHORED IN THE SHIELD BUILDING WALL.
9. PENETRATION ASSEMBLIES WILL HAVE A PREFIX INDICATING UNIT DESIGNATION, I.E., UNIT 1, 1-X-16, UNIT 2, 2-X-16.
10. CSH DENOTES, "CONTRACT STARTS HERE."
11. BELLOWS FOR K-14, K-15, K-16, AND K-17 HAVE THE FOLLOWING DESIGN CONDITIONS:
    MAXIMUM INTERNAL PRESSURE - 27 PSIG
    MAXIMUM EXTERNAL PRESSURE - 25 PSIG
12. ALL WELDS TO BE SHOP WELDS, UNLESS OTHERWISE NOTED.
13. FIELD WELD NUMBER F-1 ETC. DESIGNATE FIELD WELDS. SEE 48N406 FOR CONTAINMENT TO PENETRATION ASSY FIELD WELDS. SEE APPLICABLE DRAVO DWGS FOR PROCESS PIPE FIELD WELDS.
14. FLUED HEAD EXTENSION SPOOLS MUST BE COUNTERBORED A MINIMUM DISTANCE OF 2T WHERE IDENTIFIED AS REQUIRING INSERVICE INSPECTION.
15. DEFELCTION RINGS MAY BE SPLIT IN HALF TO FACILITATE INSTALLATION. THE 2 HALVES ARE NOT REQUIRED TO BE WELDED TOGETHER.

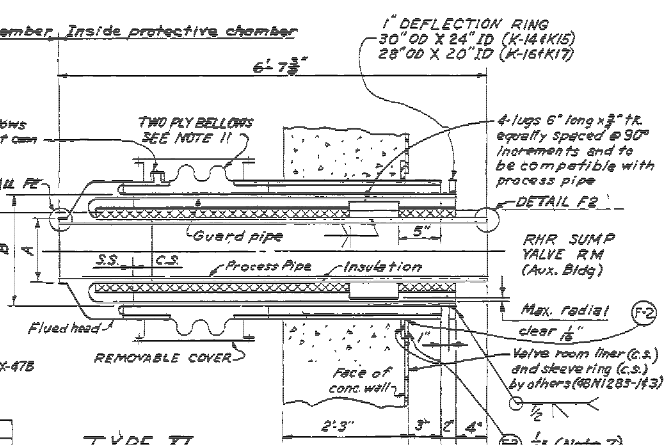


TYPE III
Residual Heat Removal
Pump return X-17
Pump supply X-107



PEN. NO.	UNIT 2 ONLY M1	UNIT 2 ONLY M2
X47A	2 3/4"	1 1/4"
X47B	3"	1 1/2"

TYPE IV & V
Type IV Socket Weld Ends
Type V Butt Weld Ends
For Pen. X-47A & X-47B
See Dwg 47W462



TYPE XI

COMPANION DRAWINGS: 47W331-2

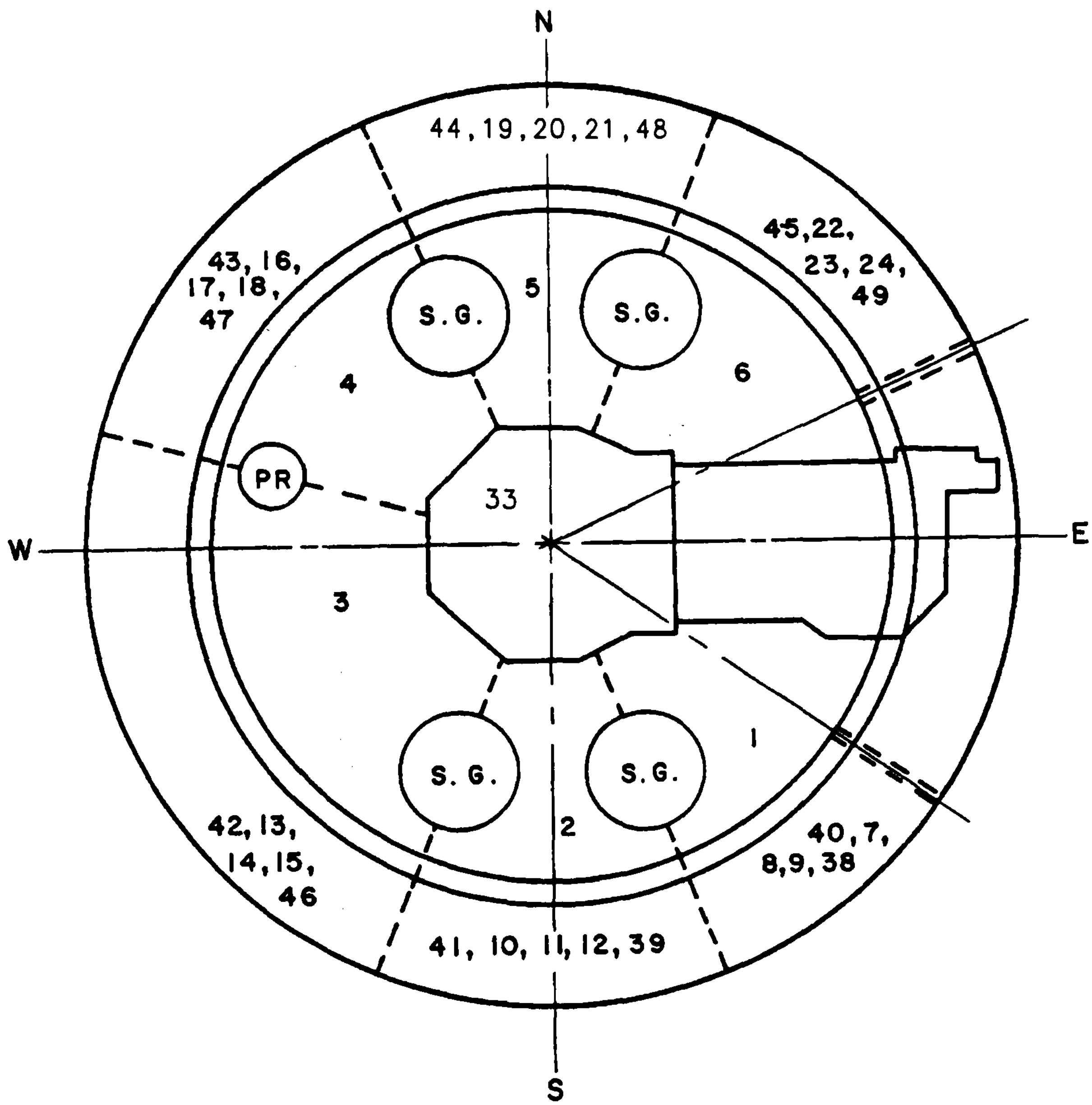
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING-UNIT 2
MECHANICAL
CONTAINMENT PENETRATIONS

TVA DWG NO. 2-47W331-1 R1
FIGURE 3.8.2-8(U2)

FIGURE 3.8.2-9

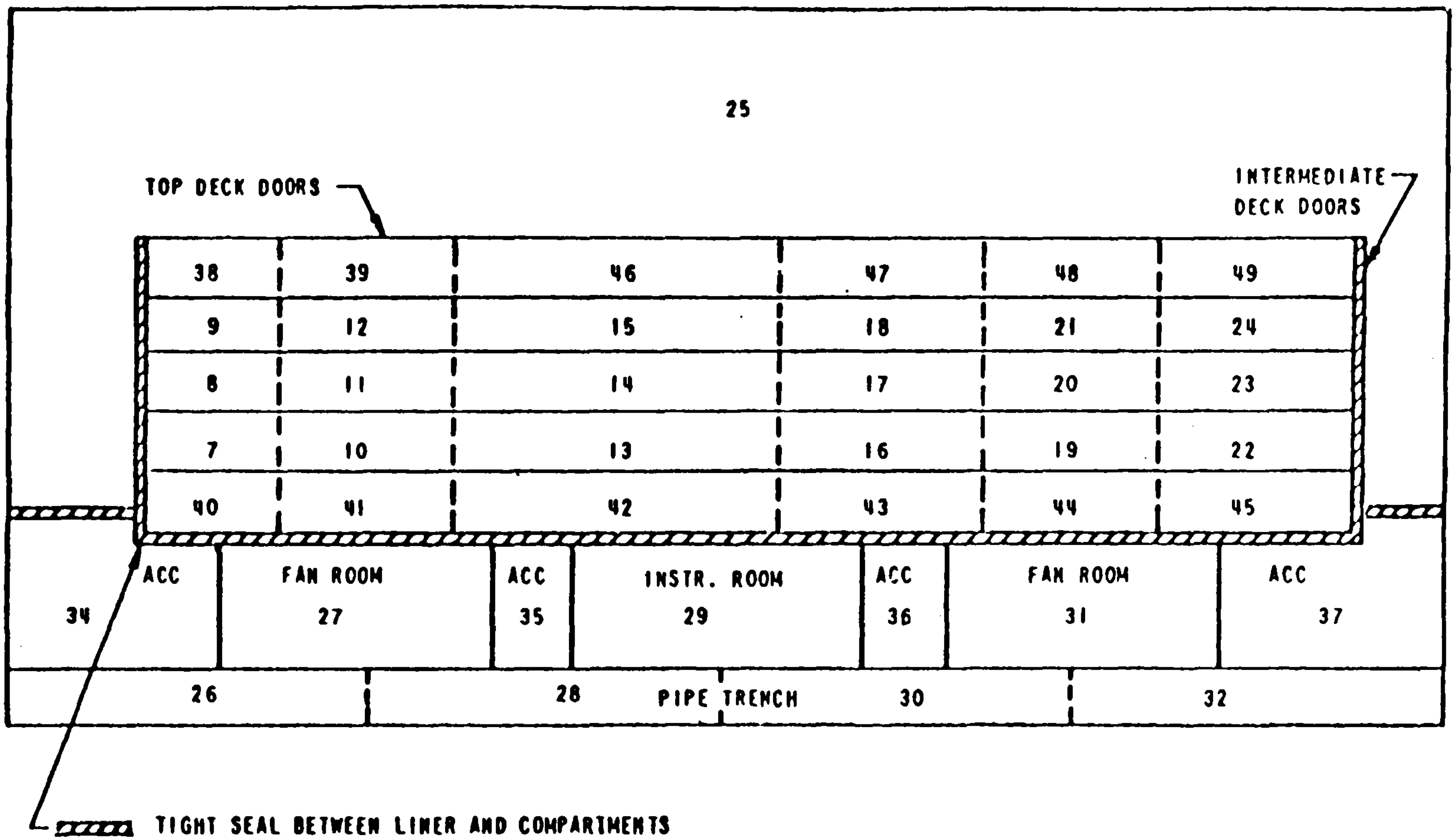
DELETED



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

TMD Nodal Volumes

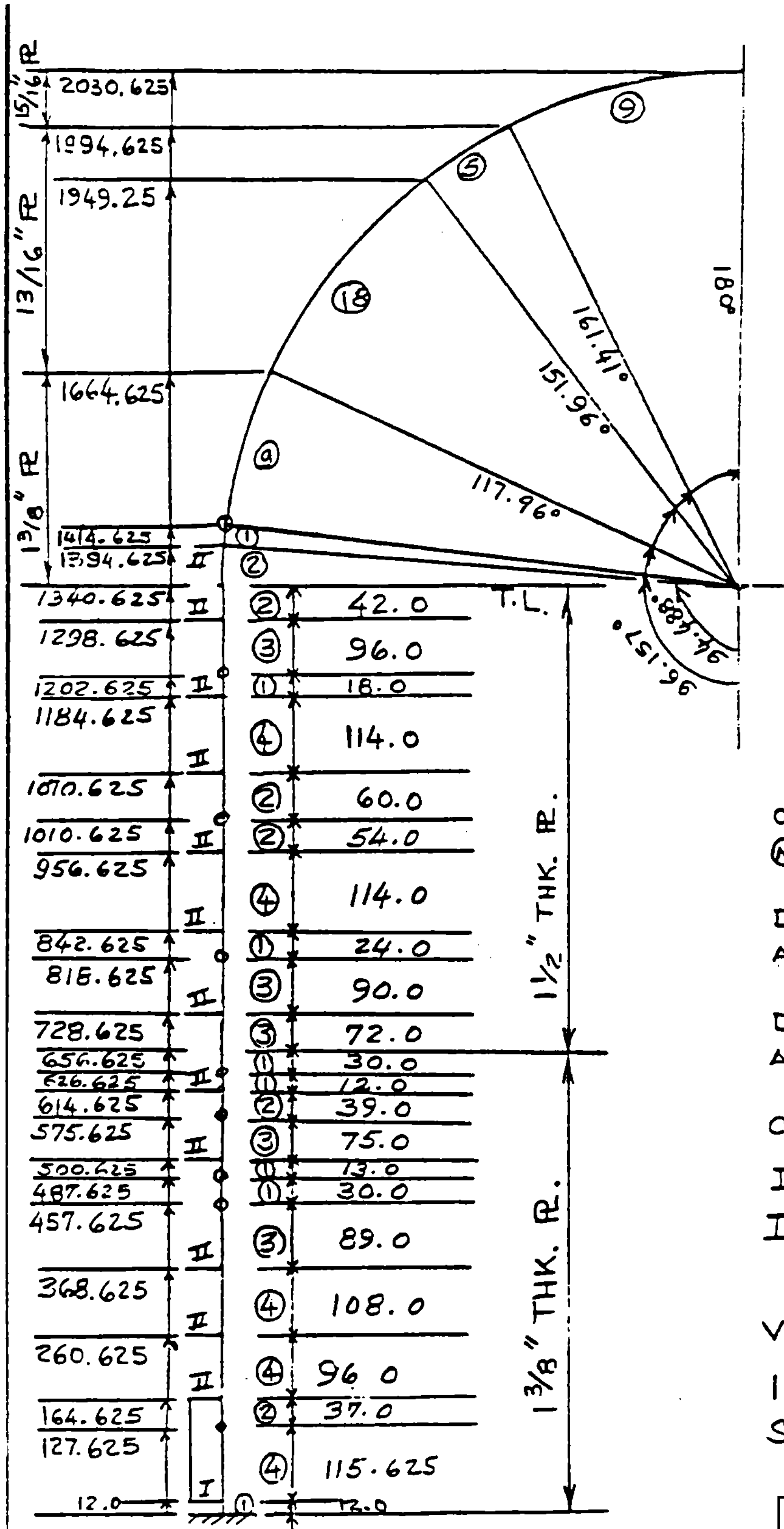
FIGURE 3.8.2-10



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

TMD Nodal Volumes

FIGURE 3.8.2-11



O- PRES. CHANGE LOCATIONS

Ⓝ- NUMBER OF SEGMENTS

DIMENSIONS TO THE RIGHT ARE LENGTHS OF EACH PANEL (IN

DIMENSIONS TO THE LEFT ARE DISTANCES ABOVE BASE (IN

CIRCUMFERENTIAL STIFFENERS

I PL 10"x1"

II PL 22"x1 3/8"

VERTICAL STIFFENER

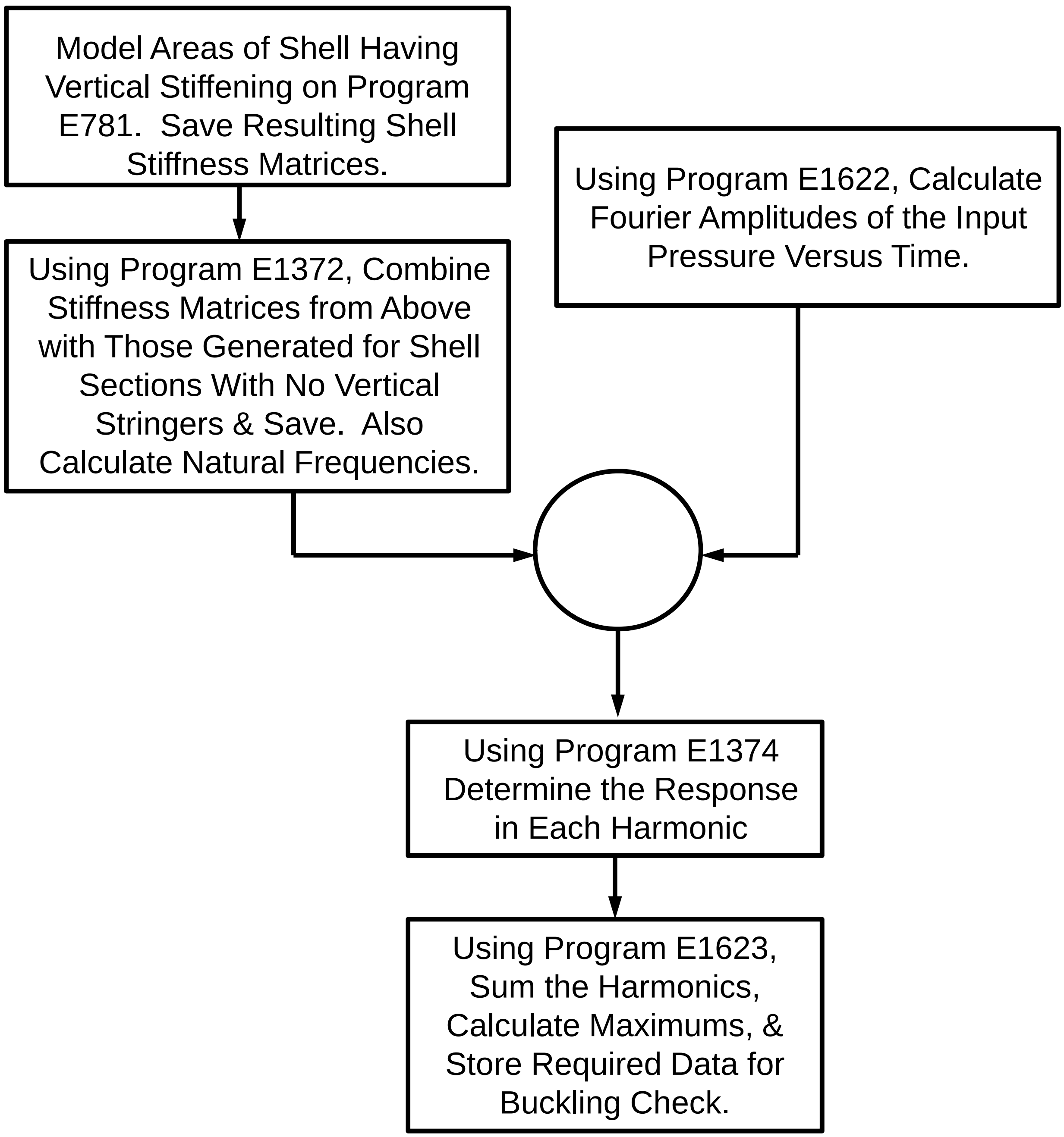
1"x8" @ 5°

SPACING (72 AROUND)

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

CB&I Containment Shell Model

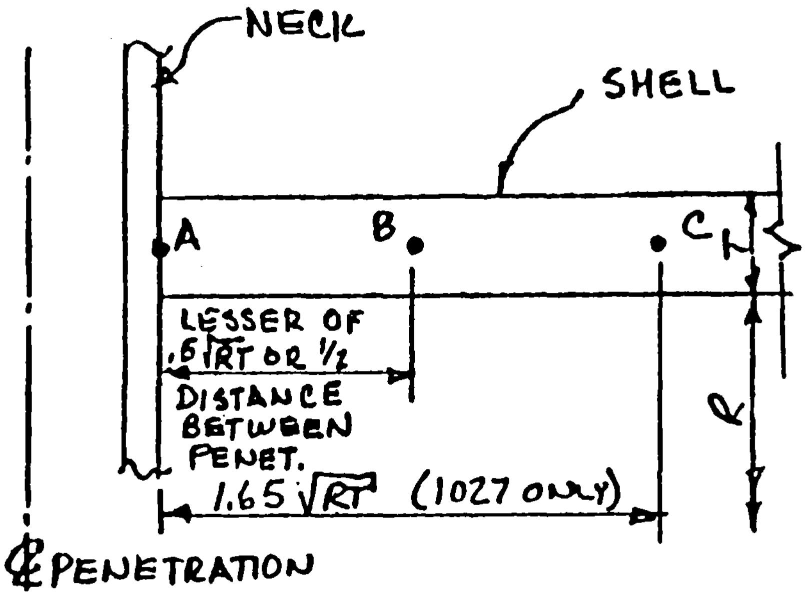
FIGURE 3.8.2-12



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

CB&I Containment Shell
Analysis Flow Model

FIGURE 3.8.2-13



POINT		ASSUMED STRESS				ALLOW STRESS
		CIRCUM		MERID		
		CYL	SPH	CYL	SPH	
A	SURFACE	1½ S_m	1½ S_m	1½ S_m	1½ S_m	3 S_m
	MEMBRANE	S_m	S_m	S_m	S_m	1½ S_m
B	SURFACE	1½ S_m	1½ S_m	1½ S_m	1½ S_m	3 S_m
	MEMBRANE	S_m	S_m	½ S_m	S_m	1.1 S_m
C	SURFACE	S_m	S_m	½ S_m	S_m	3 S_m
	MEMBRANE	S_m	S_m	½ S_m	S_m	1.1 S_m

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Stress Reduction Method

FIGURE 3.8.2-14

3.8.3 Concrete Interior Structure

The concrete interior structures are designed as described in Sections 3.8.3.1 through 3.8.3.8.

The evaluation and the modification of the interior reinforced concrete structures are optionally done using the ultimate strength design method in accordance with the codes, load definitions, and load combinations specified in Appendix 3.8E.

3.8.3.1 Description of the Interior Structure

3.8.3.1.1 General

This structure, shown in Figures 3.8.3-1 through 3.8.3-7, is a complex assemblage of reinforced concrete walls, slabs, and columns housed inside the steel containment vessel (SCV). It will act as a temporary containment while routing steam to and through the ice condenser in the event of a loss of coolant accident (LOCA). The reactor, four steam generators, four reactor coolant pumps, pressurizer, ice condenser, reactor instrumentation, air-handling equipment, and various other support systems are located inside this structure.

The portion of this structure which separates the upper compartment from the lower is defined as the divider barrier (See Figure 3.8.1-1). The failure of any part of the divider barrier is considered critical since it would allow LOCA steam to bypass the ice condenser, thereby increasing the pressure within the steel containment. For this reason the divider barrier is designed more conservatively than the rest of the internal structure.

Since the ice condenser is both a structure and an engineered safeguard system, most detail information can be found in Section 6.5.

3.8.3.1.2 Containment Floor Structural Fill Slab

The containment floor slab is a reinforced concrete slab of 3-foot nominal thickness cast on top of the bottom liner plate. Reinforcement is provided in both faces to withstand uplift pore pressure below the liner plate and to develop restraint for uplift and rotational moments at the base of the crane wall. Earthquake shearing forces are transmitted to the base slab through shear keys located below the crane wall, through a direct tie with the reactor cavity, and through direct bearing on the base of the Shield Building wall as a result of the expanded volume of the fill slab under operating temperatures. Stresses resulting from shear forces are very low since any one of the three methods is capable of transmitting the entire shearing force.

The interior concrete structure is sufficiently keyed to the reactor cavity by its configuration of walls and slabs to provide base stability against earthquake overturning moments above the bottom liner plate at Elevation 699.28. In addition, the anchorages for the steam generators and reactor coolant pumps supports tie the containment structural fill slab to the base slab in the vicinity of the crane wall providing additional stability.

3.8.3.1.3 Reactor Cavity Wall

This 17-foot inside diameter. circular wall supports and encloses the reactor vessel above the lower reactor cavity. The wall is 8-1/2 feet thick, primarily for radiation shielding and structural requirements due to the reactor support loads, and it extends from the base slab at Elevation 702.78 to Elevation 714.96 where it intersects the refueling canal floor slab. Neutron detector windows reduce the effective structural thickness to 6 feet for approximately the first 10 feet of height. The next 12 feet of height has only a 4-foot, 3-inch structural thickness due to the 3-foot, 1-inch wide by 6-foot, 6-inch high inspection cavity which surrounds the reactor vessel. Between the inspection cavity and reactor vessel is a 14-inch-wide structural wall. This is shown in Figures 3.8.3-7a through 3.8.3-7g.

14-Inch Reactor Cavity Bulkhead Wall

This wall consists of two interconnected concentric cylindrical steel shells, separated by 11 inches of concrete fill, that form the reactor cavity from Elevation 721.625 through a composite steel diaphragm to the concrete wall at radius 13.0. This is shown on Figures 3.8.3-7e through 3.8.3-7g.

The anchorage for the wall at Elevation 715.04 is the reactor support embedments directly below the wall.

Both the 14-inch wall and the diaphragm are designed to withstand the pressure and temperature transients resulting from a LOCA condition in accordance with the required factored loading combination in Table 3.8.3-1. The peak differential design pressure and temperature are 50 psi and 150°F, respectively.

3.8.3.1.4 Compartment Above Reactor

This compartment is approximately a 270° arc continuation of the reactor cavity wall. The ends of the arc intersect the two refueling canal side walls. The inside diameter of the wall is 26 feet and the thickness is 4 feet. It extends from approximately Elevation 725.15 to the bottom of the divider deck slab at Elevation 754.13. This compartment is vented to the lower compartment area, outside the wall, by six windows which reduce the wall to five columns. These columns each have a cross sectional area of 12 square feet and extend the last 4-1/2 feet of height to the bottom of the divider deck slab. This compartment is shown in Figure 3.8.3-7b.

During reactor operation this compartment is sealed across the top by the concrete and steel missile shield and is sealed across the refueling canal by a concrete and steel gate.

Seals Between Upper and Lower Compartments

The seals extend across the gap between the inside surface of each steel containment vessel and the concrete structure within each vessel. They are located along the bottom of the concrete floor under the ice condenser, at Elevations 739.5 and 751.33 between the ends of the ice condenser and the refueling canal concrete structure, and along the vertical sides of the refueling canal structure. These seals form part of the barrier between the upper and lower compartments of the containment vessels.

The seals consist of long strips of flexible elastomer coated fabric folded longitudinally with open edges butted and sewn to form two loops in cross section. Metal bars are inserted into the seal for use during attachment. These strips are field-spliced with vulcanized overlay joints or cold bond overlay to form a continuous seal.

The seals are attached to the containment vessel and the interior concrete structure using bolted clamps with bolts spaced one foot apart. These clamps grip the metal bars inserted in the seal thereby closing and sealing the gap.

These seals form part of the barrier between the upper and lower compartments of the containment vessels. During normal operating conditions, the seals prevent airflow around the ice condensers. In an accident, the seals and the other divider parts limit the amount of hot gases, steam, and vapor that can bypass the ice condensers. The seals will maintain their integrity for the first 12 hours after an accident. A small amount of leaking during this period is permissible.

The seals will maintain their integrity during earthquake conditions and effectively maintain their air seal. The seals will function effectively in a post-earthquake condition. The slack in the coated fabric seals, which was purposely provided, allows for the relative movement, between the containment vessel and the interior concrete structure, which results from earthquakes.

3.8.3.1.5 Refueling Canal Walls and Floor (Divider Barrier)

These irregular shaped walls and slabs vary in thickness and enclose an area approximately 19 feet by 32 feet. This area will be filled with water along with the compartment above the reactor during refueling operations. The water level will be about 35 feet above the canal floor slab. The reactor internals will be removed and stored in the refueling canal during refueling. Refueling canal walls and floor are shown in Figure 3.8.3-6.

3.8.3.1.6 Crane Wall

This basically 3-foot-thick, 117-foot-high cylindrical wall encloses an 83-foot inside diameter area containing the reactor, reactor coolant pumps, steam generators, and reactor coolant piping. The crane wall is 4 feet thick in some areas to satisfy structural requirements. There are four localized areas of "bumps" on the wall which are 4 feet thick. These "bumps" begin at the floor slab at Elevation 702.78 and extend to the Elevation 716.0 and are approximately 25 feet in arch length. The crane wall is also 4 feet thick between Elevation 737.42 and Elevation 756.63 over its entire circumference. This wall acts as the major support for the divider barrier slabs and walls. It also supports the floors and walls in the 13-foot annulus between it and the steel containment vessel (SCV). The 175-ton polar crane is mounted on top of this wall. Over the refueling canal the wall has a section removed leaving a curved beam 23 feet deep spanning an arc length of 41 feet between ice condenser compartment end walls. Beginning at Elevation 746.42 the crane wall has twenty-four 7-foot, 4-inch high by 6-foot, 8-inch openings for the ice condenser inlet doors. The remaining wall consists of 25 columns each having a 10-square-foot cross section. Above the operating deck floor at Elevation 756.63, the crane wall is part of the divider barrier. It is also part of the pressurizer and steam generator compartments, which constitute part of the divider barrier, and is designed to resist the same pressures as these compartments. At the top of the crane wall the steel support beams for the ice condenser bridge crane cantilever over the ice beds causing moments and forces in the crane wall. Lateral seismic loads from the ice beds are transmitted to the outer face of the crane wall.

Personnel Access Doors in Crane Wall

See Figures 3.8.3-8 through 3.8.3-11.

Four access doors in the lower half of the crane wall are provided in each Reactor Building at the following locations:

Floor Elevation	Azimuth
702.78	221°
702.78	299°
716.0	114° 16’ 11”
745.0	299°

The doors provide passageways 3-feet wide by 6-feet, 6-inch high through the concrete crane wall for workmen and tools. When closed, the doors seal the passageways against steam jets, pressure, and missiles that may originate from pipe rupture in the compartment inside the crane wall.

Each door is manually operated and hinged to a steel frame embedded in the concrete wall. Each door consists of a steel skin plate stiffened by horizontal framing. The skin plate is faced with a cushioning structure of vertically arranged square, steel tubing separated from the doors skin plate by a collapsible latticework of steel bars, the purpose of which is to absorb the energy of missiles striking the door. The cushioning structure is covered with sheet steel for appearance. Bearing of the door against the frame is through steel bars. An elastomer seal is attached to the periphery of the door to reduce the possibility of damage from jets to items beyond the door. Two lever-type latches operable from either side hold the door in the closed position. Hinges on the doors are provided with graphite impregnated bushings.

The doors, under normal operating conditions, provide an effective seal against airflow and can be operated and secured manually from either side. For pipe rupture accidents, the doors seal the passageways in the crane wall against missiles, jets and pressure that may originate within the crane wall enclosure, thus preventing consequent damage to the containment vessel and to piping and machinery between the crane wall and containment vessel.

The doors will maintain their integrity and seal for not less than the first 12 hours following an accident. Limited leakage during this period is permissible.

All parts of the doors, except the seals, are fireproof. Increased leakage may occur during a fire. It is assumed that a fire and an accident which require sealing will not occur simultaneously since the reactors will be shut down immediately if a fire develops.

3.8.3.1.7 Steam Generator Compartments (Divider Barrier)

Two double-compartment structures house the four steam generators in pairs on opposite sides of the building. Each structure consists of curved and straight sections of walls that vary in thickness from 2½ to 4 feet. Divider barrier walls around two steam generators extend 42 feet up from the divider floor and are capped with a 3-foot-thick slab spanning over the steam generators from the crane wall. A wall between the two steam generators extends from the divider barrier walls to the crane wall, completing the double compartment. The center wall extends only 32½ feet above the floor. The area above the top of this wall, except for that occupied by a beam acting as a barrier for a postulated break in a pipe, will reduce the compartment pressure buildup in a single compartment by venting the steam to the other compartment. See Figures 3.8.1-1 and 3.8.3-6.

During the Unit 1 steam generator replacement, four construction openings in the steam generator compartment concrete roofs were created to facilitate removal of the old steam generators and installation of the replacement steam generators. The compartment roofs were restored by connecting the removed section of concrete to the remaining structure using bolted splice plate connections. The gap between the removed concrete sections and the remaining structure was filled with non-shrink grout to restore the integrity of the divided barrier.

3.8.3.1.8 Pressurizer Compartment (Divider Barrier)

This compartment separates the pressurizer from the upper compartment. Its walls project about 38 feet above the Elevation 756.63 floor where they are capped with a 3-foot-thick slab. It is similar to the steam generator compartments except its wall thickness varies from 2 to 3 feet and the volume is much smaller. See Figure 3.8.3-6.

3.8.3.1.9 Divider Deck at Elevation 756.63 (Divider Barrier)

This 2½-foot-thick irregular shaped floor is the major divider barrier between upper and lower compartments. It is supported at its outer edges by the crane wall and the compartment walls for the steam generators and pressurizer. Support near the center of the building consists of the refueling canal walls and the five columns of the upper reactor compartment. This floor contains five hatches for equipment removal. The concrete covers on these hatches are designed for the same loadings as the floor. The floor outline is shown in Figure 3.8.3-3.

3.8.3.1.10 Ice Condenser Support Floor - Elevation 744.5 (Divider Barrier)

This floor extends 12-feet, 8-inches from the outside of the crane wall to the 4-inch expansion joint separating it from the steel containment vessel. A circumferential beam under its outer edge is cast with the floor. This edge beam is supported by concrete columns which extend down through the Elevation 716 floor to the fill slab at Elevation 702.78. The floor extends 300° around the outside of the crane wall between the ice condenser end walls at azimuths 245° and 305°, as shown in Figure 3.8.3-1.

3.8.3.1.11 Penetrations Through the Divider Barrier

Canal Gate

The canal gate consists of three removable concrete wall elements as illustrated on Figures 3.8.3-2 and 3.8.3-4. The elements are 2-feet, 6-inches thick and span between 7-inch-deep slots formed in the walls of the refueling canal.

Control Rod Drive (CRD) Missile Shield

The CRD missile shield consists of three removable concrete slabs as illustrated on Figures 3.8.3-3 and 3.8.3-4. The slabs are 3-feet, 6-inches thick and are anchored to the divider barrier slab at Elevation 756.63 by anchor bolt assemblies. The details of the anchor bolt assemblies are shown on Figure 3.8.3-14.

Reactor Coolant Pump Access and Lower Compartment Access

Access to the reactor coolant pumps and lower compartment is provided by removable slabs as illustrated on Figure 3.8.3-6. The reactor coolant pump access slabs are approximately 10 feet in diameter and the lower compartment access slab is approximately 6 by 10 feet. Both are 2-feet, 6-inches thick and are anchored to the divider barrier slab by anchor bolt assemblies around the edges. The details of the anchor bolt assemblies are shown on Figure 3.8.3-17.

Equipment Access Hatch

This hatch consists of a removable structural steel framed hatch cover and a structural steel support frame adjacent to the containment vessel. The arrangement and details are illustrated on Figures 3.8.3-6 and 3.8.3-18. The support frame and hatch cover consist of structural steel wide flange sections covered with steel plate. To provide adequate seals between the upper and lower compartment, the side of the frame adjacent to the containment vessel was designed to span from the refueling canal wall to the divider barrier slab, a distance of approximately 5.0 feet. The hatch cover is anchored to the concrete structure by anchor bolt assemblies at each end of the cover.

Escape Hatch

The location of the hatch and the details are shown on Figure 3.8.3-12. The hatch consists of a frame embedded in the divider barrier floor with a hinged and manually operated cover consisting of skin plate stiffened by framing. Quick-acting wheels are provided for opening and closing the cover from either side. Coil springs are incorporated with the hinges to reduce the force required for opening the cover. The hatch is equipped with a limit switch which operates to give an indication in the control room of the position of the hatch cover.

Air Return Duct Penetration

The air return ducts penetrate the divider barrier at two different locations as indicated on Figures 3.8.3-2 and 3.8.3-3. One penetration is at Elevation 746.0 and the other is at Elevation 756.63. The penetrations are 4-foot, 6-inch (inside diameter.) circular openings with flanges on both sides to provide attachment for the ventilating ducts. The details of the penetrations are shown on Figures 3.8.3-15 and 3.8.3-16.

Pressurizer Enclosure Manway

The location and details of the manway are shown of Figure 3.8.3-20. The manway consists of a 30-inch diameter sleeve embedded in concrete at Elevation 798.0 at the top of the pressurizer compartment. The manway cover is a circular steel plate that is bolted in place to provide adequate sealing between the upper and lower compartment.

3.8.3.2 Applicable Codes, Standards and Specifications

Structural design of the interior concrete structures is in compliance with the ACI 318-71 Building Code Requirements for Reinforced Concrete, and ACI-ASME (ACI 359) Article CC 3000 document, "Standard Code for Concrete Reactor Vessels and Containments."

Reinforcing steel conforms to the requirements of ASTM Designation A615, Grade 60.

[*Historical Information* - Installation, inspection and testing requirements for plain and reinforced concrete used in the construction of Category I structures, as well as for fly ash used as an admixture in concrete, were in general accordance with the ASTM standards, ACI 318-71, ANSI N45.2.5 and Regulatory Guides 1.15 and 1.55, except for the following TVA specific requirements:]

1. Required Qualification Tests - *Historical Information*

   a. Fly ash - TVA uses its own specification for fly ash rather than ASTM C 618. Significant differences occur in requirements for fineness, pozzolanic activity index, and loss or ignition. ASTM C 618 has two requirements for fineness. The first, a surface area obtained by an air permeability apparatus, is not conformed to by TVA. The second, the amount retained on a No. 325 sieve, is conformed to by TVA. TVA's requirement for pozzolanic activity index with portland cement is 65% of the ASTM C 618 requirement, but TVA's procedures result in substituting fly ash for fine aggregate in a mix and thus increase the quantity of fly ash available for reaction with the cement. TVA's limit on loss on ignition is 50% of that in ASTM C 618. The most recent addition to ASTM C 618 is a limit on the product of loss on ignition and the amount retained on the No. 325 sieve. This was added when a statistical analysis indicated that it correlated with the effect of fly ash in concrete. TVA's limits on the individual items will result in conformance to the ASTM C 618 limit on the product. TVA's experience at hydro, fossil, and nuclear plants indicates that their specification for fly ash produced acceptable concrete.

   b. Water and ice - TVA complies with the suggested limits of CRD C 400 rather than the suggested limits of AASHO T-26. The suggested limits on compressive strength of mortar are the same. AASHO T-26 utilizes an autoclave soundness test developed specifically to test free lime or magnesia in cement. ASTM C 150 has a specified limit on autoclave expansion of 0.8%, but many elements exhibit less than 0.1%. The ASTM analysis for precision indicates that repeat tests by the same operator can differ 21% by their means. This illustrates that a clear indication of unsoundness due to water will be difficult to obtain. The test for soundness is not recommended in ASTM STP 169-A "Significance of Tests and Properties of Concrete-Making Materials."

c. Concrete mixes - TVA does not conform with two of the recommendations of ACI 211. The recommended limiting water-cement ratios were developed for Type 1 cements. ACI 211 recommendations do not agree with ACI 318. Where fly ash is utilized, neither can be directly applicable. The recommendation that trial batches for strength be made at maximum slump and air contents should not be applied where statistical analysis establishes an average over-strength requirement. Use of maximum air content and slump will offset the average strength and invalidate the analysis.

2. Required In-process Tests - *Historical Information*

a. The construction procedure for this project is in substantial agreement with ANSI N45.2.5 frequencies for those tests required by TVA.

b. Mixer-uniformity - TVA's requirements for unit weight of air-free mortar and for coarse aggregate content are more restrictive than ASTM C 94.

c. Compressive strength - The sampling frequency for compressive strength provided by ANSI N45.2.5 appears to be intended for a transit mix operation. TVA purchase specifications for ready-mix are even more restrictive, however, the vast majority of TVA concrete is produced in a central mix plant where the provided frequency appears excessive. TVA varied the testing frequency requirements based on the specified strength of concrete with no one sample to represent more than:

- 300 cubic yards for a specified strength of 2000 psi.

- 175 cubic yards for specified 28 day strengths of 3000 psi or more.

These test frequencies were in effect until January 1978 during the majority of concrete placement at WBN. The testing frequency requirements were then modified such that no one sample represents more than:

- 300 cubic yards for a specified strength of 2000 psi,

- 200 cubic yards for a specified strength of 3000 psi,

- 150 cubic yards for specified strengths more than 3000 psi.

For actual application, the quantities of each mix produced per shift were such that the average quantities represented by test samples were less than that specified.

d. Aggregate - Tests are specified by ANSI N45.2.5 which appear inappropriate to certain aggregates. A carefully selected crushed limestone fine aggregate should not require testing for organic impurities. TVA required periodic reinspection of the quarry. The quarry strata and weathering effects did not change and therefore testing listed with 6-month frequency in ANSI N45.2.5 were not repeated.

e. Water and ice - (See 1.b above) The chemical tests in CRD C 400 were repeated every 2 months, and any time a change in the water was suspected. The strength test was repeated only when chemical tests results changed significantly.

f. Fly ash was sampled every 3 truck loads and tested for fineness. Six samples were combined and tested for total requirements, see 1.a above.

g. Cement - TVA accepted manufacturers' mill tests which represented no more than 400 tons. TVA made tests at greater intervals which checked manufacturers' strength test within 600 psi or duplicate tests were required.

3. *Historical Information* - TVA's concrete acceptance does not conform to ACI 318. It does conform to ACI 214. TVA requires that no more than 10% of the strength test results be below the specified strength for specified strengths equal to or greater than 3,000 psi. For lower strength concrete, 20% of the strength test results may be below the specified strength. Such concrete is used where a batch of somewhat lower strength concrete is not critical and where hydration temperature limitations are critical. ACI 318 applies the criteria that the averages of all sets of three consecutive strength test results at least equals the specified strength and that not more than 1 of 100 strengths test results will be more than 500 psi below the specified strength. If the standard deviation of the strength test results is 500 psi, the required over-strengths from the three criteria range between 640 psi and 670 psi. TVA does not believe that the three criteria produce significantly different results. ACI 318 states that acceptability is based on no strength test result being more than 500 psi below the specified strength, but its commentary and ACI 359 point out the probability that 1 test in 100 will have results outside the standard deviation and make the ACI criteria more severe.

   TVA's requirement for regular compressive strength tests at 3 days and thorough evaluation requirements if the tested concrete strength deviates from the specified limits provide reasonable assurance that the use of low strength concrete in structures is effectively prevented.

4. *Historical Information* - Personnel qualifications will be maintained as required by Nuclear Quality Assurance Plan.[1]

TVA considers the applicability of ANSI N45.2.6 (Section 1.1, Scope) to be limited to those personnel performing inspection, examination, and test functions. Responsibility for examination and certification of these individuals has been established. These certifications do not correspond to the levels established in ANSI N45.2.6, except for NDE personnel who are certified in accordance with SNT-TC-1A. Construction site inspection, examination, and testing personnel are selected and assigned mechanical, electrical, instrumentation, civil, material, or welding classifications. Responsible supervisors in the respective areas perform the functions identified in Table 1 as L-III in ANSI N45.2.6. Inspection, examination, and testing personnel in the various classifications perform the functioning identified in Table 1 as L-I and L-II in ANSI N45.2.6.

Bolts-Anchors set in hardened concrete were installed in accordance with comprehensive TVA specific requirements developed for the material, installation and testing of these anchors, utilizing anchor manufacturer's instructions as applicable.

Welding, non-destructive examinations, heat treatment, and all field fabrication procedures used during construction were in accordance with the ASME Boiler and Pressure Vessel Code as applicable (see Item 3 below), and the American Welding Society, (AWS) "Structural Welding Code," AWS D1.1 (see Item 5 below). Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 (see Section 3.8.1.2, Item 12) may be used after June 26, 1985, to evaluate weldments that were designed and fabricated to the requirements of AISC/AWS. Unless otherwise indicated in the UFSAR, the design and construction of the interior structures are based upon the appropriate sections of the following codes, standards, and specifications. Modifications to these codes, standards and specifications are made where necessary to meet the specific requirements of the structures. Where date of edition, copyright, or addendum is specified, earlier versions of the listed documents were not used. In some instances, later revisions of the listed documents were used where design safety was not compromised.

1. American Concrete Institute (ACI)

ACI 214-77	Recommended Practices for Evaluation of Strength Test Results of Concrete
ACI 315-74	Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI 359	Standard Code for Concrete Reactor Vessels and Containments (Proposed ACI-ASME Code ACI-359 (Article CC-3300) as issued for trial use April 1973.
ACI 318-71	Building Code Requirements for Reinforced Concrete
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting
ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete
ACI 304-73	Recommended Practice for Measuring Mixing, Transporting, and Placing Concrete

2. American Institute of Steel Construction (AISC) 'Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings,' adopted February 12, 1969.

3. American Society of Mechanical Engineers (ASME), <u>Boiler and Pressure Vessel Code</u> Sections II, III, V, VIII, and IX, 1971 Editions, as amended through summer 1972 Addenda.

4. American Society for Testing and Materials (ASTM), 1975 Annual Book of ASTM Standards.

5. American Welding Society (AWS)

"Structural Welding Code, AWS D1.1-72, with Revisions 1-73 and 2-74 except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders.

Visual inspection of structural welds will meet the minimum requirements of Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 as specified on the design drawings or other engineering design output. See Item 14 below.

AWS D12.1-61, 'Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Connections.'

6. Crane Manufacturers Association of America, Inc. C.M.A.A. No. 70, Specification for Electric Overhead Traveling Cranes, 1971.

7. 'Uniform Building Code,' International Conference of Building Officials, Los Angeles, 1970 Edition.

8. Southern Standard Building Code, 1969 Edition, 1971 Revision.

9. 'Nuclear Reactors and Earthquakes,' USAEC Report TID-7024, August, 1963.

10. American Society of Civil Engineers Transactions Volume 126, Part II, Paper No. 3269, 'Wind Forces on Structures,' 1961.

11. Code of Federal Regulations Title 29, Chapter XVII, 'Occupational Safety and Health Standards,'Part 1910.

12. NRC Regulatory Guides (RG)

RG 1.12 Instrumentation for Earthquakes
RG 1.31 Control of Ferrite Content in Stainless Steel Weld Metal
RG 1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures
RG 1.15 Testing of Reinforcing Bars for Category I Concrete Structures
RG 1.55 Concrete Placement in Category I Structures

13. Structural Engineer Association of California, 'Recommended Lateral Force Requirements and Commentary;' 1968 Edition.

14. Nuclear Construction Issues Group (NCIG)

NCIG-01, Revision 2 - Visual Weld Acceptance Criteria (VWAC) for Structural Welding

NCIG-02, Revision 0 - Sampling Plan for Visual Reinspection of Welds

The referenced NCIG documents may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS.

NCIG-02, Revision 0, was used as the original basis for the Department of Energy (DOE) Weld Evaluation Project (WEP) EG&G Idaho, Incorporated, statistical assessment of TVA performed welding at WBNP. Any further sampling reinspections of structural welds subsequent to issuance of NCIG-02, Revision 2, are performed in accordance with NCIG-02, Revision 2 requirements.

The applicability of the NCIG documents is specified in controlled design output documents such as drawings and construction specifications. Inspectors performing visual weld examination to the criteria of NCIG-01 are trained in the subject criteria.

15. TVA Reports

CEB 86-12 - Study of Long Term Concrete Strength at Sequoyah and Watts Bar Nuclear Plants

CEB 86-19-C - Concrete Quality Evaluation

16. NRC Standard Review Plan, NUREG-0800, Rev. 2, Section 6.2.1.2, "Subcompartment Analysis".

3.8.3.3 Loads and Loading Combinations

Loading combinations and allowable stresses are shown in Table 3.8.3-1. General loads are described below.

Dead Loads

These loads consist of the weight of the structure and equipment, plus any other permanent load contributing stress such as hydrostatic pressure.

Live Loads

These are movable loads such as loads which occur during servicing equipment, crane loads, and water loads due to temporary flooding of various compartments.

Normal Temperature

These are the straight line temperature gradients which exist through member thicknesses due to differences in operating temperatures of various compartments.

LOCA Pressure

These loads are time-varying pressure differentials that will result between compartments in the event of a double-ended-break of a reactor coolant pipe, as discussed in Chapter 6. They vary

in magnitude depending on the location of the pipe break. During the construction permit stage, the maximum calculated differential compartment pressures were increased by 40% in accordance with NRC requirements to account for uncertainties. At the operating license stage, the design pressures equaled or exceeded the peak calculated differential pressures. Dynamic load factors were not applied in the structural analysis except for the ice condenser support floor, the walls at the end of the ice condenser compartment, and the beam over the main steam pipe in the steam generator enclosures since the vibration period of other components in the structure is small in comparison to the rate of application and duration of the pressure loads.

LOCA Temperature

Time-varying nonlinear temperature gradients due to a LOCA cause stresses in the restrained members of this structure. These gradients will vary depending upon time and member location in relation to the pipe break. Stresses were computed for these loadings using a TVA developed program which has the same basic assumptions as the Reinforced Concrete Chimneys Code (ACI 505-54). A typical gradient for the divider floor, is shown in Figure 3.8.3-19.

Creep and Shrinkage

Creep was not considered in the design of interior concrete for the reasons outlined in Section 3.8.1.4.

Shrinkage effects were considered as outlined in Section 3.8.1.4. The peak hydration temperature of the concrete used in the interior structures was estimated to be approximately 130° F for summer placement with controlled placing temperatures of 65°F. From Figure 3.8.3-19 the average normal operating temperature for combination of shrinkage effects with other loads was 80°F resulting in a design temperature drop of 50°F. Under LOCA temperature gradients average temperatures exceed hydration temperatures and shrinkage stresses are relieved. Therefore, shrinkage effects are considered only with normal operating temperature gradients.

Operating Basis Earthquake (OBE)

Reference Section 3.7.2.

Safe Shutdown Earthquake (SSE)

This is the maximum postulated earthquake the plant is designed to withstand and still permit a safe shutdown. Reference Section 3.7.2.

Pipe Forces

These forces are the pressure jet effects that can occur due to breaks in the system's piping. They may be the jet force impinging upon the structure, or the equipment and piping anchorage forces as the result of such a jet. The static equivalent of the major equipment anchorage loadings were furnished by Westinghouse Corporation, the support designer.

Jet forces from postulated pipe ruptures, both longitudinal and transverse, were assumed to load the interior concrete structure.

$F = (1.2pA)k$ where: F = Force on structure, lbs

p = Pressure, psi

A = Inside cross sectional area of pipe, in^2

k = Load factor

The effects of jet force was taken in combination with the uniform compartment differential pressure.

A minimum load factor k = 1.3 was used with both conditions based on localized yielding of the structural member and a ductility factor of 3.

The only jet force in the compartment above the reactor cavity is from the control rod opening. This is due to a pressure of 2250 Psi. The reactor coolant pipe will not produce a jet force in this area.

<u>Missiles</u>

The systems located inside the reactor containment have been examined to identify and classify potential missiles. The basic approach is to assure design adequacy against generation of missiles, rather than allow missile formation and try to contain their effects. Reference Section 3.5.

<u>Ice Condenser Loads and Loading Combinations</u>

The ice bed structure shall be designed to meet the loads described below within the behavior criteria limits presented in Section 3.8.3.5 of these criteria. The following load combinations are defined for design purposes:

1. Dead Load + Operating Basis Earthquake Loads (D + OBE)*.

2. Dead Load + Accident induced loads (D + DBA).

3. Dead Load + Safe Shutdown Earthquake (D + SSE).

4. Dead Load + Safe Shutdown Earthquake + Accident induced loads (D + SSE + DBA).

*Includes thermal induced load and D+L

The loads are defined as follows:

Dead load (D) - Weight of structural steel and full ice bed at the maximum ice load specified.

Live Load (L) - Live load includes any erection and maintenance loads, and loads during the filling and weighing operation.

Thermal Induced Load - Includes those loads resulting from differential thermal expansion during operation plus any loads induced by the cooling of ice containment structure from an assumed ambient temperature at the time of installation.

Accident Fluid Dynamic and Pressure Loads (DBA) - Accident pressure load includes those loads induced by any pressure differential drag loads across the ice beds, and loads due to change in momentum.

Operating Basis Earthquake (OBE) - As previously defined.

Safe Shutdown Earthquake (SSE) - As previously defined.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 General

Each component of the interior concrete structure was considered individually. Its boundary conditions and degrees of fixity were established by comparative stiffness; loads were applied, and moments, shears, and direct loads determined by either moment distribution or finite element methods of analysis. Reinforcing steel was proportioned for the component sections using the allowable stresses given in Table 3.8.3-1, the provisions of the ACI 318-71 Building Code and the proposed Standard Code for Concrete Reactor Vessels and Containments, ACI-ASME (ACI-359) Code, as issued for trial use, April 1973.

During the construction stage, a factor of 1.4 was applied to the design pressures resulting from LOCA. The structure was designed using the 40% margin and the recommendations of the ACI-ASME Joint Committee contained in Proposed Standard Code for Concrete Reactor Vessels and Containment. The results are tabulated in Table 3.8.3-2. NRC Standard Review Plan, NUREG-0800, Rev. 2, Section 6.2.1.2, Section II.B.5, permits reduction of design pressure so that the peak calculated differential pressure does not exceed the design pressure. This reduction in design pressure was utilized in the review of the concrete strength evaluation.[2]

A completely independent design was performed on all portions of the divider barrier. Procedures used in this design and analysis are discussed in Sections 3.8.3.4.3 through 3.8.3.4.13.

3.8.3.4.2 Structural Fill Slab on Containment Floor

The fill slab is designed to span between walls with hydrostatic uplift pressure on 100% of its bottom face from water surface at Elevation 710. Loads from the steam generators and reactor coolant pumps are transferred directly into the base mat by continuous steel connections through the liner plate. As the base mat deflects under load, the fill slab deflects with it and is designed for these deflections. Analysis was made using the effects of crane wall uplift and rotation as well as the uplift of the support columns for the Elevation 716.0 floor. The fill slab is also designed for loads imposed upon it by the reactor coolant pipe crossover supports. Classical deflection formulas, as well as the computer code ICES STRUDL II, were used to determine moments, shears, and reactions. The finite element method of analysis was used to determine direct stresses in shear keys.

3.8.3.4.3 Reactor Cavity Wall

In the event of a circumferential split of a reactor coolant pipe at a reactor vessel nozzle, a nonsymmetric pressure occurs in the inspection cavity region. The pressures resulting from this break, a maximum of 148 psi, are applied to the design of the 8-foot, 6-inch thick and 4-foot, 3-inch thick reactor cavity wall. Large horizontal restraint forces from the steam generators and reactor coolant pumps of approximately 2,200 kips will be transmitted to the 4-foot, 3-inch thick section of wall.

A linear temperature gradient will occur during reactor operation. In the event of a LOCA the reactor is shut down and time-varying nonlinear, gradients similar to those in Figure 3.8.3-19 will occur in the wall. All temperature gradient cases are considered in the design up to a time of 48 hours following a LOCA.

Radiation generated heat on the structures is considered only for the primary shield immediately next to the reactor vessel. There the radiation generated heat is obtained as a function of position of the reactor core with respect to the structure and the temperature distribution is calculated. The effect of the temperature on the structure is then evaluated.

The average temperature of the wall during reactor operation exceeds hydration temperatures during construction. Therefore, tensile stress from hydration temperature considerations will be less than the stress induced by the temperature gradient. Long-time creep relaxation can be expected to substantially reduce temperature stresses; however, such a reduction was not utilized since the effective operating temperature differential across the wall of the reactor cavity was only 35°F. The 8½-foot-thick portion of the wall is basically a thick cylinder. Formulas for stresses in thick cylinders were used to determine the ring moments and tensile stresses induced by the pressure loading and thermal gradients. The 8½-foot-thick portion of the wall was also analyzed as fixed at its base, Elevation 702.78, and an analysis for vertical moments and forces was made for LOCA, reactor support loads and thermal effects. The 4-foot structural portion from the top of the 8½-foot-thick ring to the top of the reactor continues in the same shape and configuration as the compartment above the reactor. This portion of the wall was modeled as a continuation of the compartment above the reactor using the computer program SAP IV (1973). The thin plate and shell element of this program was utilized. The wall was considered fixed at its base by the 8-1/2-foot-thick ring. The large openings for the reactor coolant system pipes occur in this portion of the wall and they were taken into account in the computer model. The axisymmetric pressure due to a reactor coolant pipe break was applied to the wall as well as the large concentrated forces from the steam generator and reactor coolant pump anchorages. Moments, shears, axial loads, and displacements were obtained from the computer analysis and reinforcing selection was made from these results.

Independent Design - *Historical Information*

The analysis of the reactor cavity was approached from the standpoint that a dynamic and inelastic type analysis might be required. However, comparison of the natural frequencies of the structure and the shock spectra of the pressure transients as forcing functions indicated that dynamic amplification was negligible. The results of static stress analyses showed that the stresses in the concrete were generally less than the cracking stress of concrete except in the immediate area of the break. Consequently, only a static elastic analysis was necessary. Analyses were performed for the pressure loads at specific times combined with support loads from reactor coolant loop and reactor pressure vessel. Thermal stresses resulting from this LOCA were also combined directly with pressure-induced stress.

The structural model used in the independent review was a three-dimensional assemblage of intersecting walls simulated by multiple layers of solid isoparametric finite elements. The

general purpose computer program ANSYS, a well documented and widely used program, was used to calculate stress intensities. The reactor cavity structure was considered fixed at the intersection of the reactor cavity columns and operating deck and at the base.

A detailed analysis to determine unit stresses in the concrete and reinforcing steel was based on a cracked section and evaluated using the allowable stresses given in Table 3.8.3-1.

3.8.3.4.4 Compartment Above Reactor

This compartment, which has a design internal pressure of 32 psi, will also be subjected to water pressure when the refueling canal is full of water and refueling of the reactor is taking place. Wave effects of the water during earthquake are taken into account.

This compartment was analyzed in conjunction with the refueling canal walls using the SAP IV (1973) computer program. The reactor cavity wall, as well as the compartment above the reactor, are post-tensioned during construction. Tendons are located at approximately 45° increments around the wall at seven locations. These post-tensioning tendons are stressed as soon as the concrete in the walls has reached a proper strength. This stressing operation puts a 1,000 kips compressive load in the concrete walls at each of the tendon locations. This compressive force is taken into consideration in the reinforcing design of both the reactor cavity and the compartment above the reactor walls.

Independent Design - *Historical Information*

This compartment was analyzed in conjunction with the reactor cavity wall as described in Section 3.8.3.4.3. The 4-foot-thick compartment wall was designed for, a) a break at the reactor vessel nozzle in the lower cavity, b) a break in a primary coolant loop outside the reactor cavity structure along with associated support loads, c) a hydrostatic load applied to the interior face during refueling, and d) CRD mechanism restraint loads. The compressive stress due to post-tensioning of the walls was considered in the formulation of the finite element model.

3.8.3.4.5 Seals Between Upper and Lower Compartments

The design of the seals was by TVA without the use of a computer program.

The flexible coated fabric part of the seal was considered as a thin-wall half cylinder as the fabric width was sized to form an approximate semicircle when subject to internal pressure. With the semicircle, there is adequate slack in the seals to provide for relative movement between the attaching surfaces during all conditions without damage to the seals.

Earthquakes are the only natural environmental conditions which apply to the seals. The seals, being inside the containment vessel are protected from floods, wind, tornadoes, snow and ice. The seals are not in the area affected by missiles and therefore were not designed for missiles.

3.8.3.4.6 Refueling Canal Walls and Floor (Divider Barrier)

Primary Design

The canal walls and slab are designed to take the gravity and earthquake forces from the upper and lower internals storage stand. The face of the walls inside the lower compartment will be subject to a maximum LOCA pressure of 24 psi and localized jet forces due to a LOCA. The walls are subject to concentrated forces and moments from the reactor coolant pump restraints. The walls are subject to an uplift condition due to LOCA pressure acting on the divider barrier slab at Elevation 754.13. The canal walls and slab are designed for the water pressure in the canal during refueling operations. The seismic effect on this water was also considered.

The walls of the refueling canal and the compartment above the reactor were analyzed as a unit consisting of both straight and curved sections of walls. These walls were analytically modeled using the SAP IV (1973) finite element computer program. Shell curved-rectangular elements were used in the mesh assembly and spring constants were used to represent the stiffnesses of walls and slabs framing into the canal walls. Spring constants were used at the intersection of the canal walls with the crane wall, operating deck slab, and the canal floor slab.

The refueling canal slab was analyzed using the STRUDL finite element computer program. Both the rectangular and triangular flat plate element were used in the analysis.

Independent Design - *Historical Information*

1. Refueling Canal Floor

   The refueling canal floor slab was modeled and analyzed utilizing the SAP IV finite element program. The floor, due to its irregular shape, was analyzed in two sections. The larger area of the floor inside the crane wall radius was analyzed utilizing a finite element plate bending model. Boundary conditions were input reflecting the relative stiffnesses of the supporting walls. The slab was fixed at the crane wall boundary. A smaller segment beyond the crane wall support was analyzed using the conservative "strip" method of analysis.

   SAP IV is a general structural analysis program for the static and dynamic analysis of linearly elastic structures. This program was developed and published by Bathe, Wilson, and Peterson, The College of Engineering, University of California at Berkeley, in June 1973, and has seen extensive usage since that time. The plate element used in the analysis is a quadrilateral of arbitrary geometry formed from four compatible triangles. A constant strain triangle was used to represent membrane stresses and the LCCT9 element was incorporated to represent bending behavior.

   The design loads consisted of base plate forces from the upper and lower reactor internals, and fluid pressure forces from the flooded state during refueling. Earthquake and thermal gradient loading was also considered. Factored and unfactored load cases were checked and the reinforcing was sized for maximum stresses using the criteria of Table 3.8.3-1.

2. Refueling Canal Walls

These walls were analyzed in conjunction with the reactor cavity walls as described in Section 3.8.3.4.3. Primary loads considered were, a) a break in a main steam line outside the reactor cavity with a resulting jet force and associated reactor coolant system support loads, b) a hydrostatic load during refueling with associated upper and lower internal support stand loads, and c) the effects of a main steam line break inside the reactor cavity. The model employed is described in the aforementioned Section.

3.8.3.4.7 Crane Wall

Wall Below Operating Deck

In the lower compartment the crane wall is subject to jet forces due to a possible break in the reactor coolant or main steam piping. The largest of these postulated jet forces is 2,650 kips occurring at a crossover leg between a steam generator and a reactor coolant pump. In this same area, the crane wall is subject to a large missile impingement load. Other areas of the crane wall in the lower compartment are exposed to uniform pressure differentials of approximately 23 psi.

The steam generators and reactor coolant pumps are braced laterally with restraints anchored into the crane wall. These restraints impose large concentrated loads on the wall. The largest of these loads is approximately 2,300 kips.

Crane wall temperature gradients, before and after a LOCA, were investigated. At several elevations in the crane wall, maximum and minimum vertical loads were computed using results from the "Dynamic Earthquake Analysis of the Interior Concrete Structure, prepared by TVA. In addition, various parts of the crane wall were designed to handle concentrated loads, 100 to 300 kips, resulting from breaks in small piping systems.

The crane wall was analyzed by isolating areas spanning between slabs and cross walls. Moments, shears, and axial forces were calculated using the STRUDL finite element program. Fixed-end moments were distributed between adjacent sections of wall using conventional distribution methods.

Columns between ice condenser doors are subjected to moments and forces distributed from the ice condenser floor and divider barrier floor, as well as moments, shears, and axial forces from the wall sections above and below the columns. The columns were designed for these moments, shears, and axial forces plus earthquake loads. The columns in the vicinity of the steam generator and pressurizer compartments are greatly influenced by the lateral restraints from the aforementioned equipment, which is anchored in the crane wall immediately at the top of the columns.

Personnel Access Doors in the Crane Wall

Main structural members of the doors were considered as simple beams. Energy absorbing members were considered as collapsible members. Members of the embedded frames were considered as being rigidly supported by concrete. Loads from the embedded frames are transferred to the concrete by embedded anchors.

Design of the doors and embedded frames was by TVA without the use of a computer program. Design of collapsible members on the doors was based on tests made by Oak Ridge National Laboratory. Results of these tests are recorded in their publication titled Structural Analysis of Shipping Casks, Volume 9, "Energy Absorption Capabilities of Plastically Deformed Struts Under Specified Impact Loading Conditions." Collapsible members were sized to limit loads transmitted to the embedded frame to 13,000 pounds per linear inch.

The doors were designed to function during normal conditions, earthquakes, and pipe rupture accidents.

Earthquakes are the only natural environmental condition which applies to the doors. The doors are protected from flood, wind, tornadoes, ice and snow, since they are located inside the containment vessels.

The doors will be closed any time reactor containment is required, except when a workman is passing through the access.

When containment is not required, the doors are not required to seal or to retain their integrity. Since the doors are left open only when containment is not required, seismic qualifications of the doors in the open position is not required.

Earthquake loads used in designing the doors were from accelerations determined for the crane wall at the horizontal centerline of each door by dynamic analysis of the Reactor Building for an OBE and an SSE. These acceleration loads were used as static loads since the doors are firmly secured to the wall when closed. Doors were reanalyzed using Set "B" ARS spectrum.

Some air leakage may occur at the periphery of the doors during earthquakes, but this leakage will not exceed the permissible leakage of 30 square inches per door.

Under normal conditions, seals on the doors will have a life of not less than 10 years, and the other parts of the doors will have a life of not less than 40 years. Some air leakage may occur at the periphery of the doors, but this leakage will not exceed the permissible leakage area for normal operation of 10 square inches per door.

Wall Above the Operating Deck (Divider Barrier)

Primary Design

Under accident conditions the crane wall above the operating deck is designed for maximum pressure differentials between the ice compartments outside the crane wall and the steam generator and pressurizer compartments. It is also designed for the loads imposed on the wall by the lattice frame anchorages of the ice condenser. Maximum and minimum vertical loads imposed by the earthquake analysis are combined with these loads for the maximum stress conditions. The end walls of the ice condenser and the spacing of the steam generator and pressurizer walls stiffen the crane wall to such an extent that it essentially spans horizontally between these supporting walls. The stud loadings on the wall from the lattice frame may either add to or subtract from the pressure loading depending on whether the maximum pressure is inside the crane wall or in the ice compartment. The portion of the crane wall behind the steam generator and pressurizer compartments is subject to jet impingement loads.

The steam generator and pressurizer upper lateral restraints are anchored in the crane wall and exert large forces on it. The effect of pressure on the top slab of the pressurizer and steam generator compartments causing an uplift on the crane wall was considered.

The STRUDL II frame program, STRUDL II finite element program, and the SAP IV (1973) finite element program are the principal computer programs used in the analysis of the upper crane wall above the operating deck slab.

The upper restraint of the steam generators is designed such that the crane wall only receives load from a steam line break. Seismic restraints in the radial direction are transmitted through hydraulic snubbers to the floor of the operating deck. In the other direction they are transmitted to the walls of the steam generator compartment. The two 720-kip steam generator restraint loads on the crane wall are assumed to occur coincidentally with the maximum pressure differential in the steam generator compartment.

During construction, a 36-foot-wide opening, used for moving major equipment into the building, was left in the crane wall at approximately the 90° azimuth. This opening began at Elevation 756.63 and extended 46 feet high. This leaves a 3-foot-wide, 17-foot-deep curved beam spanning a 37½-foot arc over the opening. This beam and the permanent beam over the refueling canal are designed to carry the construction loads of the polar crane, approximately 1200 kips maximum while installing major equipment. The permanent beam is also designed to take the reactions from the cantilevered beams supporting the ice condenser bridge crane. The analysis of these beams was made using the STRUDL computer program. The top of the crane wall is designed to withstand a force in the radial direction due to the polar crane bumping into it as a result of seismic action. This seismic force from the polar crane is considered to act at any point on the circumference of the wall and approximately 2 feet below the top of the wall.

<u>Independent Design - *Historical Information*</u>

The crane wall resists two general types of loads. They are, 1) localized forces from equipment supports, pressure forces, structure discontinuities, etc., and 2) forces from gross structure motions of the interior concrete structure induced by the design earthquakes and the design basis accident (DBA). Calculations of gross forces in the interior concrete structure due to a design earthquake are described in Section 3.7.2.1.1. The lumped mass cantilever beam model used in the seismic analyses was loaded by a time-dependent forcing function representing the non-axisymmetric pressure loads from a DBA. The analysis used modal superposition and determined the total responses in the time domain. The results of the analysis consisted of gross overturning moments and shears and accelerations, deflections, and acceleration response spectra at various elevations.

Portions of the crane wall subjected to very isolated forces were isolated and designed as substructures. Boundary conditions were always chosen to give conservative results.

The forces at the junction of crane wall and other divider barrier components such as the operating deck, ice condenser floor, and end walls were included in the design.

The crane wall at the ice condenser inlet doors between Elevations 746.42 to 753.63 were designed as beam columns. The columns are subjected to both localized and gross motion forces. Localized loads resulted from the steam generator support loads, pressure forces, and interactions from the operating deck and the ice condenser floor. The column design was verified by both working stress and ultimate strength methods.

The portion of the crane wall within the steam generator compartment was analyzed as part of the steam generator enclosure. This portion was modeled using the MARC-CDC nonlinear finite element computer program. The element utilized was the 20-node isoparametric solid.

The loadings on the wall consisted of reaction loads from the steam generator supports near the crane wall columns and pressure loads from postulated breaks in the main steam line. The crane wall columns were designed to resist maximum moments, shears, and axial loads due to the local forces on the crane wall as well as forces due to gross motions of the interior concrete. The crane wall segment of the steam generator compartment model was extended past the juncture of the enclosure and the crane wall to minimize boundary effects on the solution. Boundary conditions assumptions were selected to provide conservative stresses. The crane wall within the pressurizer compartment was designed by a similar method.

3.8.3.4.8 Steam Generator Compartments (Divider Barrier)

These compartments are designed to resist a maximum of 38 psi differential pressure on the wall common with the upper compartment and 26 psi differential pressure on the center wall that would result following a main steam pipe break inside any single compartment. The center wall is also designed for the effect of a 1160-kip jet force that would result from a main steam pipe break. Also accounted for are thermal effects accompanying a pipe break (see Figure 3.8.3-19).

The compartments span mainly in the horizontal direction resulting in tensile stresses and horizontal moments in the walls near the center of their height. Close to the ends of the compartments, discontinuity stresses result in the vertical direction, similar to those of a flat head cylinder.

The STRUDL frame program was used to find the maximum horizontal forces in the walls by modeling a vertical 1-foot height of walls including a 113-degree sector of crane wall. Short chord lengths were used to represent curved sections of walls. Manual calculations were done at the top and bottom of the wall which is common to the upper compartment to investigate the effects of the slabs restraining the wall. In addition, the flat plate finite element STRUDL computer program was used to analyze the center wall for moments and shears in both directions. The top slab was analyzed using stiffened members in the flat plate finite element STRUDL program. The inverted "tee"-shaped beam which stiffens the top slab and which is located at the top of the center wall was analyzed for the dynamic effects of a main steam pipe breaking and striking the flange of the beam.

Reanalysis Due to Unit 1 Steam Generator Replacement

The modified configuration of the Unit 1 steam generator compartment was analyzed for design loads using a 3D finite element ANSYS model. Although the roof slab remains the focus of the evaluation, the model included five components - the 3 feet thick roof slab, entire steam generator compartment wall, center wall, 180° sector of the crane wall and the whip-restraint beam; to obtain an accurate representation of the system. The material properties used in the model for the concrete were consistent with those used in the original analysis. The concrete strength used in the roof evaluation is the in-place compressive strength of the steam generator compartment roof concrete at 90 days.

The modified steam generator compartment roof was analyzed for the following design loads: dead load, live load, design pressure differential from a DBA (main steam pipe break), operating and accident temperature effects, seismic effects (OBE and SSE), and pipe thrust load on the whip-restraint beam from a broken main steam pipe. The modified steam generator compartment roof was evaluated to the load combinations and allowable stresses tabulated in Table 3.8.3-1.

Independent Design - *Historical Information*

1. Roof Slab

   The steam generator enclosure roof slab was analyzed as a thick plate using the three-dimensional 20-node isoparametric solid element available in the MARC-CDC finite element program. The T-beam stiffener attached to the inside of the slab was included in this finite element model.

   The MARC-CDC finite element program is based on research work carried out by Professor Pedro V. Marcal of Brown University and colleagues at the University of London. In 1969 the program was released commercially by the Marc Analysis Research Corporation. The Control Data Corporation has recently documented the program and offered it for general usage through their computer system. A sample problem was run to verify the results obtained from the program.

   The boundary conditions of the roof slab were approximated by modeling the actual stiffness of the steam generator enclosure wall and crane wall.

   The design loads considered originated from a postulated rupture in the main steam line in a steam generator compartment. Differential pressures, jet force and pipe reaction resulting from the postulated break were checked for factored and unfactored load cases. The T-beam beneath the roof slab was designed to resist pipe whip forces from a postulated guillotine break in the main steam line. Moments, shears, and torsional stresses were checked to ensure adequate reinforcing utilizing the stress criteria of Table 3.8.3-1.

2. Enclosure Walls and Separation Wall

   The steam generator enclosure wall, separator wall, and a segment of the crane wall were modeled utilizing the three-dimensional 20-node isoparametric element available in the MARC-CDC nonlinear finite element program.

   Spring constants simulating the crane wall columns were obtained by analyzing a small portion of the crane wall. The walls of the steam generator enclosure were then modeled to include appropriate boundary conditions at the intersection of the operating deck, upper crane wall, and the crane wall columns.

   The loadings considered were of two basic types: a) a pressure load obtained from a hypothetical DBA, and b) snubber and embedment plate loads caused by combined LOCA and seismic action. Live loads, dead loads, seismic loads, and thermal loads were also considered.

   Flexural, axial, shear, and torsional stress levels resulting from the factored and unfactored loading combinations were evaluated using the allowable stress criteria of Table 3.8.3-1. Reinforcement was selected based on these criteria.

3.8.3.4.9 Pressurizer Compartment (Divider Barrier)

Primary Design

The compartment is designed to resist a 50 psi differential pressure. Methods of analysis were similar to those of the steam generator compartments.

Independent Design - *Historical Information*

The enclosure wall and the adjacent crane wall were modeled using solid isoparametric finite elements. The computer program utilized was the previously documented SAP IV. The crane wall model was extended beyond its juncture with the enclosure wall to minimize boundary effects. Boundary conditions at the intersection of the enclosure wall with adjacent elements of the interior concrete structure were chosen to provide conservative stress results.

The roof slab was modeled as a thick plate using the ANSYS computer program, a widely used and previously documented program. Boundary conditions at the junction with the crane and enclosure walls were chosen to provide maximum stress levels.

With these stress levels, a detailed analysis to determine unit stresses in the concrete and reinforcing steel was based on a cracked section and evaluated using the allowable stresses given in Table 3.8-3-1.

3.8.3.4.10 Operating Deck at Elevation 756.63 (Divider Barrier) Primary Design

The floor is designed to carry a 24 psi upward pressure and thermal effects due to a LOCA plus the jet pressure of 340 psi acting over a local area. Upward loads from the missile shield are taken by this floor around the reactor cavity where the shield is bolted down.

This floor is designed for a 1,000-psf live load. This loading suffices for several concentrated loads from the reactor head set down which occur during refueling and periodic maintenance of the equipment.

During construction, the floor has no edge support at the steam generator and pressurizer compartment walls, since the first lifts of these walls are carried by the floor. This special construction condition was examined separately using a 300-psf design live load in addition to the wet weight of the first 45-foot pour of walls.

The floor analyses were made using the STRUDL finite element program for flat plates utilizing both rectangular and triangular elements to assemble the irregular shape.

Anchorage for the upper steam generator restraints is provided in this floor. These anchorage points have approximately a 5,500 kip design force due to a LOCA combined with SSE. This force is horizontal and applied at points where openings create horizontal single span beams. These beams were analyzed using manual methods.

Independent Design - *Historical Information*

The operating deck was analyzed using four finite element models to represent this irregularly shaped slab. Two models were used to analyze the larger segment from 0° to 180° while two additional models were used to analyze the segments adjacent to the refueling canal walls. Loads normal to the surface and inplane loads act on the operating deck. Stresses from concentrated snubber (inplane) loads were determined from a plane stress finite element analysis. These snubbers act as the lateral supports for the steam generators. In addition to the snubber loads, pressure loads and jet forces from postulated pipe ruptures acting normal to the operating deck were determined from a plate bending finite element model. Dead, live, seismic, pipe support, and construction loads were added to the differential pressure and snubber forces to complete the factored and unfactored load cases.

The SAP IV plate bending and plane stress elements were used to calculate moments, shears, reactions, and inplane stresses. Boundary conditions at the crane wall, steam generator enclosure, pressurizer enclosure, reactor cavity, and refueling canal wall junctions were chosen to give the most conservative results. Using the calculated moment, shears, and axial forces, stresses in the reinforcement for an assumed cracked section were checked against the stress criteria of Table 3.8.3-1.

3.8.3.4.11 Ice Condenser Support Floor - Elevation 744.5 (Divider Barrier)

Primary Design

The finite element program SAP IV (1973) was primarily used in the analysis and design of the Elevation 744.5 floor. The outer circumferential beam was represented along with the floor by using a combined flat plate and grid member system. The supporting columns were modeled using spring constants for both rotation and deflection. Shear and moment values were obtained from the computer program at the crane wall, ice condenser end walls, and supporting columns. Reinforcing selections were made from these results.

Independent Design - *Historical Information*

The ice condenser floor was analyzed as a series of circumferentially beam-stiffened curved slabs with a continuous support along the inner radius and ends. These slabs were supported along the outer radius by flexible columns. The analysis was made utilizing SAP IV, a previously documented computer program. Models were generated for segments of the ice condenser floor at 0°, 90°, 44°, 180°, and 230°. Boundary conditions between the segments were input to provide maximum forces at the boundary and midspan.

The various segments of the ice condenser floor were dynamically analyzed to determine natural frequencies. Next, time varying differential pressure loads were used to calculate dynamic load factors associated with the natural frequencies of each segment. The dynamic load factors obtained were then applied to the maximum differential pressures in addition to the factors for the ice condenser structure support loads.

The loadings evaluated included concentrated forces and moments from the ice condenser support system, jet forces resulting from change in momentum of the steam flow, and LOCA induced pressure differentials. The ice condenser support system loads resulted from a combination of seismic accelerations and drag on the ice baskets by the channeled gases. Normal and factored load cases were examined to determine maximum moments and shears. Stresses in the concrete and reinforcement were evaluated against the criteria of Table 3.8.3-1.

3.8.3.4.12 Ice Condenser

Analysis, meeting the criteria presented in Section 3.8.3.5, has been done on the basis of elastic system and component analyses. Limit load analysis was used as an alternate to the elastic analysis. Limit loads are defined using limit analysis by calculating the lower bound of the collapse load of the structure. Load factors are applied to the defined design basis loads and compared to the limit loads. The load factors determined for design basis load are used to provide margins of safety of the structure against collapse. A load factor of 1.43 was used when considering the mechanical loads due to dead weight and OBE. A load factor of 1.3 was used for (D + SSE) and (D + DBA). The material was assumed to behave in an elastic-perfectly-plastic manner where strain-hardening effects are neglected. The minimum specified yield strength was used. Mechanical plus thermal induced load combination and fatigue was analyzed in an elastic basis and satisfy the limits of Section 3.8.3.5. The stress analyses and results are described in Sections 3.7 and 6.7.

Experimental or Test Verification of Design

In lieu of analysis, experimental verification of design using actual or simulated load conditions was used.

In testing, account was taken of size effect and dimensional tolerances (similitude relationships) which exist between the actual component and the test models, to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading. The load factors associated with such verification are: 1.87 for D + SSE, 1.43 for D + DBA or D + SSE, and 1.3 for (D + SSE) or 1.3 for (D + DBA).

A single test sample is permitted but in such cases test results were derated by 10%. Otherwise, at least three samples were tested and the design was based on the minimum load carrying capability.

Additional analysis results are found in Section 6.7.

3.8.3.4.13 Penetrations Through the Divider Barrier

Canal Gate

Primary Design

The canal gate sections are designed to span as simply supported beams across the refueling canal; a clear span of some 19 feet. Hand calculations using conventional methods were used for this design. The canal gate was designed to withstand a 39-psi pressure differential between the compartment above the reactor and the upper compartment due to a LOCA. The effect of seismic action on the canal gate sections is considered as well as the effect of maximum temperature differential across the gate.

Independent Design - *Historical Information*

The canal gate was designed as a simply supported plate spanning between the two refueling canal walls. The canal gate is required to maintain integrity between the upper and lower compartments during a LOCA and was designed for the maximum probable differential pressure. The effects of seismic and thermal action were evaluated.

Moments and shears were calculated using conventional hand methods. The evaluation of concrete and reinforcing steel stresses was based on a cracked section and the allowable stresses of Table 3.8.3-1.

Control Rod Drive (CRD) Missile Shield

Primary Design

The CRD missile shield sections are designed to span as simply supported slabs across the compartment above the reactor. The slabs are held down at the ends by anchor bolts embedded in the operating deck slab. The missile shield is designed to withstand a maximum differential pressure of 39 psi between the compartment above the reactor and the upper compartment due to a LOCA. The missile shield is subject to loading from the CRD mechanism as a missile. An accompanying jet force due to pressure escaping through the head of the reactor is also considered. The slabs are investigated for the maximum penetration resulting from the missile effects of the control rod drive shaft. The underside of the slab is faced with a 1-inch-thick steel plate to aid in resisting missile penetration. The penetration depths are calculated by use of the Petry formula and a formula by C. V. Moore, "The Design of Barricades for Hazardous Pressure Systems," Nuclear Engineering and Design 5 (1967), 81-97, North-Holland Publishing Company, Amsterdam. The calculated penetration depth is 2.2 inches into the 3-foot, 6-inch thick slab. The effect of maximum temperature differential across the missile shield is also considered in the design.

Independent Design - *Historical Information*

The missile shield sections were analyzed as simply supported slabs spanning the compartment above the reactor. This shield must resist the maximum probable differential pressure from a LOCA to maintain integrity between the lower and upper compartments. Additionally, it must resist certain missiles from the control rod drive mechanism. Penetration into the steel plate and concrete were calculated and equivalent static loads for the impacting missiles were calculated and evaluated. Thermal stresses resulting from temperature differentials between lower and upper volumes were considered in the stress evaluation.

Concrete and reinforcing steel stresses were determined considering a cracked section and the stress allowables of Table 3.8-3-1.

Reactor Coolant Pump Access and Lower Compartment Access Hatches

Primary Design

The reactor coolant pump access and lower compartment access slabs are designed to span simply supported between anchor bolt. The slabs are designed for both downward and upward loads acting on them. The downward loads are dead load and a 1,000-psf live load. For upward loads, the slabs are designed to carry a 24-psi differential pressure between the lower and upper compartments due to a LOCA. A jet impingement loading associated with this LOCA is also considered. The effect of maximum temperature differential as well as seismic effects on the slabs are accounted for in the design.

Independent Design - *Historical Information*

The reactor coolant pump access hatch and the lower compartment equipment access hatch were analyzed and designed as simply supported circular and rectangular plates. Maximum moments and shear forces were obtained from a plate bending analysis. Dead, live, seismic, and thermal loads were combined with differential pressures and jet forces due to a postulated LOCA to give controlling factored and unfactored load cases. Shear stresses at the periphery of the hatch openings in the operating deck and stress levels in the perimeter anchor bolts were checked to ensure compliance with criteria of Table 3.8.3-1.

Equipment Access Hatch

The hatch cover is designed to span as a simply supported beam between the anchor bolt assemblies with the anchor bolts designed to withstand a load at least 5% greater than that calculated for the end reactions resulting from the actual load on the hatch. The allowable stresses are given in Table 3.8.3-6.

Escape Hatch

Structural components of the hatch have been designed such that, the allowable stresses given in Table 3.8.3-7 will not be exceeded.

Air Return Duct Penetrations

The controlling design condition is Design Basis LOCA Pressure - Dead Load + Safe Shutdown Earthquake Loads. Maximum differential temperature is considered in the design, but does not occur coincidentally with a jet force or the maximum differential Design Basis LOCA Pressure. Calculated and allowable stresses are given in Table 3.8.3-8.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 General

Structure

Calculated and allowable stresses for the principal concrete structure are listed in Table 3.8.3-2. Locations of these areas are shown referenced to Figures 3.8.1-1, 3.8.3-6 and 3.8.3-7. The working stress design method was used for finding stresses.

3.8.3.5.2 Structural Fill Slab on Containment Floor

During the original design (construction permit) phase with 40% added to LOCA pressure, the controlling combination was "Abnormal/Severe Environmental." See Table 3.8.3-2.

Shear transfer through the fill slab is discussed in Section 3.8.3.1.2.

3.8.3.5.3 Reactor Cavity Wall and Compartment Above Reactor

Loading combinations 1 through 7 in Table 3.8.3-1 were considered in the design. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combinations are "Abnormal" and "Abnormal/Severe Environmental." See Table 3.8.3-2. The 4-foot, 3-inch structural thickness provided adequate depth for limiting peripheral shear stresses due to anchorage loads on the steam generator and reactor coolant pump restraints.

Earthquake shears for the interior structures were distributed to the various walls in proportion to their rigidity. This term is defined as $1/\Delta$ where delta is the deflection due to a unit force, and it includes deflection due to bending and shear. The procedure is outlined in "Analysis of Small Reinforced Concrete Buildings" by the Portland Cement Association.

The columns at the top of the compartment above the reactor wall were designed for the effect of earthquake shears by proportioning shear to them based upon their rigidity as previously described.

3.8.3.5.4 Refueling Canal Walls and Floor

Loading combinations 1 through 7 in Table 3.8.3-1 were examined. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combinations are "Abnormal" and "Abnormal/Severe Environmental." See Table 3.8.3-2.

3.8.3.5.5 Crane Wall

The crane wall was analyzed for loading combinations 1 through 7 in Table 3.8.3-1. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal/Severe Environmental." See Table 3.8.3-2.

Earthquake shears were calculated utilizing the procedure discussed in Section 3.8.3.5.3.

Shear reinforcement was required in many areas of the crane wall for radial shears generated by jet impingement loading, as well as pipe and equipment restraint reactions, missile impingement loading and pressure due to LOCA.

3.8.3.5.6 Steam Generator and Pressurizer Compartment

Loading combinations 1 through 7 in Table 3.8.3-1 were examined. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal" for the pressurizer compartment and "Abnormal/Severe Environmental" and "Abnormal/ Extreme Environmental" for the steam generator compartment. See Table 3.8.3-2.

3.8.3.5.7 Operating Deck at Elevation 756.63

Loading combinations 1 through 7 in Table 3.8.3-1 were examined for the floor. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal/Severe Environmental." See Table 3.8.3-2.

3.8.3.5.8 Ice Condenser Support Floor - Elevation 744.5

Loading combinations 1 through 7 in Table 3.8.3-1 were examined in the design of the floor. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal." See Table 3.8.3-2.

3.8.3.5.9 Penetrations Through the Divider Barrier

Canal Gate and Control Rod Drive (CRD) Missile Shield

Loading combinations 1 through 7 in Table 3.8.3-1 were examined. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal/Severe Environmental." See Table 3.8.3-2.

Reactor Coolant Pump and Lower Compartment Access Hatches

Loading combinations 1 through 7 in Table 3.8.3-1 were examined. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal/Severe Environmental."

Escape Hatch

Loading combinations 1 through 7 in Table 3.8.3-1 were examined. During the original design (construction permit) phase with calculated values of LOCA pressure load increased by 40%, the controlling load combination is "Abnormal/Severe Environmental."

3.8.3.5.10 Personnel Access Doors in Crane Wall

Allowable stresses for non-collapsible members for load combinations used for the various parts are given in Table 3.8.3-3. Normal load conditions are shown for mechanical members only. Loads on structural members during normal conditions are negligible and therefore are not shown on Table 3.8.3-3. For normal load conditions, factors of safety for mechanical parts are 5 to 1 on ultimate. For limiting conditions such as SSE and a pipe rupture accident, stresses do not exceed 0.9 yield.

For collapsible members during a pipe rupture accident, stresses exceed yield and members are plastically deformed. Plastic deformation of energy absorbing members does not affect the sealing integrity of the doors.

3.8.3.5.11 Seals Between Upper and Lower Compartments

Under normal and earthquake conditions, there are no loads on the seals. However, the seals are subject to radiation, as outlined previously, during normal operating conditions. The seal has been tested under accident pressures and temperatures after undergoing heat aging to 40 years equivalent age, and irradiation to 40 years normal operation plus accident integrated doses in order to qualify it for the life of the plant.

The seals are not required to maintain their integrity during a fire. It is assumed that a fire and an accident which require sealing will not occur simultaneously since the reactor will be shut down immediately if a fire develops.

3.8.3.5.12 Ice Condenser

Table 3.8.3-4 provides a summary of the allowable limits to be used in the design of the ice condenser components.

For all cases the stress analysis was performed by considering the load combinations producing the largest possible stress values.

Stress Criteria

The stress limits for elastic analysis are:

1. D + OBE

Stress shall be limited to normal AISC, Part I Specification allowables (S). The members and their connections shall be designed to satisfy the requirements of Part I, Sections 1.5, 1.6, 1.7, 1.8, 1.9, 1.10, 1.15, 1.16, 1.17, 1.20, 1.21 and 1.22 of the AISC Specification (stress increase in Sections 1.5 and 1.6 is disallowed for these loads). Where the requirements of Section 1.20 are not met, differential thermal expansion stresses shall be evaluated and the maximum range of the sum of mechanical and thermal induced stresses shall be limited to three times the appropriate allowable stresses provided in Section 1.5 and 1.6 of AISC Specification.

2. D + SSE, D + DBA

Stresses shall be limited to normal AISC Specification allowables given in Sections 1.5 and 1.6, increased by 33% (1.33S). No evaluation of thermal induced stresses or fatigue is required. In a few areas, where the stresses exceed 1.33S but are below l.5S, specific justification is provided on a case by case basis.

3. D + SSE + DBA

Stresses shall be limited to normal AISC Specification allowable given in Sections 1.5 and 1.6, increased by 65% (1.65S). No evaluation of thermal induced stresses or fatigue is required.

For all cases, direct (membrane) mechanical stresses shall not exceed $0.7S_u$, where S_u is the ultimate tensile strength of the material.

The summary of the ice condenser allowable limits is given in Table 3.8.3-4.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

General

Refer to Section 3.8.1.6.

3.8.3.6.1 Materials

Refer to Section 3.8.1.6.1 with the following additions.

Concrete

Aggregates for radiation-shielding concrete which was used in limited locations conformed to ASTM C 637-73.

The specified strengths of concrete used for interior concrete structures were 3000 psi, 4000 psi, 5000 psi, and 8000 psi.

Reinforcing Steel

Prestress steel which was used in the reactor cavity walls conformed to ASTM A 421-65.

Personnel Access Doors in Crane Wall

ASTM standards were used for all material specifications and certified mill test reports were provided by the contractor for materials used for all load carrying members.

Seals Between Upper and Lower Compartments

The seals consist of long strips of flexible elastomer coated fabric with both edges hemmed to form pockets into which metal clamp bars are inserted. The coated fabric is two ply dacron coated on both sides with an elastomer (ethylene-propylene-dienepolymer). The elastomer is compound E603 or E603-A by the Presray Company.

Escape Hatches in Elevation 756.63 Floor

ASTM standards were used for all material specifications and certified mill test reports were provided by the contractor for materials used for all load carrying members.

3.8.3.6.2 Quality Control - Historical Information

Concrete

The quality control requirements were essentially the same as in Section 3.8.1.6.2. Some concrete did not meet specification requirements. This was evaluated and documented in Reference [2]. Results have been documented in affected calculation packages and drawings.

Personnel Access Doors in Crane Wall, Escape Hatches in Elevation 756.63 Floor

Design by TVA and erection by TVA were in accordance with TVA's quality assurance program. Design and fabrication by the contractor were in accordance with the contractor's quality assurance program which was reviewed and approved by TVA's design engineers. The contractor's quality assurance program covers the criteria in Appendix B of 10 CFR 50. Fabrication procedures such as welding and nondestructive testing were included in Appendices to the contractor's quality assurance program.

ASTM standards were used for the material specifications and certified mill test reports were provided by the contractor for materials used for the load carrying members.

Seals Between Upper and Lower Compartments

The flexible elastomer coated fabric used for seals was certified by a qualified rubber technologist as being adequate for the normal and accident conditions. In addition, certified mill test reports were provided by the contractor for materials used for the load carrying members.

The seal has been tested by the original seal supplier under contract with TVA. The test was designed to evaluate seal specimens under simulated accident temperature and pressure conditions in a configuration emulating actual plant as-constructed installation. The test specimens, which were fabricated from seal material removed from the Unit 1 containment,

were heat aged to 40 years equivalent age, and irradiated to 40 years normal operation plus accident integrated doses prior to testing. This testing process represented the material properties that would exist following a design basis accident at the end of a 40 year plant life.

3.8.3.6.3 Construction Technique - *Historical Information*

No unusual construction procedures were employed in the construction of the interior structures.

3.8.3.6.4 Ice Condenser

Structural steels for ice condenser components are selected from the various steels listed in the AISC Specification or Code. When materials such as steel sheets, stainless steel or nonferrous metals are required and are not obtainable in the AISC Code, these materials are chosen from ASTM Specifications. Proprietary materials such as insulating materials, gaskets and adhesives are listed with the manufacturers' name on the component drawings.

Material certifications for chemical analysis and tensile properties were required with testing procedure and acceptance standards meeting the AISC or ASTM requirements.

Because the concept of non-ductile fracture of ferritic steel is not a part of the AISC Code, and Westinghouse recognizes its importance in certain ice condenser components where heavy plates and structurals are used, such as the lower support structure, Charpy V-notch (CVN) energy absorption requirements are stipulated as shown in Table 3.8.3-5.

These criteria apply to the design of the following ice condenser components:

1. Ice basket and coupling.

2. Lattice frame and columns including attachments and bolts.

3. Structural steal supporting structures comprising the lower support structure, door frames and bolts.

4. Wall panels and cooling duct support studs attached to the crane wall and walls.

5. The supports of auxiliary components which are located within the ice condenser cavity but which have no safety function.

The various candidate materials, i.e., steel sheets, structural shapes, plates and bolting used in the ice condenser system were selected on the following bases:

1. Provide satisfactory service performance under design loading and environment and pressure or construction performance.

2. Assure adequate fracture toughness characteristics at ice condenser design conditions.

3. Be readily fabricated, welded, and erected.

4. Be readily coated for corrosion resistance when required.

The candidate materials are of high quality and were made by steel-making practices to be specified by Westinghouse. Principal candidate materials meeting the above bases are listed below. Other materials for specific applications are selected on a case-by-case basis.

Sheets

Carbon steel sheets are commercial quality (CQ), drawing quality (DQ), or drawing quality-special killed (DQ-SK). The selection of the quality depends upon the part being formed. When higher strength, structural quality sheets are required, ASTM specification A607 is used. AISI Type 409 modified stainless steel is a potential alternate sheet material for the ice baskets.

The ice baskets were made from perforated sheet material. The wall duct panels were made from sheet material and the cradle supports from structural sections and plates.

Structural Sections, Plates and Bar Flats

Structural sections, plates and bar flats are generally high-strength, low alloy steel selected for suitable strength, toughness, formability and weldability.

The high-strength low-alloy steels are A441, A588, A572 or A633. These steels are readily oxygen cut and possess good weldability.

Bolting

High-strength alloy steel Type A320 L7 bolting for low temperature service is used for the lower support structure. Stocked bolting made from A325, A449 and ASTM A354, Grade BD (SAE J429, Grade 8) materials are used for other parts. The above bolts met CVN 20 ft-lb at -20°F, for sizes greater than 1 inch in diameter.

Nonmetallic materials such as gaskets, insulation, adhesives and spacers are selected for specific uses. Freedom from detrimental radiation effects is required.

All structural welding was in accordance with the AWS Structural Code for Welding, D1.1 (AWS Code). The AWS Code is an overall welding system for the design of welded connections, technique, workmanship, qualification and inspection for buildings, bridges, and tubular structures. (See Section 3.8.3.2, Item 5).

The quality of welds for the ice condenser system is based on Paragraph 9.25 of the AWS Code. (See Section 3.8.3.2, Item 5).

Resistance welding was in accordance with AWS, Recommended Practices for Resistance Welding, C1.1.

Magnetic particle examination was performed on at least 5% of the welds in each critical member of the lower support structure. Magnetic particle or liquid penetrant examinations, where applicable, were performed on 5% of the welds in each critical member of the balance of the ice condenser structure. The welds selected for examination were designated in the Design Specifications. The nondestructive examination methods and acceptance standards are given in Section 6 and Paragraph 9.25, Quality of Welds, of the AWS Code. (See Section 3.8.3.2, Item 5).

3.8.3.7 Testing and Inservice Surveillance Requirements

Testing of the interior concrete structures was not planned. A completely independent design has been prepared for divider barrier features in order to ensure that during a LOCA the escaping steam will not bypass the ice condenser.

Personnel Access Doors in Crane Wall

Periodic visual inspections of the doors are to be made. Parts inspected during the visual inspection are to include all bolted connections, structural members for paint deterioration, latches, hinges, and elastomer seals. The seals are to be inspected for cracks, blemishes, or any other indications of deterioration of the elastomer and for proper seating at the sealing surfaces.

Seals Between Upper and Lower Compartments

On periodic unit shutdowns, visual inspections of the seals are to be made. Parts inspected are to include all bolted connections, clamp bars, metal to fabric joints, and the elastomer-coated fabric. The seals are to be replaced if they show any evidence of deterioration.

Escape Hatches in Elevation 756.63 Floor

Periodic visual inspections of the hatch covers are to be made. Parts inspected during the visual inspection are to include all bolted connections, structural members for paint deterioration, latching mechanisms, hinges, limit switches, and elastomer seals.

The escape latch seals are to be carefully inspected for cracks, blemishes, or any other indications of deterioration of the elastomer and for properly seating at the sealing surfaces.

3.8.3.8 Environmental Effects

The atmosphere in the ice bed environment is at 10°F and the absolute humidity is very low. Therefore, corrosion of uncoated carbon steel is negligible.

To ensure that corrosion is minimized while the components of the ice condenser are in storage at the site or in operation in the containment, components are galvanized, painted, protective coatings installed, or placed in a protective container. Galvanizing is in accordance with ASTM A123 or A386.

Materials such as stainless steels with low corrosion rates shall be used without protective coatings.

Corrosion has been considered in the detailed design of the ice condenser components, and it has been determined that the performance characteristics of the ice condenser materials of construction are not impaired by long-term exposure to the ice condenser environment.

Since metal corrosion rates are directly proportional to temperature and humidity, corrosion of ice condenser components at operating temperatures has been assumed to be almost nonexistent. Data available in the open literature does not reflect the exact temperature range and chemistry conditions that are expected to exist in the ice condenser, but does indicate that corrosion rates decreased with decreasing temperatures for the materials and conditions being considered. Although the data in the literature indicated that corrosion of components is not expected, Westinghouse has chosen to employ several preventive measures in the construction of the ice condenser system. To inhibit corrosion, galvanizing is used on the ice baskets. Westinghouse has performed tests which show that galvanized material would not be expected to fail due to corrosion during a 40-year exposure to a 5-15°F ice condenser refrigerated air environment. Other structural members are galvanized, protected by corrosion resistant paints that meet the requirements of ANSI 101.2-1972 (Protective Coatings [Paints] for Light Water Nuclear Reactor Containment Facilities) as a minimum, or were constructed of stainless steel, or self-passivating steel. Heavy plate and structural fabrications may be installed in the blasted and/or bare condition.

REFERENCES

1. TVA Nuclear Quality Assurance Plan, TVA-NQA-PLN89-A.

2. Concrete Quality Evaluation, CEB-86-19-C.

TABLE 3.8.3-1

LOADING COMBINATIONS, LOAD FACTORS AND ALLOWABLE STRESSES FOR INTERIOR CONCRETE STRUCTURE

	LOADING COMBINATION	LOAD												ALLOWABLE STRESSES		
		D	L(1)	T_0	T_a	P_a	E	E'	R_0	R_a	Y_j	Y_r	Y_m	fs	fc	V_c (4)
1.	NORMAL	1.0	1.0	1.0			1.0		1.0					0.5fy(2)	0.45fc'(3)	$.5(3.5\sqrt{f_c'})$
2.	EQUIVALENT TEST	1.0	1.0			1.0								0.67 fy	0.45 fc'	$.5(3.5\sqrt{f_c'})$
3.	EXTREME ENVIRONMENTAL	1.0	1.0	1.0				1.0	1.0					0.9 fy	0.75 fc'	$3.5\sqrt{f_c'}$
4.	ABNORMAL	1.0	1.0		1.0	1.5				1.0				0.9 fy	0.75 fc'	$3.5\sqrt{f_c'}$
5.	ABNORMAL	1.0	1.0		1.0	1.25				1.0	1.0	1.0	1.0	0.9 fy	0.75 fc'	$3.5\sqrt{f_c'}$
6.	ABNORMAL/ SEVERE ENVIRONMENTAL	1.0	1.0		1.0	1.25	1.25			1.0	1.0	1.0	1.0	0.9 fy	0.75 fc'	$3.5\sqrt{f_c'}$
7.	ABNORMAL/ EXTREME ENVIRONMENTAL	1.0	1.0		1.0	1.0		1.0		1.0	1.0	1.0	1.0	0.9 fy	0.75 fc'	$3.5\sqrt{f_c'}$

(1) INCLUDES ALL TEMPORARY CONSTRUCTION LOADING
(2) fy = SPECIFIED YIELD STRENGTH OF REINFORCEMENT
(3) fc' = SPECIFIED COMPRESSIVE STRENGTH OF CONCRETE
(4) V_c = THIS IS MAXIMUM ALLOWABLE SHEAR STRESS, CARRIED BY THE CONCRETE, WHICH MAY BE REDUCED DEPENDING ON THE SECTION AND TYPE OF LOADING, REF. ACI 359 AS ISSUED FOR TRIAL USE APRIL 1973.

LOAD NOMENCLATURE:

D - DEAD LOADS, OR THEIR RELATED INTERNAL MOMENTS AND FORCES
L - LIVE LOADS, OR THEIR RELATED INTERNAL MOMENTS AND FORCES
T_0 - OPERATIONAL TEMPERATURE LOADS
T_a - ACCIDENT TEMPERATURE LOADS
P_a - ACCIDENT MAXIMUM DIFFERENTIAL PRESSURE
E - OPERATIONAL BASIS EARTHQUAKE
E' - SAFE SHUTDOWN EARTHQUAKE
R_0 - PIPE REACTION DURING OPERATING CONDITIONS
R_a - PIPE REACTION DUE TO INCREASED TEMPERATURE RESULTING FROM THE DESIGN ACCIDENT
Y_j - JET IMPINGEMENT DUE TO FLUID DISCHARGE FROM BROKEN PIPE
Y_r - PIPE REACTION DUE TO FLUID DISCHARGE FROM BROKEN PIPE
Y_m - MISSILE IMPINGEMENT LOAD

TABLE 3.8.3-2

SUMMARY OF REPRESENTATIVE MAXIMUM STRESSES FOR THE INTERIOR CONCRETE STRUCTURE[1,5]

DESIGN FEATURE		FLEXURE[2,3]						SHEAR[4]		
FOR LOCATION SEE FIGURES 3.8.3-1 THRU 7	LOAD COMB.[6]	fc_{ACTL}	$fs_{\cancel{ACTL}}$	fs'_{ACTL}	fc_{ALL}	fs_{ALL}	fs'_{ALL}	V_{ACTL}	V_{ALL}	V_c
PRESSURIZER COMPT AT CRANE WALL	4	3.5	49.3	9.1	3.75	54	54	148	168	106
ST GEN COMPT, SIDEWALL AT CRANE WALL	6	2.4	49.2	24.0	3.75	54	54	254	290	106
ST GEN COMPT, CENTER WALL AT CRANE WALL	7	2.3	52.6	15.1	3.75	54	54	82	160	79
ICE COND. COMPT, END WALL AT CRANE WALL	4	3.5	39.9	18.4	3.75	54	54	207	275	177
FILL SLAB EL 702.78 AT CRANE WALL	6	3.1	50.9	13.5	3.75	54	54	306	319	155
FLOOR EL 756.63 AT ST GEN COMPT WALL	6	2.9	49.5	22.9	3.75	54	54	489	530	164
CRANE WALL AT EL 702.78 FILL SLAB	6	2.9	51.1	8.7	5.06	54	54	454	462	67
CRANE WALL AT ICE COND COLUMNS	6	3.6	52.2	12.9	6.0	54	54	776	972	99
CRANE WALL AT ST GEN COMPT SIDEWALL	6	2.4	51.0	10.3	5.06	54	54	681	850	164
REACTOR CAVITY WALL-4.25 FEET THICKNESS	6	1.9	46.7	5.6	5.06	54	54	481	485	147
COMPT ABOVE REACTOR-REACTOR CAVITY COLUMNS	4	1.2	40.4	2.4	5.06	54	54	532	681	97
REFUELING CANAL WALL AT CANAL FLOOR SLAB	6	1.1	34.9	2.2	5.06	54	54	351	473	144
REFUELING CANAL FLOOR SLAB	1	0.6	27.4	0.8	3.04	30	30	163	168	60
ICE COND SUPPORT FLOOR-EL 744.5	4	4.2	49.0	13.0	6.0	54	54	569	594	179
CANAL GATE	6	4.2	50.2	19.1	5.06	54	54	341	347	164
CONTROL ROD DRIVE (CRD) MISSILE SHIELD	6	4.0	50.5	16.7	5.06	54	54	295	311	164
REAC COOL PUMP & LOWER COMPT ACCESS HATCHES	6	2.0	46.1	4.4	5.06	54	54	376	408	164

NOTES:

(1) FLEXURAL STRESSES ARE IN KIPS PER SQ IN (KSI) SHEAR STRESSES ARE IN POUNDS PER SQ IN (PSI)

(2) fc_{ACTL}, fs_{ACTL}, fs'_{ACTL} - THE ACTUAL CALCULATED STRESS IN THE CONCRETE, TENSION REINFORCING STEEL AND COMPRESSION REINFORCING STEEL, RESPECTIVELY.

(3) fc_{ALL}, fs_{ALL}, fs'_{ALL} - THE ALLOWABLE STRESS IN THE CONCRETE, TENSION REINFORCING STEEL AND COMPRESSION REINFORCING STEEL, RESPECTIVELY.

(4) V_{ACTL} - THE ACTUAL CALCULATED SHEAR STRESS IN THE STRUCTURE.
V_{ALL} - THE TOTAL ALLOWABLE SHEAR STRESS THE SECTION CAN CARRY TO INCLUDE THE ALLOWABLE SHEAR STRESS CARRIED BY THE CONCRETE AS WELL AS THAT PROVIDED BY SHEAR REINFORCING.
V_c - THE ALLOWABLE SHEAR STRESS CARRIED BY THE CONCRETE ONLY.

(5) THIS TABLE DOES NOT REFLECT THE EVALUATIONS DOCUMENTED IN REPORT CEB 86-19-C. TABULATED STRESSES ARE FROM THE ORIGINAL CALCULATIONS. CHANGES HAVE BEEN DOCUMENTED IN CALCULATION PACKAGES.

(6) FOR LOAD COMBINATION DEFINITIONS, REFER TO TABLE 3.8.3-1.

TABLE 3.8.3-3 (SHEET 1 of 4)

PERSONNEL ACCESS DOORS IN CRANE WALL

LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Normal operating conditions are as follows:

Pressure	-	Negligible
Temperature	-	30° to 120° F
Radiation	-	2.0×10^7 rads for 40 year life

The effect of pipe rupture accidents on the doors varied with the location and intensity of the accidents. The three types of pipe accidents producing maximum effect on the doors and conditions accompanying these accidents are as follows:

a. Accidents Without Jets or Missles Hitting the Doors

Temperature	327° F for first hour 225° F for next 11 hours
Radiation	8.7×10^7 rads total for 12 hours
Pressure	12 psig acting from inside crane wall for 12 hours

b. Accidents With Jet Hitting a Door

Temperature	700°F maximum
Force and impact	As produced by maximum jet
Radiation	4.8×10^6 rads per hour (gamma) 2.5×10^7 rads per hour (beta)

Duration of maximum temperature and maximum force from jet is for not more than 10 seconds and then gradually decreases. Pressure and temperature after maximum temperature and force are as outlined in (a) above.

c. Accidents with Missile and Jet from the Same Source Striking a Door

Temperature	700° F maximum
Force and impact	As produced by jet and missile
Radiation	$\times 10^6$ rads per hour (gamma) 2.5×10^7 rads per hour (beta)

Duration of maximum temperature and maximum force from jet is for not more than 10 seconds and then gradually decreases. Pressure and temperature after maximum temperature and force are as outlined in (a) above.

TABLE 3.8.3-3 (SHEET 2 of 4)

PERSONNEL ACCESS DOORS IN CRANE WALL
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

Potential missiles which the doors were designed to withstand are as follows:

Temperature element A, without well, boss, and pipe

Temperature element B, with well, boss, and pipe

Temperature element C, without well

Temperature element D, with well

Reactor coolant pump temperature element

Pressurizer temperature detector

Pressurizer heater

2-inch check valve (born injection)

3/4-inch globe valve (sampling system)

(flow transmitters)
(pressure transmitters)

3/4-inch air-operated valve (head gasket monitoring)

1-inch manually-operated globe valve (excess letdown)

TABLE 3.8.3-3 (SHEET 3 of 4)

PERSONNEL ACCESS DOORS IN CRANE WALL

LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

Structural Door and Frame Assembly

No.	Load Combinations	Allowable stresses (psi)[1] Tension	Compression[3]	Shear
I.	With door closed or open: Dead load plus OBE	0.6 F_y	0.6 F_y	0.4 F_y
II.	With door closed or open: Dead load plus SSE	0.9 F_y	0.9 F_y	0.6 F_y
III.	With door closed: Dead load plus SSE plus 12 psig from inside of crane wall	0.9 F_y	0.9 F_y	0.6F_y
IV.	With door closed: Dead load plus SSE plus Load from maximum jet hitting doors at 615 psi	0.9F_y	0.9 F_y	0.6 F_y
V.	With door closed: Dead load plus SSE plus Load from missile with maximum energy 6900 lb/ft hitting door plus jet from that missile source at 295 psi	0.9 F_y	0.9 F_y	0.6 F_y

Mechanical Parts

No.	Load Combination	Allowable Stresses (psi)[1] Tension	Compression[3]	Shear
I.	With door closed or open: Dead load plus Operator force of 75 pounds	$\frac{Ult}{5}$	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
II.	With door closed or open: Dead load plus OBE	0.6F_y	0.6F_y	0.4F_y
III.	With door closed or open: Dead load plus SSE	0.9F_y	0.9F_y	0.6F_y

TABLE 3.8.3-3 (SHEET 4 of 4)

PERSONNEL ACCESS DOORS IN CRANE WALL

LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

Structural Door and Frame Assembly

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi)[1] Tension	Compression[3]	Shear
IV.	With door closed: Dead load plus SSE plus Load from maximum jet hitting doors at 615 psi	$0.9F_y$	$0.9F_y$	$0.6F_y$
V.	With door closed: Dead load plus SSE plus Load from missile with maximum energy (6900 lb/ft) hitting door plus jet from that missile source at 295 psi	$0.9F_y$	$0.9F_y$	$0.6F_y$

NOTES:

(1) Listed allowable stresses are for non-collapsible members only. Collapsible members are plastically deformed.

(2) Earthquake loads act in one horizontal direction only at any given time and act in vertical and horizontal directions simultaneously.

(3) The value indicated for the allowable compression stresses is the maximum value permitted when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^{\,2}} \right] when\ \frac{Kl}{r} \le C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8.3-4

ICE CONDENSER ALLOWABLE LIMITS[3]

Elastic Analysis

Load Combination	Mechanical[2]	Mechanical and Thermal	Fatique	Limit Analysis[1] (Load Factors)	Test (Load Factors)
D + OBE	S[4]	3S	AISC Part I	1.43	1.87
D + DBA	1.33 S	N.A.	N.A.	1.30	1.43
D + SSE	1.33 S	N.A.	N.A	1.30	1.43
D + SSE + DBA	1.65 S	N.A.	N.A	1.18	1.30

NOTES:

(1) For mechanical loads only. Mechanical plus thermal expansion, combination and fatique shall satisfy the elastic analysis limits.

(2) Membrane (direct) stresses shall be no larger than 0.7 S_u (70% of ultimate stress).

(3) For particular components that do not meet these limits specific justification shall be provided on a case by case basis.

(4) S = Allowable stresses as defined in Sections 1.5 and 1.6 of the AISC Part I Specification.

TABLE 3.8.3-5

SELECTION OF STEELS IN RELATION TO PREVENTION
OF NON-DUCTILE FRACTURE OF ICE CONDENSER COMPONENTS[1]

Properties	Section Thickness: 5/8 inch thick and under	Section Thickness: over 5/8-inch thickness
Energy Absorption Level	None required	i) 20 ft-lb CVN[2] at -20°F for steel over 36,000 psi yield strength ii) 15 ft-lb CVN[2] at -20°F for steel under 36,000 psi yield strength
Heat Treatment	None required Steel can be used in the hot rolled condition	i) Normalizing ii) Quench and Temper
Type of Steel	i) Rimmed[3] ii) Semi-killed[4] iii) Killed[4,5] iv) Killed - fine grain practice	i) Killed ii) Killed-fine grain practice

General Notes:

(1) Hot rolled, normalized or quenched and tempered steels are used where applicable.
(2) Charpy V-notch (CVN) impact testing shall be performed in accordance with the requirements of ASTM A370.
(3) Rimmed steel shall be used only for carbon steel sheet products.
(4) These type steels shall be applied for components which remain within AISC Code stress limits for all load conditions.
(5) Killed steels for above AISC Code stress limits shall be upgraded by heat treatment, e.g., bolting.

TABLE 3.8.3-6

EQUIPMENT ACCESS HATCH

SUMMARY OF ALLOWABLE STRESSES FOR DESIGN CONDITION

	I[2],II[2] Allowable	III[2] Allowable
Bending stress in structural shapes and plates (F_y = 36,000 psi)	21,600 psi (0.60 F_y)	32,400 psi (0.90 F_y)
Shear stress in structural shapes and plates (F_y = 36,000 psi)	14,400 psi (0.40 F_y)	21,600 psi (0.60 F_y)
Tensile stress in anchor bolts (F_y = 36,000 psi)	19,800 psi (0.55F_y)	31,700 psi (1.6(0.55)F_y)
Bearing stress under anchor bolt end plate (Fc' = 5,000 psi)	1,250 psi (0.25 F_c'[1])	

Notes:

(1) See Table 1002(a), ACI 318-63 Code

(2) I = DL + L1 or DL + L2

II = DL + L1 + OBE or DL + L2 + OBE

III = DL + L1 + SSE or DL + L2 + SSE

L1 = Live load of 14,000 lb (loaded weight of forklift)

L2 = Live load of 15 psi pressure from below (LOCA)

TABLE 3.8.3-7 (SHEET 1 of 2)

ESCAPE HATCH - DIVIDER BARRIER FLOOR
LOAD COMBINATIONS - ALLOWABLE STRESSES

Structural Parts - (F_y - 36,000 psi)

No.	Load Combinations	Allowable Stress (psi) Tension	Compression[(2)]	Shear
	Hatch Closed			
I.	Dead load Live load at 100 lb/ft^2 Load from latching device	18,000 (0.5 F_y)	18,000 (0.5 F_y)	12,000 (0.33F_y)
II.	Dead load Live load of 15 psi from below Load from latching device SSE[(1)]	25,900 (0.72 F_y)	25,900 (0.72 F_y)	17,300 (0.48 F_y)
	Hatch Open			
III.	Dead load OBE[(1)]	22,000 (0.6F_y)	22,000 (0.6F_y)	14,400 (0.4F_y)
IV.	Dead load SSE[(1)]	25,900 (0.72 F_y)	25,900 (0.72 F_y)	17,300 (0.48 F_y)

Mechanical Parts[(3)] (Excluding Springs)

No.	Load Combinations	Allowable Stress (psi) Tension	Compression[(2)]	Shear
	Hatch Closed			
I.	Dead load Live load at 100 lb/ft^2 Load from latching device	$\frac{\text{Ultimate}}{5}$	$\frac{\text{Ultimate}}{5}$	$\frac{2}{3} \times \frac{\text{Ultimate}}{5}$
	Hatch Open			
II.	Dead load Live load of 15 psi from below Load from latching device SSE	0.72 yield	0.72 yield	$\frac{2}{3} \times$ 0.72 yield
	Hatch Open			
III.	Dead Load OBE	0.6F_y	0.6F_y	0.4F_y
IV	Dead Load SSE	0.9F_y	0.9F_y	0.6F_y

Table 3.8.3-7 (SHEET 2 of 2)

ESCAPE HATCH - DIVIDER BARRIER FLOOR
LOAD COMBINATIONS - ALLOWABLE STRESSES

NOTES:

(1) Acts in one horizontal direction only at any given time and acts in vertical and horizontal directions simultaneously.

(2) The value given for allowable compression stress is the maximum value permitted, when buckling does not control. The critical buckling stress, Fcr, shall be used in place of Fy when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\,C_c^{\,2}} \right] \text{ when } \frac{Kl}{r} \le C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2} \text{ when } \frac{Kl}{r} > C_c \qquad 2$$

(3) Pins and shafts, bolts and nuts, bushings, and seals.

WBN

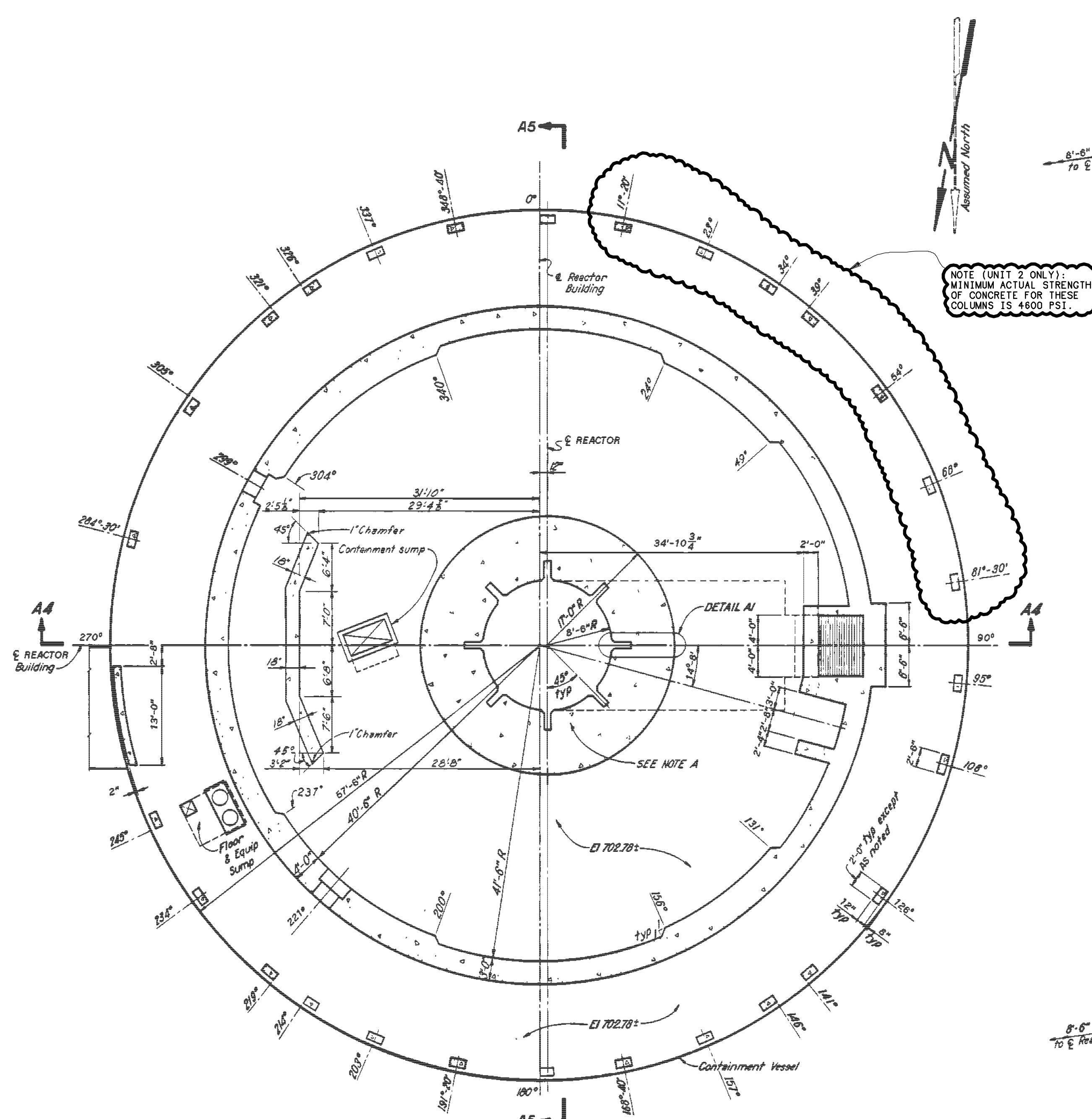
TABLE 3.8.3-8

AIR RETURN DUCT PENETRATION

SUMMARY OF STRESSES FOR CONTROLLING DESIGN CONDITION
DB LOCA - DL ± SSE

	Calculated	Allowable
Bending stress in structural shapes and plates (F_y = 36,000 psi)	17,900 psi	21,600 psi (0.60 F_y)
Tensile stress in structural shapes and plates (F_y = 36,000 psi)	1,890 psi	21,600 psi (0.60 F_y)
Headed concrete anchors (shear) (f_s' = 60,000 psi)	17,000 psi	27,000 psi (0.45 f_s')

CAD MAINTAINED DRAWING

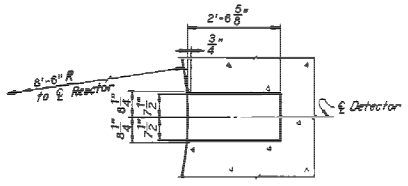


PLAN El 702.78

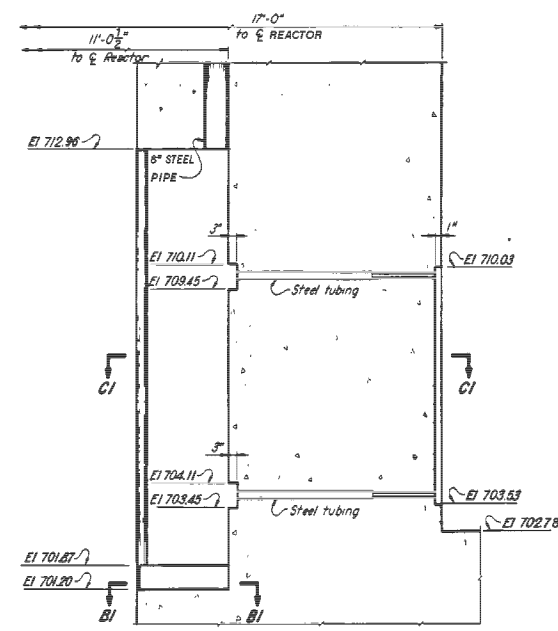
UNIT 1 AS SHOWN

UNIT 2 OPP HAND EXCEPT AS NOTED

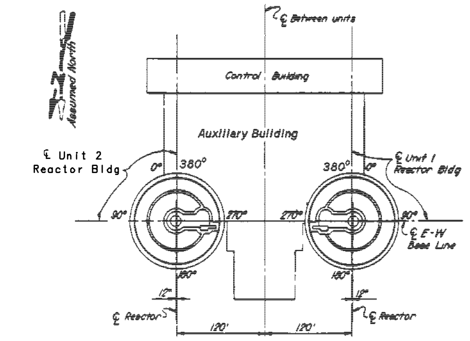
NOTE A:
THE REACTOR SUPPORTS FOR UNIT 2 ARE
NOT OPPOSITE HAND BUT HAVE AN
IDENTICAL ORIENTATION TO THOSE OF
UNIT 1.



B1-B1



A1-A1



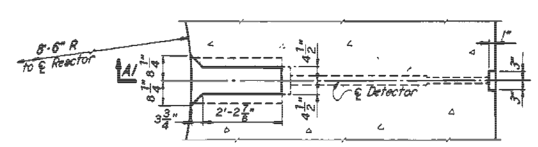
KEY PLAN

Scale 1"=100'

UNIT 2 – ACTUAL STRENGTH OF CONCRETE SHOWN
ON THIS DRAWING DERIVED FROM REPORT
CEB 86-19-C.

REFERENCE DRAWINGS:
41N18060-15A CONCRETE POUR DRAWING

NOTES:
1. THE REACTOR BUILDING IS A CLASS I STRUCTURE, AND QUALITY ASSURANCE IS REQUIRED.
2. THESE DRAWINGS SHOW AN ASSEMBLAGE OF THE INTERIOR CONCRETE STRUCTURE. FOR DETAILED OUTLINE DIMENSIONS, CLASS OF CONCRETE, AND CONSTRUCTION JOINTS SEE DETAILED COMPANION DRAWINGS.
3. CONSTRUCTION SPECIFICATION G-2 APPLIES UNLESS OTHERWISE NOTED.
4. FLOORS SHALL BE FINISHED BY STEEL TROWELING (BUT NOT "HARD TROWELED" TO A SATIN FINISH) IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-22, UNLESS OTHERWISE NOTED.
5. FOR AREAS OF PROTECTIVE COATINGS FOR SLABS AND WALLS, SEE ARCHITECTURAL DRAWINGS, 46W466 SERIES.
6. FOR EMBEDDED PARTS, PIPES, AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.
7. CHAMFER ALL EXPOSED CORNERS 3/4", UNLESS OTHERWISE NOTED.
8. FOR DETAILS OF ELECTRICAL CONDUIT ENCASEMENTS, SEE DETAIL E1, 41N710-1. ENCASEMENTS TO BE PROVIDED ONLY ON THOSE FLOORS SPECIFIED BY ELECTRICAL CONDUIT AND GROUNDING DRAWINGS.



SECTIONAL PLAN C1-C1

DETAIL A1

NEUTRON DETECTOR WINDOWS

Scale ¾"=1'-0"

NOTE:
SEE CALCULATION WCG-1-1370 FOR ENGINEERING REVIEW
AND ASSESSMENT OF THE REACTOR BUILDING CONCRETE FEATURES.

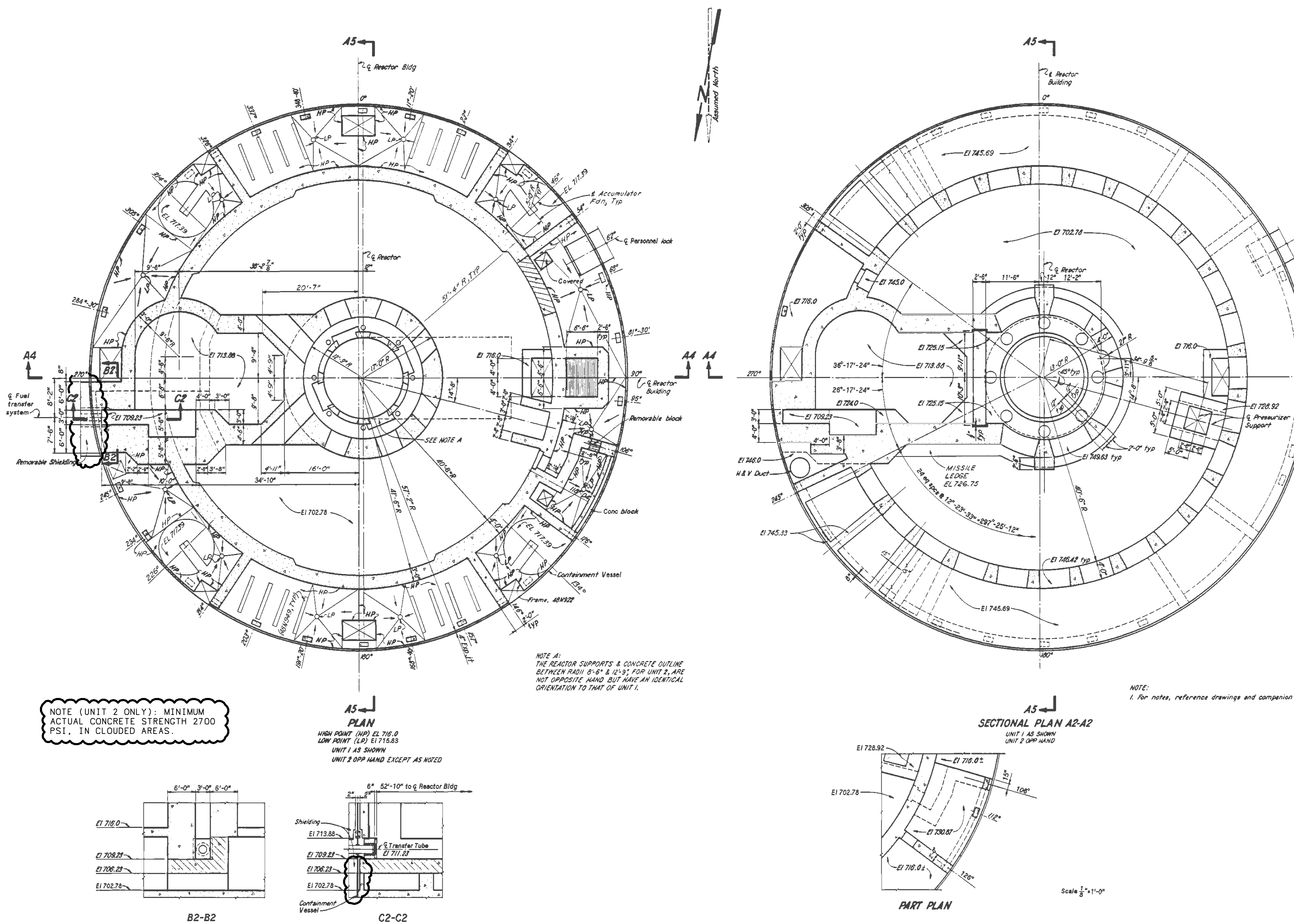
Scale ⅛"=1'-0"
Except as noted

COMPANION DRAWINGS:
41N716-2 THROUGH 41N716-5
41N721-1 --------------- CRANE WALL
41N726-1 --------------- REACTOR SHIELD WALL & REFUELING CANAL
41N729-1 --------------- STEAM GENERATOR & PRESSURIZER ENCLOSURE
41N731-1 --------------- FLOOR SLAB - EL 716
41N732-1 --------------- MISCELLANEOUS WALLS
41N733-1 --------------- ICE CONDENSER FLOOR SLAB - EL 744
41N735-1 --------------- FLOOR SLAB - EL 757
41N741-1 --------------- PRECAST COVERS & BULKHEADS
41N710-1 --------------- FLOOR SLAB - EL 702.78

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
INTERIOR STRUCTURE
OUTLINE
TVA DWG NO. 41N716-1 RD
FIGURE 3.8.3-1

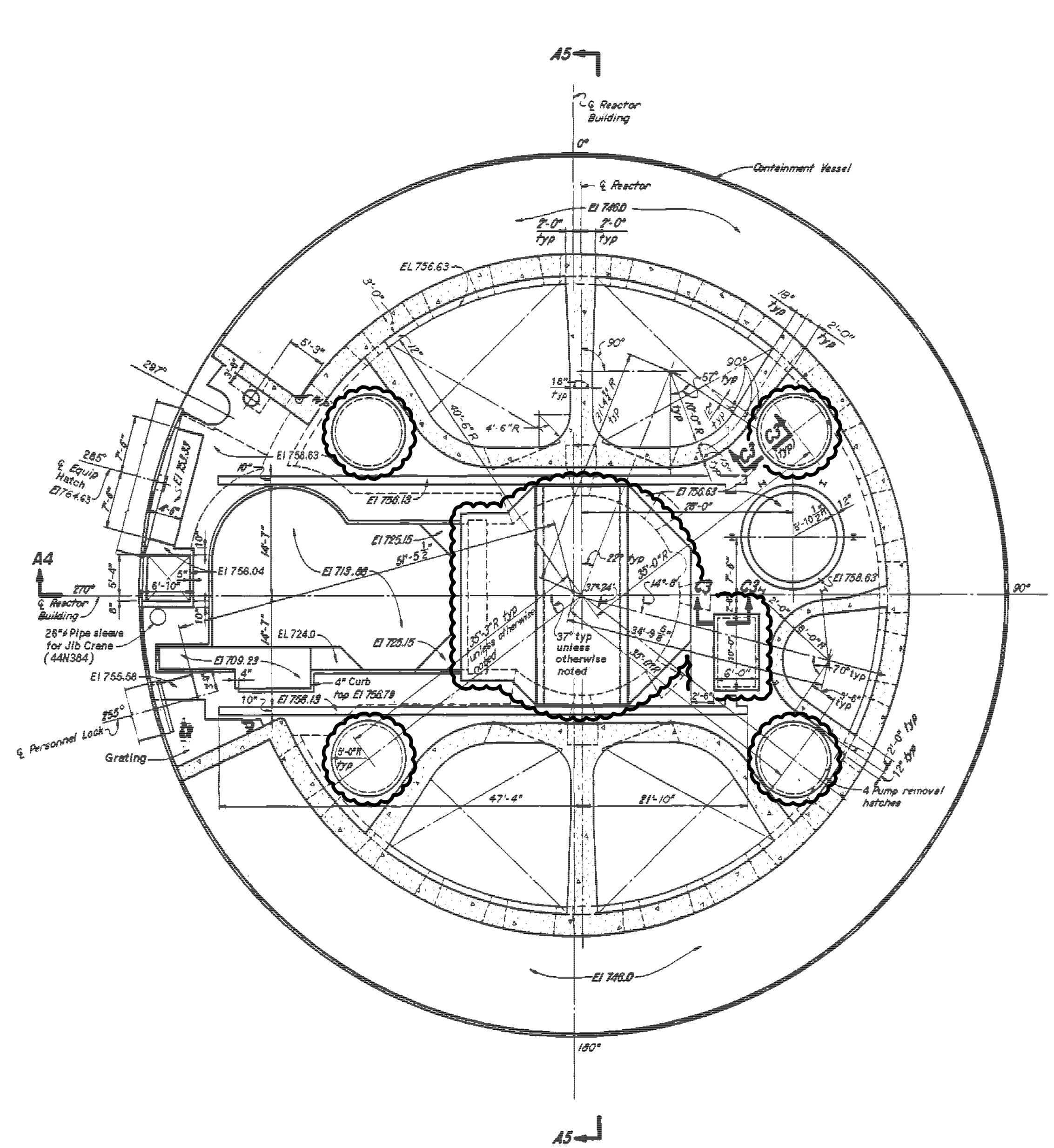


UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

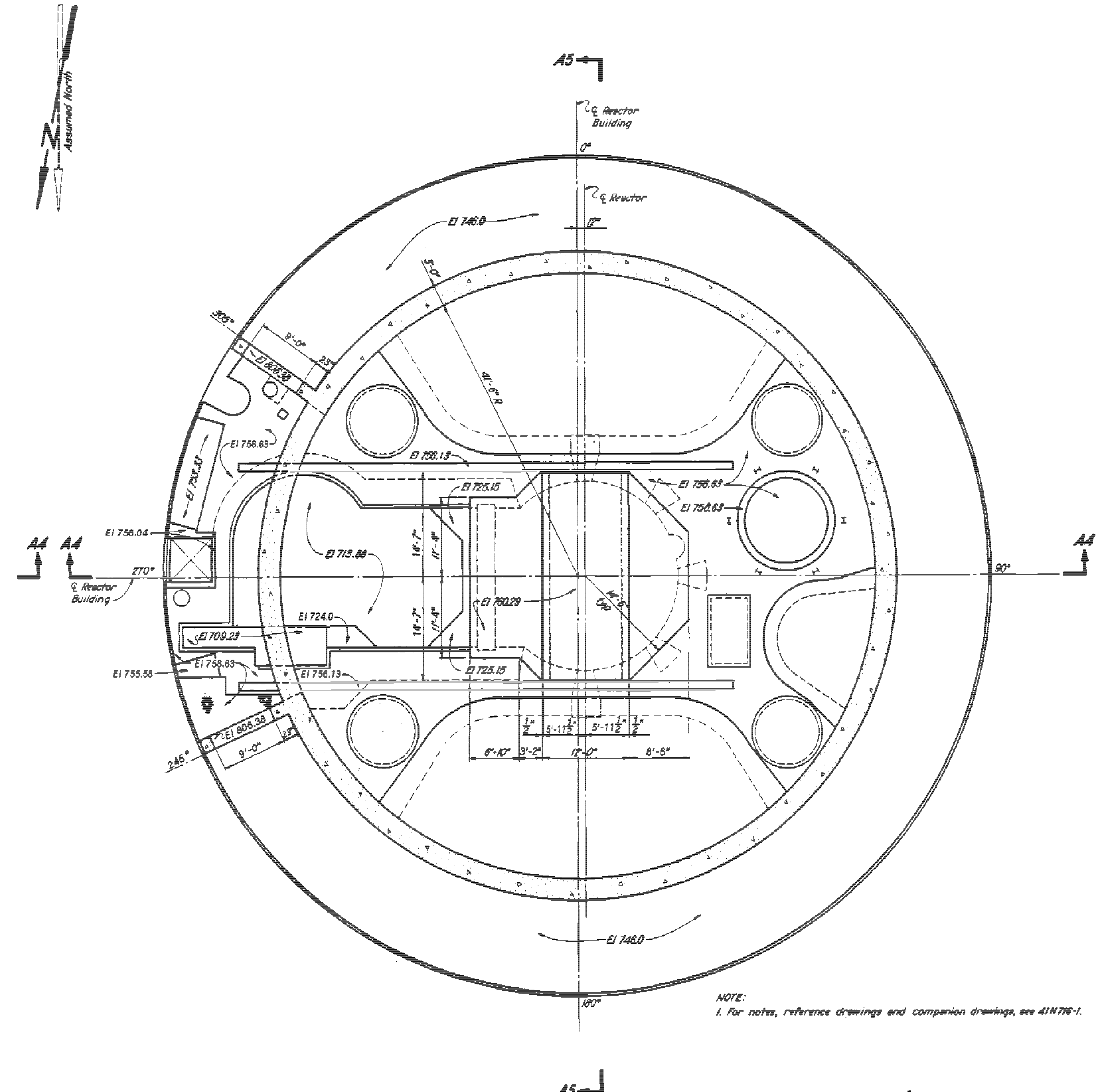
REACTOR BUILDING
UNITS 1 & 2
CONCRETE
INTERIOR STRUCTURE
OUTLINE
TVA DWG NO. 41N716-2 RD
FIGURE 3.8.3-2

CAD MAINTAINED DRAWING



SECTIONAL PLAN A3-A3

UNIT 1 AS SHOWN
UNIT 2 OPP HAND

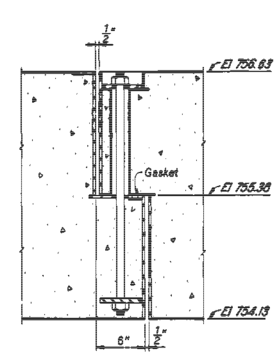


SECTIONAL PLAN B3-B3

UNIT 1 AS SHOWN
UNIT 2 OPP HAND

NOTE:
1. For notes, reference drawings and companion drawings, see 41N716-1.

Scale $\frac{1}{8}$" = 1'-0"



C3-C3
Scale $1\frac{1}{2}$" = 1'-0"

(UNIT 2 ONLY)
MIN. ACTUAL CONCRETE STRENGTH
IS 6600 PSI FOR ALL CLOUDED AREAS.

UNIT 1- DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20.1.1. R7

UNIT 2- ACTUAL STRENGTH OF CONCRETE SHOWN IN THIS DRAWING DERIVED FROM REPORT CEB 86-19-C.

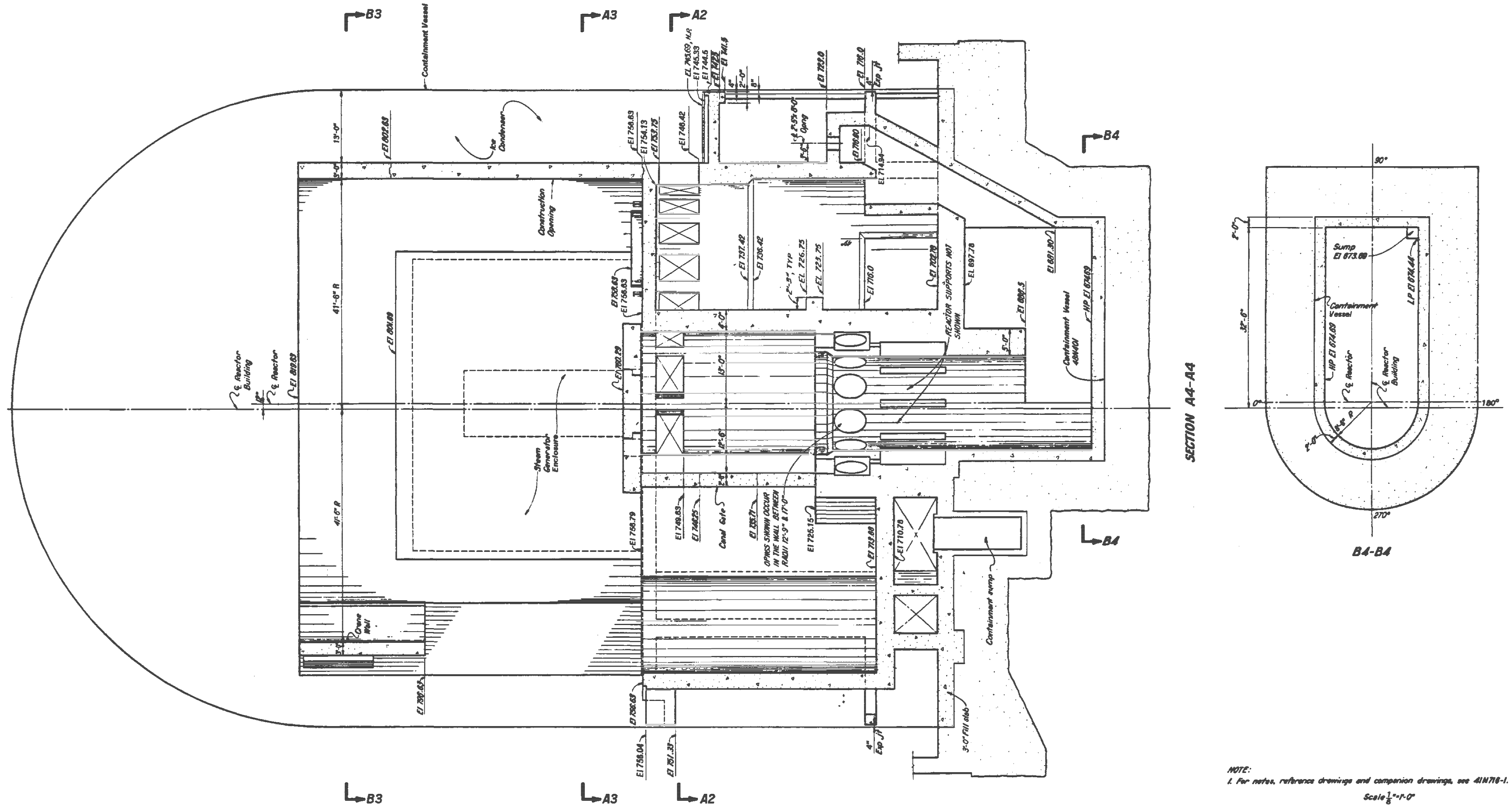
REFERENCE DRAWINGS
41N10060-18....CONCRETE POUR DRAWING

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
INTERIOR STRUCTURE
OUTLINE
TVA DWG NO. 41N716-3 RD
FIGURE 3.8.3-3

CAD MAINTAINED DRAWING



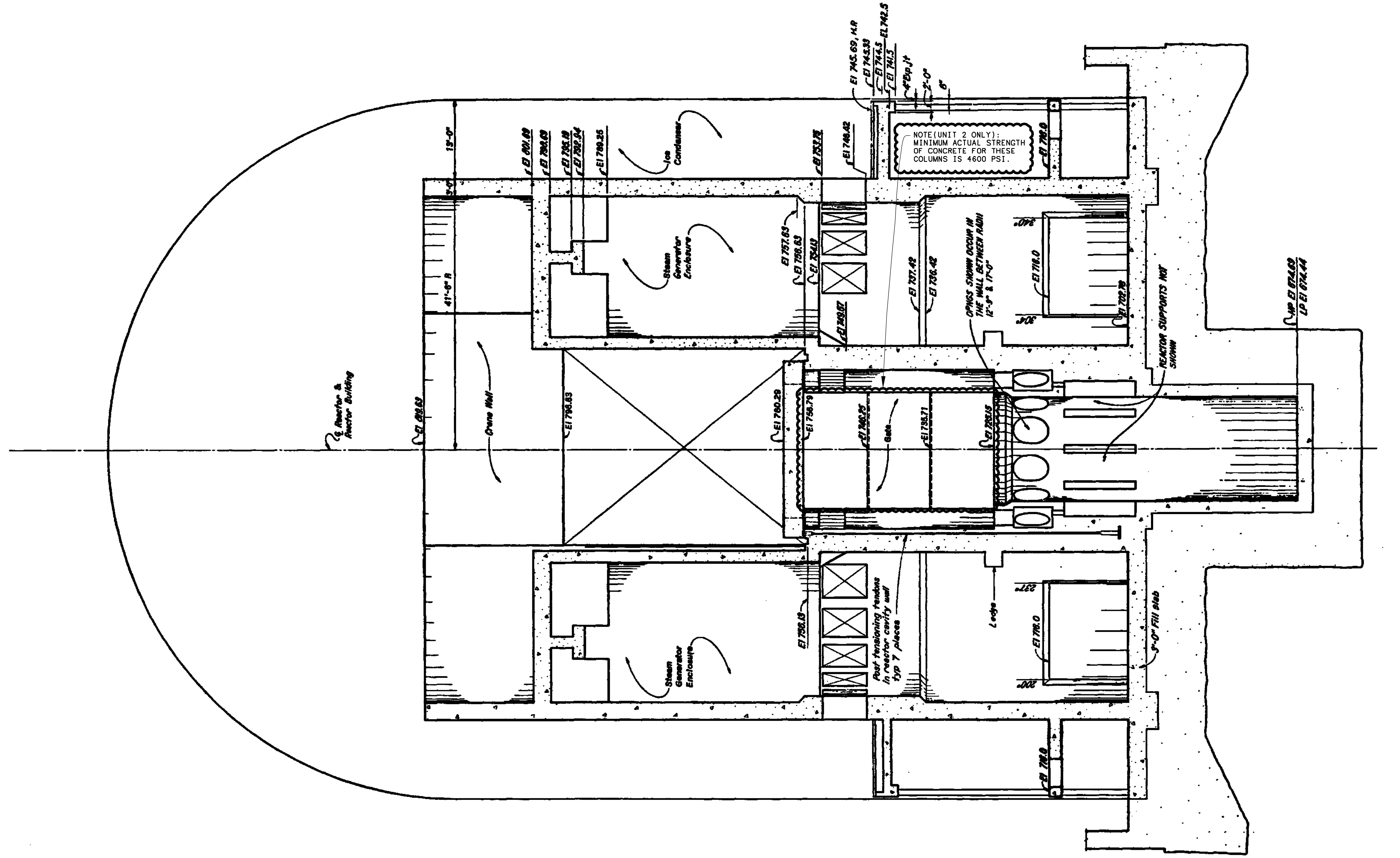
NOTE:
1. For notes, reference drawings and companion drawings, see 41N716-1.

Scale $\frac{1}{8}$"=1'-0"

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
INTERIOR STRUCTURE
OUTLINE
TVA DWG NO. 41N716-4 RD
FIGURE 3.8.3-4

NOTE:
1. FOR NOTES, REFERENCE DRAWINGS AND COMPANION DRAWINGS, SEE 41N716-1



REFERENCE DRAWINGS:
41N10060-18 --- CONCRETE POUR DRAWING

SCALE: ¼"=1'-0"

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
INTERIOR STRUCTURE
OUTLINE
TVA DWG NO. 41N716-5 RD
FIGURE 3.8.3-5

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.3-6

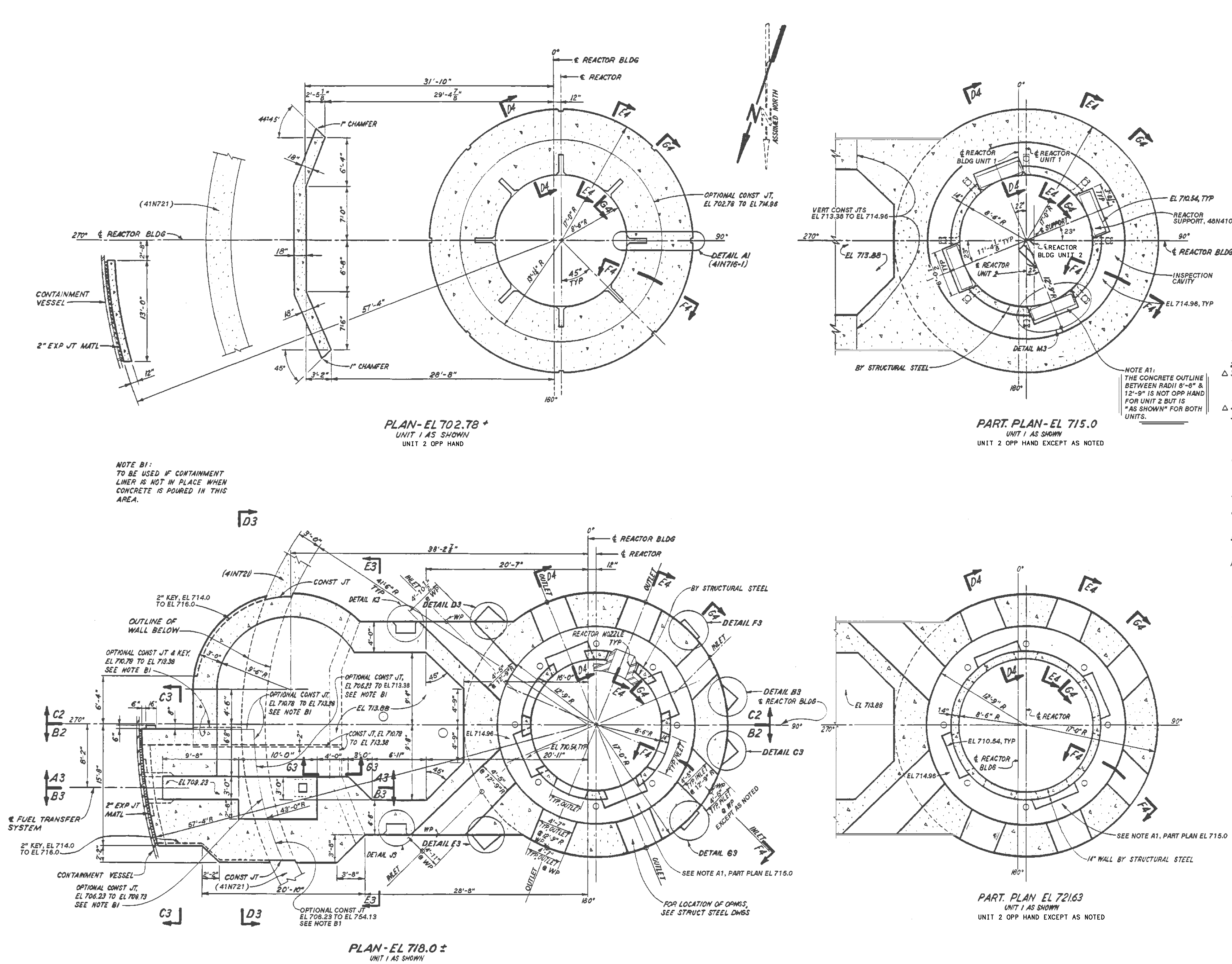
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Plan - Upper Compartment 1 Ft = 0.3048m FIGURE 3.8.3-6

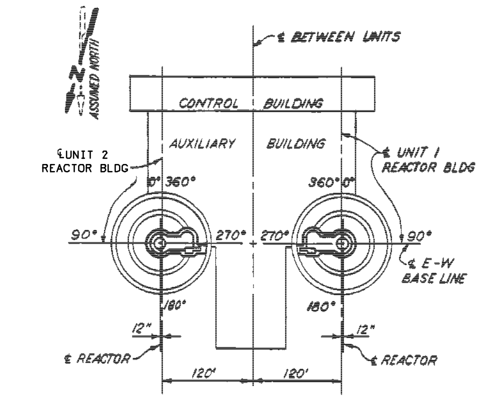
SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.3-7

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Plan - Lower Compartment 1 Ft = 0.3048m FIGURE 3.8.3-7

CAD MAINTAINED DRAWING





KEY PLAN
SCALE 1" = 100'

△UNIT 1 - DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

UNIT 2 - ACTUAL STRENGTH OF CONCRETE SHOWN ON THIS DRAWING 41N726-2 DERIVED FROM REPORT CEB 86-19-C.

NOTES:

1. THE REACTOR BUILDING IS A CLASS I STRUCTURE AND QUALITY ASSURANCE IS REQUIRED.
2. FOR GENERAL NOTES, SEE 41N716-1.
3. △ CONCRETE SHALL BE CLASS 500.75A FW, EXCEPT AS NOTED. IN ADDITION TO NORMAL REQUIREMENTS, COMPRESSIVE STRENGTH TESTS OF THIS CONCRETE ARE REQUIRED AT 90 AND 180 DAYS TO VERIFY ULTIMATE DESIGN STRENGTH IN EXCESS OF 6750 PSI. (SEE NOTE 10)
4. △ THE 6" FILL CONCRETE SHALL BE CLASS 400.75A FW.
5. DIMENSIONS GIVEN TO INSIDE FACE OF REFUELING CANAL AND REACTOR SHIELD WALLS ARE TO THE EXPOSED FACE OF THE STRUCTURAL STEEL LINER PLATE, NOT SHOWN. FOR LINER PLATE, SEE STRUCT. STEEL DWGS 48W930, 48N931, 48W932, 48W933 AND 48N935-1 THRU 48N935-18.
6. GROUT SHALL CONSIST OF 1 PART CEMENT AND 1-1/2 PARTS SAND BY WEIGHT, WITH MINIMUM WATER REQUIRED TO PRODUCE A MIX WHICH WILL FLOW WITHOUT SEGREGATING BUT NOT TO EXCEED 4 GAL PER SACK OF CEMENT.
7. FOR EMBEDDED ANCHORAGES, SLEEVES AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL AND ELECTRICAL DWGS.
8. THE CONCRETE BETWEEN RADII 8'-6 AND 12'-9" AND ABOVE EL 714.96 MAY BE POURED IN LIFTS AS REQUIRED TO FACILITATE PLACEMENT OF REACTOR VESSEL & STRUCTURAL STEEL IN THIS AREA.
9. THE CONCRETE REPRESENTED BY CROSS-HATCHED AREAS SHALL BE CLASS 301.5 (D)A FW, SEE MEMO TO J.C. KILLIAN FROM R.M. PIERCE AUG 22, 1974. (B26900306100)
10. WHERE CONCRETE CANNOT BE PROPERLY COMPACTED IN FORMS DUE TO HIGH CONCENTRATIONS OF EMBEDMENTS, THE PROJECT MATERIALS ENGINEER MAY SUBSTITUTE A CONCRETE MIX OF EQUAL OR GREATER STRENGTH AND WITH SMALLER SIZE COARSE AGGREGATE FOR THAT REQUIRED BY NOTE 3. THIS ALTERNATE MIX SHALL BE APPROVED BY DED.
11. PERMANENT SIGNS SHALL BE PLACED IN CROSS HATCHED AREAS OF HIGH DENSITY CONCRETE TO AVOID THE POSSIBILITY OF ATTACHMENT TO THESE AREAS WITHOUT ENGINEERING EVALUATION. FOR THE DETAILS OF SIGN SEE SHEET 2. ALSO PROVIDE 2" WIDE BORDER LINE COATED IN BLACK EXCEPT AT INTERFERENCES ALONG THE EDGES OF THE HIGH DENSITY CONCRETE AREAS.

    UNIT 1: (ONLY) COATING SYSTEM WBNP-N-978 DRY FILM THICKNESS CRITERIA IS NOT REQUIRED.

    UNIT 2: (ONLY) INTERRUPTED BOUNDARY LINES MAY BE STOPPED MAX 1/2" TO THE INTERFERENCES. COATING SYSTEM WBNP-N-978 DRY FILM THICKNESS CRITERIA IS NOT REQUIRED. BACKGROUND TO BE PAINTED YELLOW PRIOR TO PAINTING SIGN AND BORDER. BACKGROUND TO BE USED IS AMERON AMERLOCK 400NT YELLOW.
12. SEE CALCULATION WCG-1-1370 FOR ENGINEERING REVIEW AND ASSESSMENT OF THE REACTOR BUILDING CONCRETE FEATURES.

REFERENCE DRAWING:
41BM726-1........BILL OF MATERIAL
41N10060-14...CONCRETE POUR DRAWING

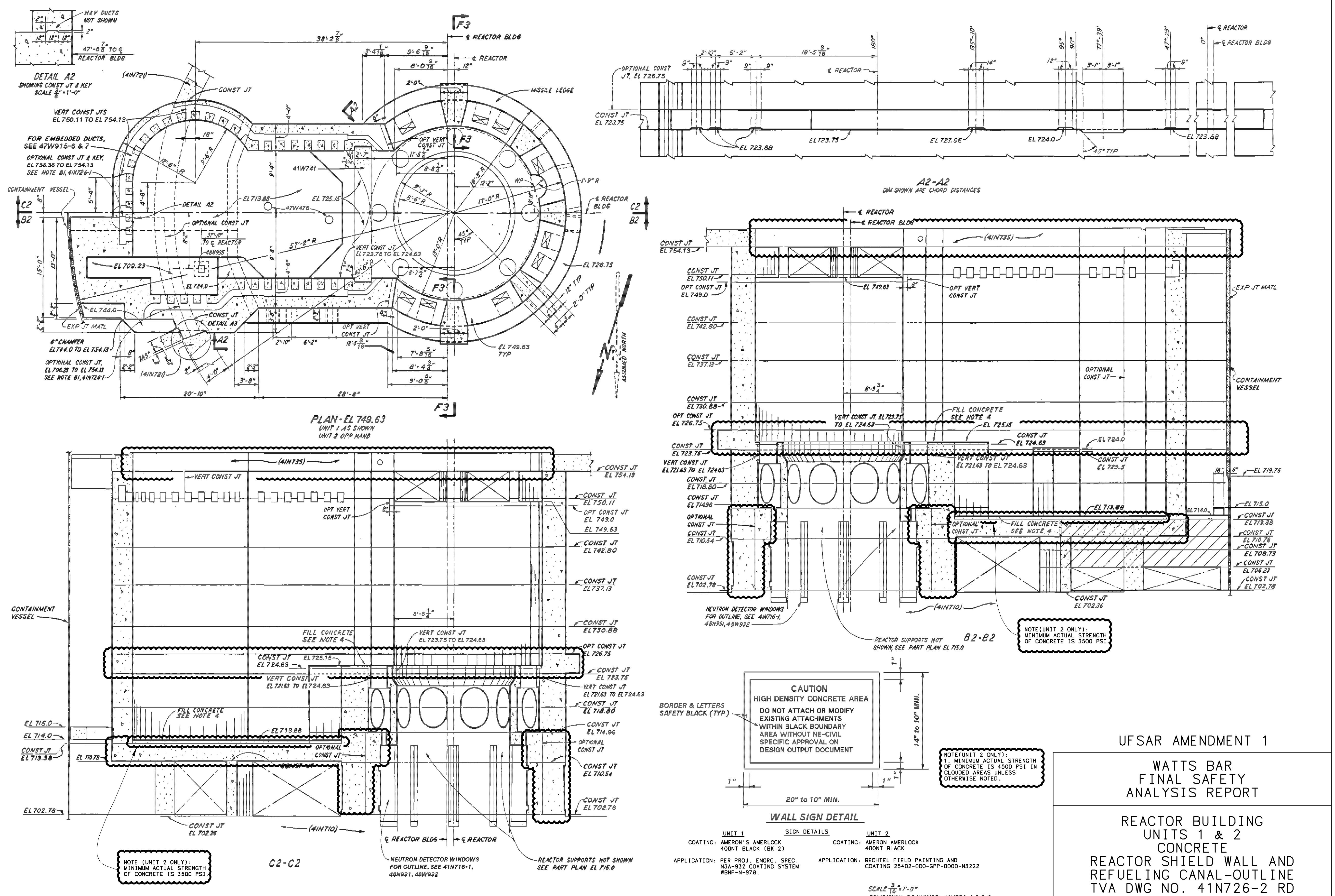
SCALE 3/8" = 1'-0" EXCEPT AS NOTED
COMPANION DRAWINGS: 41N726-2, 3 & 4

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
REACTOR SHIELD WALL AND
REFUELING CANAL-OUTLINE
TVA DWG NO. 41N726-1 RE
FIGURE 3.8.3-7A

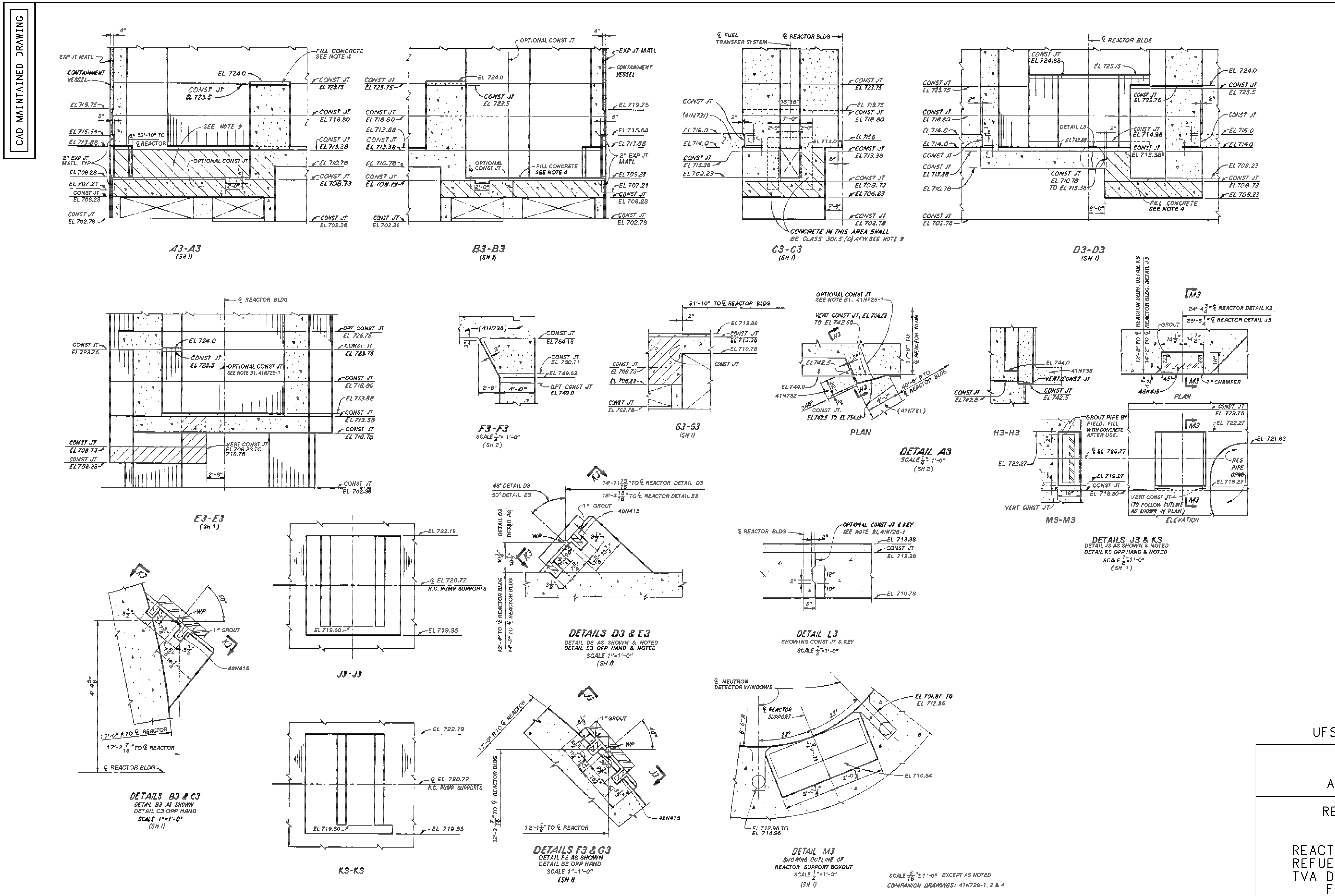
CAD MAINTAINED DRAWING



UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
REACTOR SHIELD WALL AND
REFUELING CANAL-OUTLINE
TVA DWG NO. 41N726-2 RD
FIGURE 3.8.3-7B

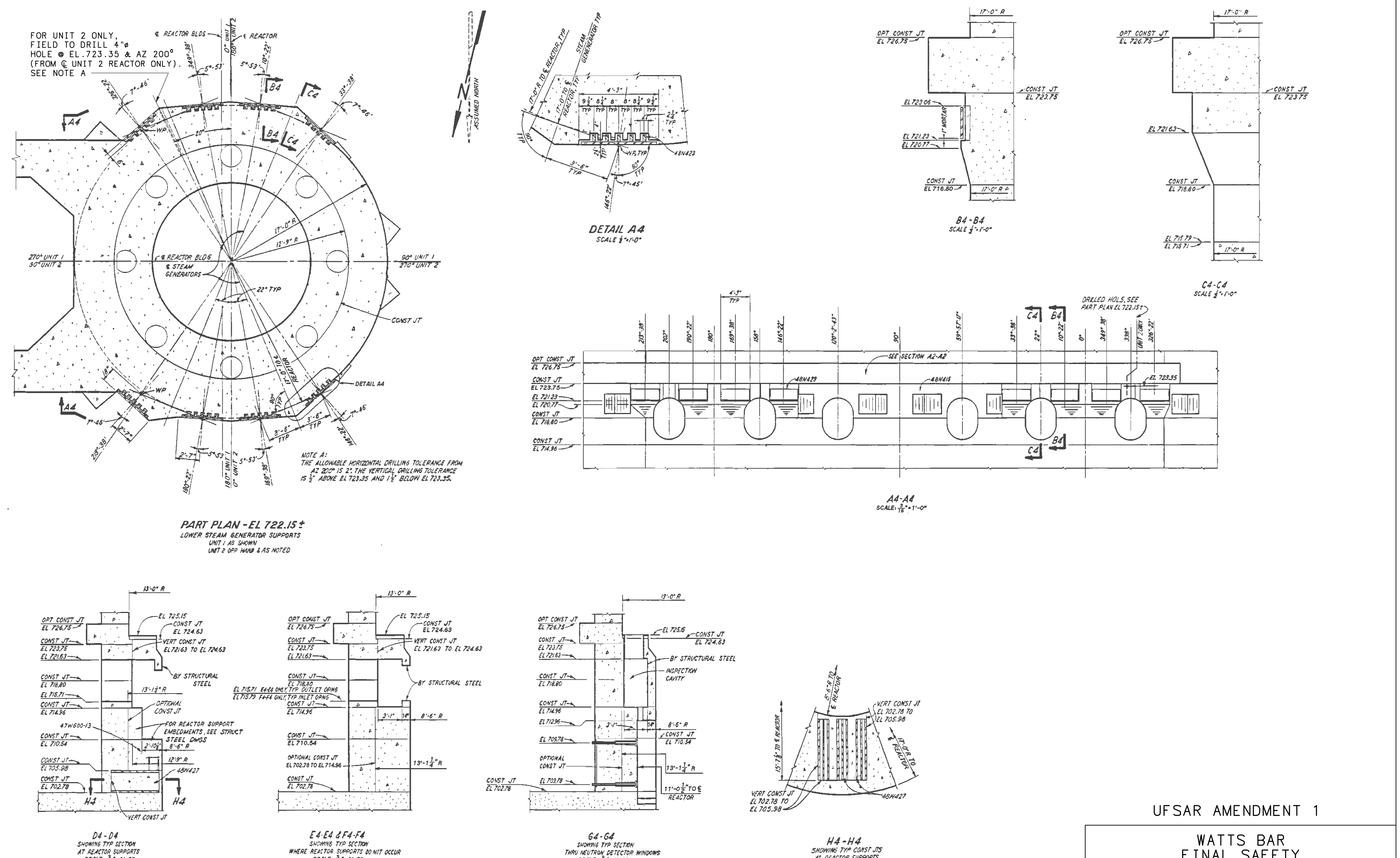


UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
REACTOR SHIELD WALL AND
REFUELING CANAL-OUTLINE
TVA DWG NO. 41N726-3 RD
FIGURE 3.8.3-7C

CAD MAINTAINED DRAWING

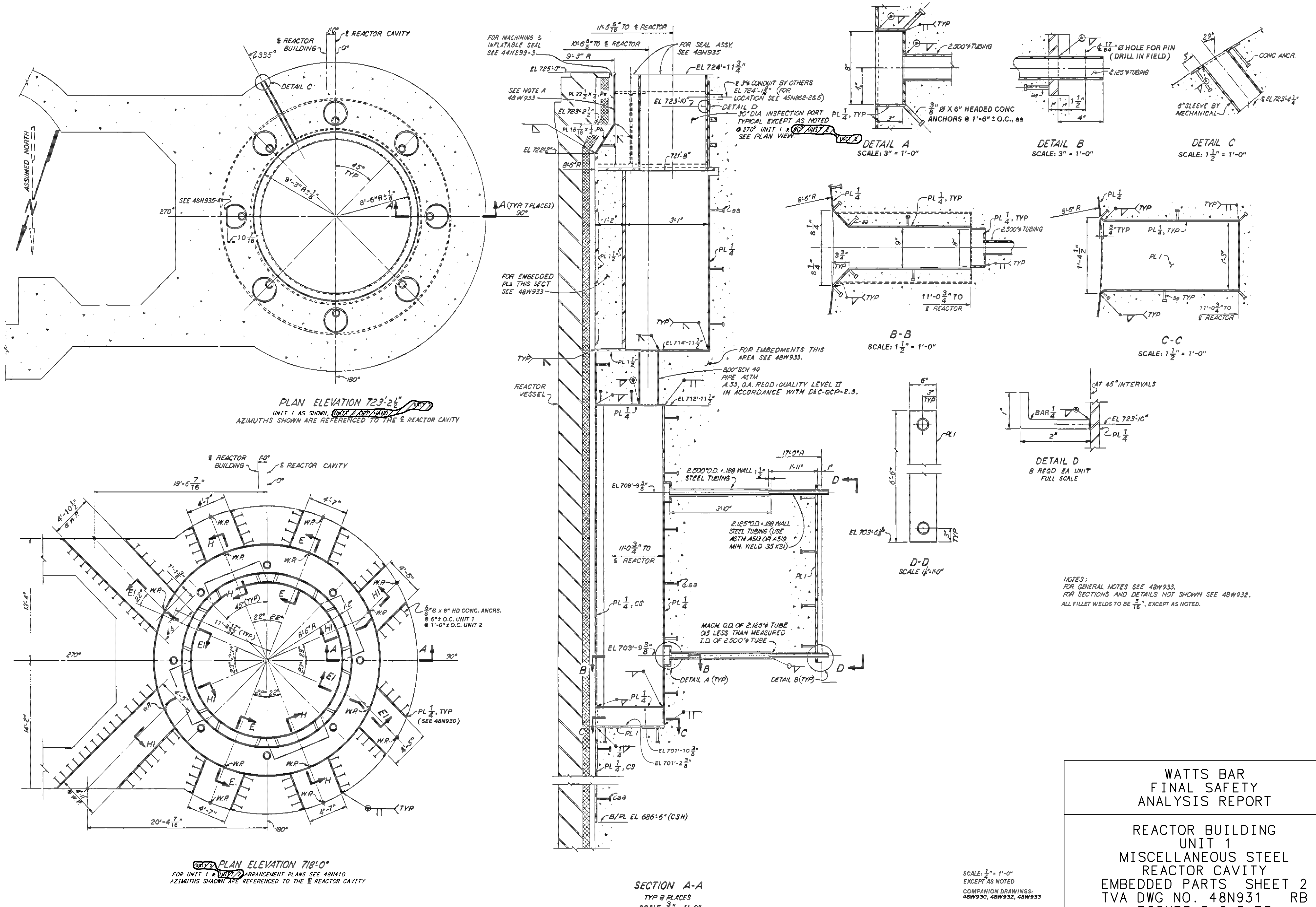


UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
REACTOR SHIELD WALL AND
REFUELING CANAL-OUTLINE
TVA DWG NO. 41N726-4 RE
FIGURE 3.8.3-7D

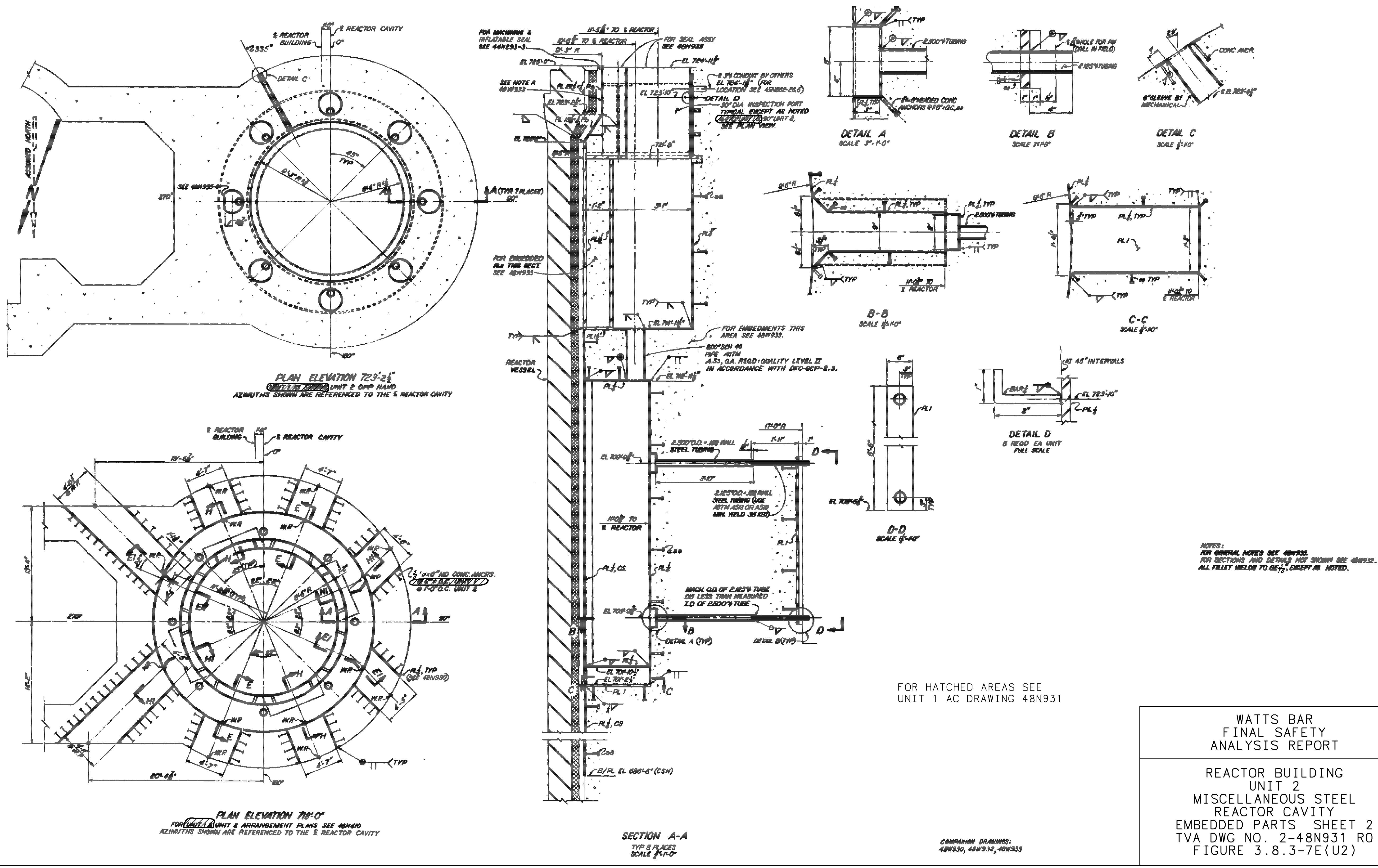
CAD MAINTAINED DRAWING



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL
REACTOR CAVITY
EMBEDDED PARTS SHEET 2
TVA DWG NO. 48N931 RB
FIGURE 3.8.3-7E

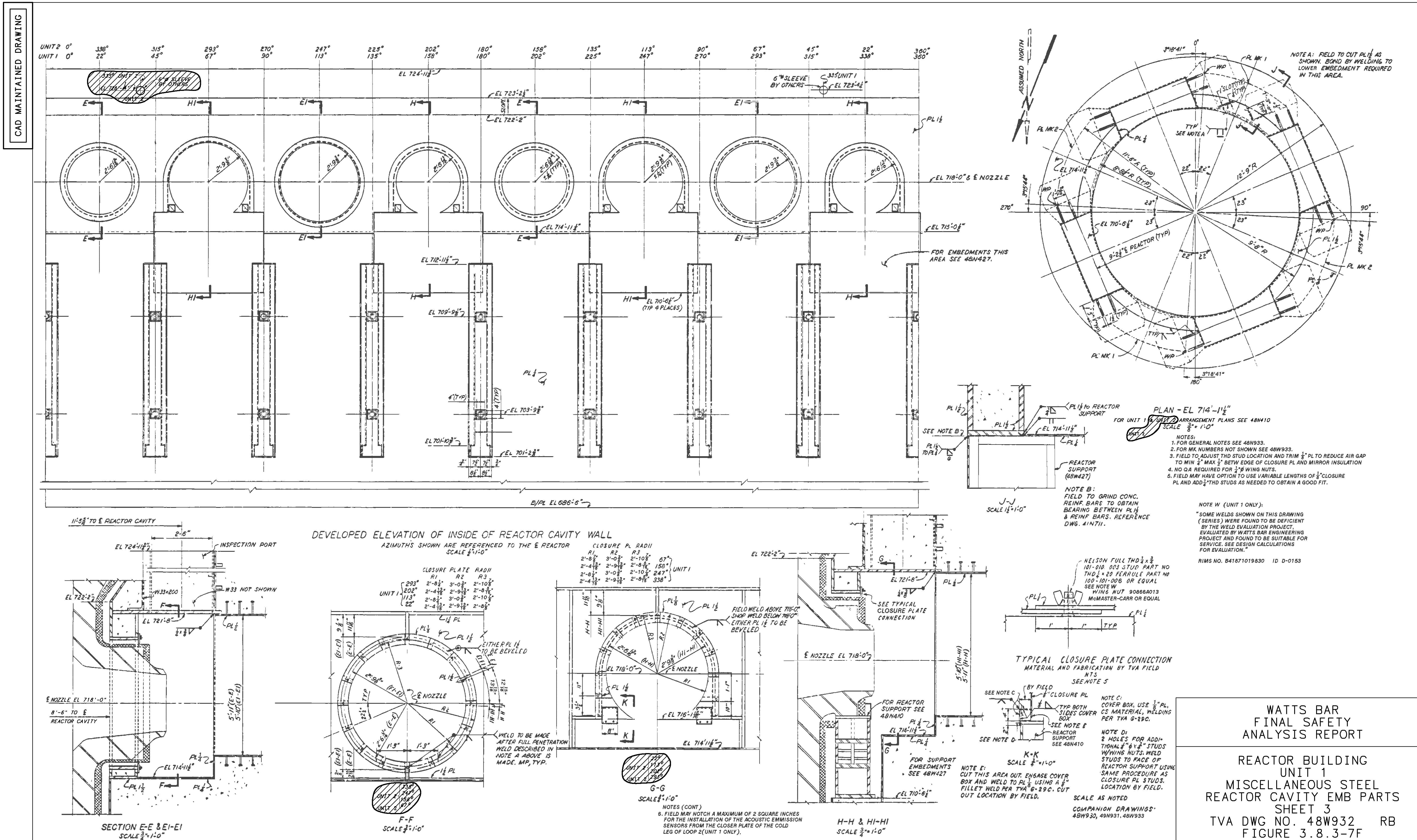
CAD MAINTAINED DRAWING



FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 48N931

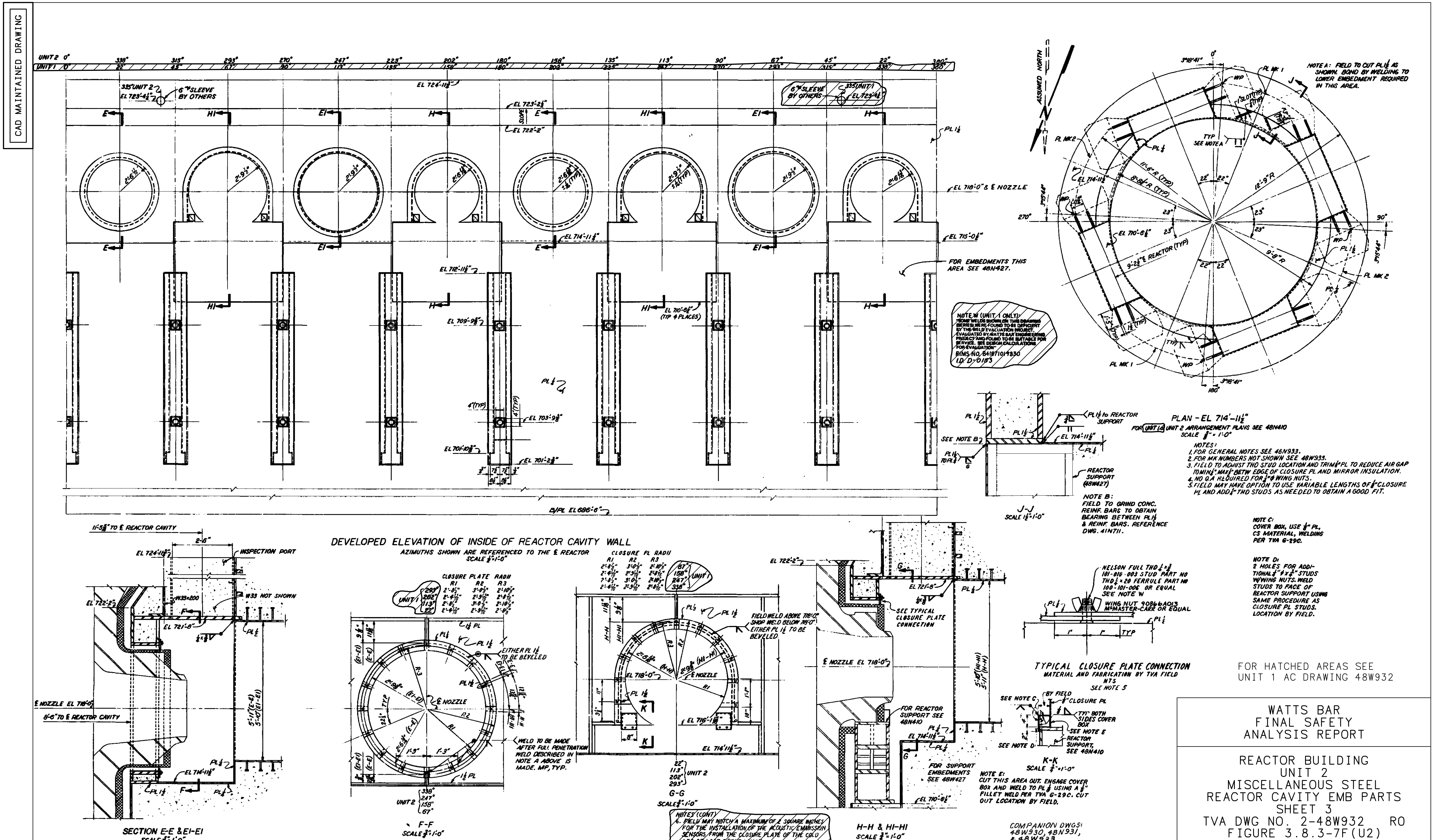
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL
REACTOR CAVITY
EMBEDDED PARTS SHEET 2
TVA DWG NO. 2-48N931 R0
FIGURE 3.8.3-7E(U2)

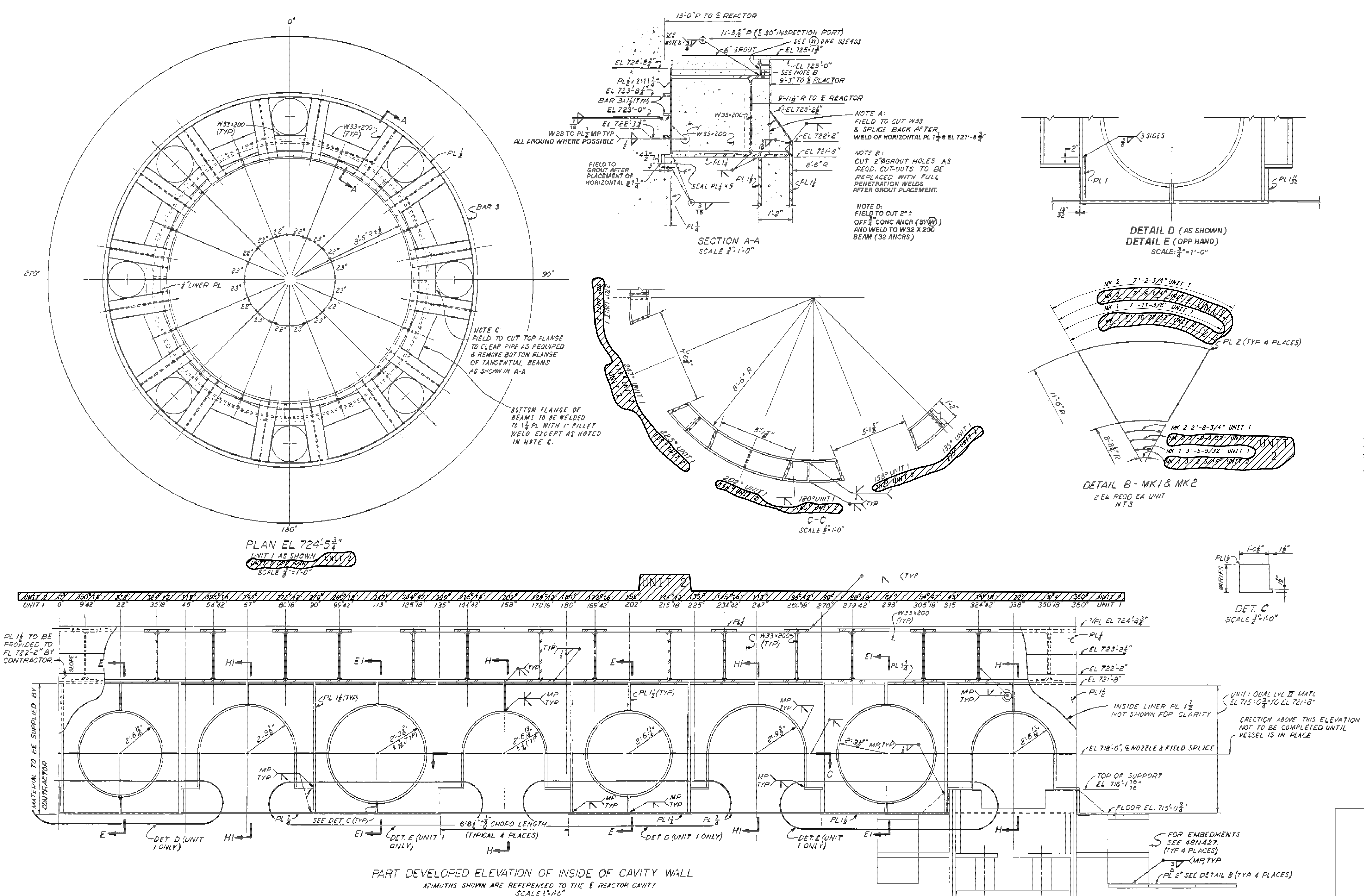


WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL
REACTOR CAVITY EMB PARTS
SHEET 3
TVA DWG NO. 48W932 RB
FIGURE 3.8.3-7F


CAD MAINTAINED DRAWING
DEVELOPED ELEVATION OF INSIDE OF REACTOR CAVITY WALL
AZIMUTHS SHOWN ARE REFERENCED TO THE ℄ REACTOR
SCALE 1/2"=1'-0"
PLAN - EL 714'-11 1/2"
FOR UNIT 1 & UNIT 2 ARRANGEMENT PLANS SEE 48N410
NOTES:
1. FOR GENERAL NOTES SEE 46N933.
2. FOR MK NUMBERS NOT SHOWN SEE 48W933.
3. FIELD TO ADJUST THD STUD LOCATION AND TRIM 1/4" PL TO REDUCE AIR GAP TO MIN 1/4" MAX 1/2" BETW EDGE OF CLOSURE PL AND MIRROR INSULATION.
4. NO Q.A REQUIRED FOR 1/2" Ø WING NUTS.
5. FIELD MAY HAVE OPTION TO USE VARIABLE LENGTHS OF 1/4" CLOSURE PL AND ADD 1/4" THD STUDS AS NEEDED TO OBTAIN A GOOD FIT.
NOTE A: FIELD TO CUT PL 1 1/2 AS SHOWN. BOND BY WELDING TO LOWER EMBEDMENT REQUIRED IN THIS AREA.
NOTE B: FIELD TO GRIND CONC. REINF. BARS TO OBTAIN BEARING BETWEEN PL 1 1/2 & REINF. BARS. REFERENCE DWG. 41N711.
NOTE C: COVER BOX, USE 1/4" PL, CS MATERIAL, WELDING PER TVA G-29C.
NOTE D: 2 HOLES FOR ADDITIONAL 1/2" Ø x 2" STUDS W/WING NUTS. WELD STUDS TO FACE OF REACTOR SUPPORT USING SAME PROCEDURE AS CLOSURE PL STUDS. LOCATION BY FIELD.
NOTE E: CUT THIS AREA OUT. ENGAGE COVER BOX AND WELD TO PL 1/2 USING A 1/4" FILLET WELD PER TVA G-29C. CUT OUT LOCATION BY FIELD.
NOTE W (UNIT 1 ONLY)
NOTES (CONT)
6. FIELD MAY NOTCH A MAXIMUM OF 2 SQUARE INCHES FOR THE INSTALLATION OF THE ACOUSTIC EMISSION SENSORS FROM THE CLOSURE PLATE OF THE COLD LEG OF LOOP 2 (UNIT 1 ONLY).
FOR EMBEDMENTS THIS AREA SEE 48N427.
TYPICAL CLOSURE PLATE CONNECTION
MATERIAL AND FABRICATION BY TVA FIELD
NTS
SEE NOTE 5
SECTION E-E & E1-E1
SCALE 3/8"=1'-0"
F-F
G-G
H-H & H1-H1
J-J
K-K
FOR HATCHED AREAS SEE UNIT 1 AC DRAWING 48W932
COMPANION DWGS: 48W930, 48N931, & 48W933
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT
REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL
REACTOR CAVITY EMB PARTS
SHEET 3
TVA DWG NO. 2-48W932 R0
FIGURE 3.8.3-7F(U2)

CAD MAINTAINED DRAWING



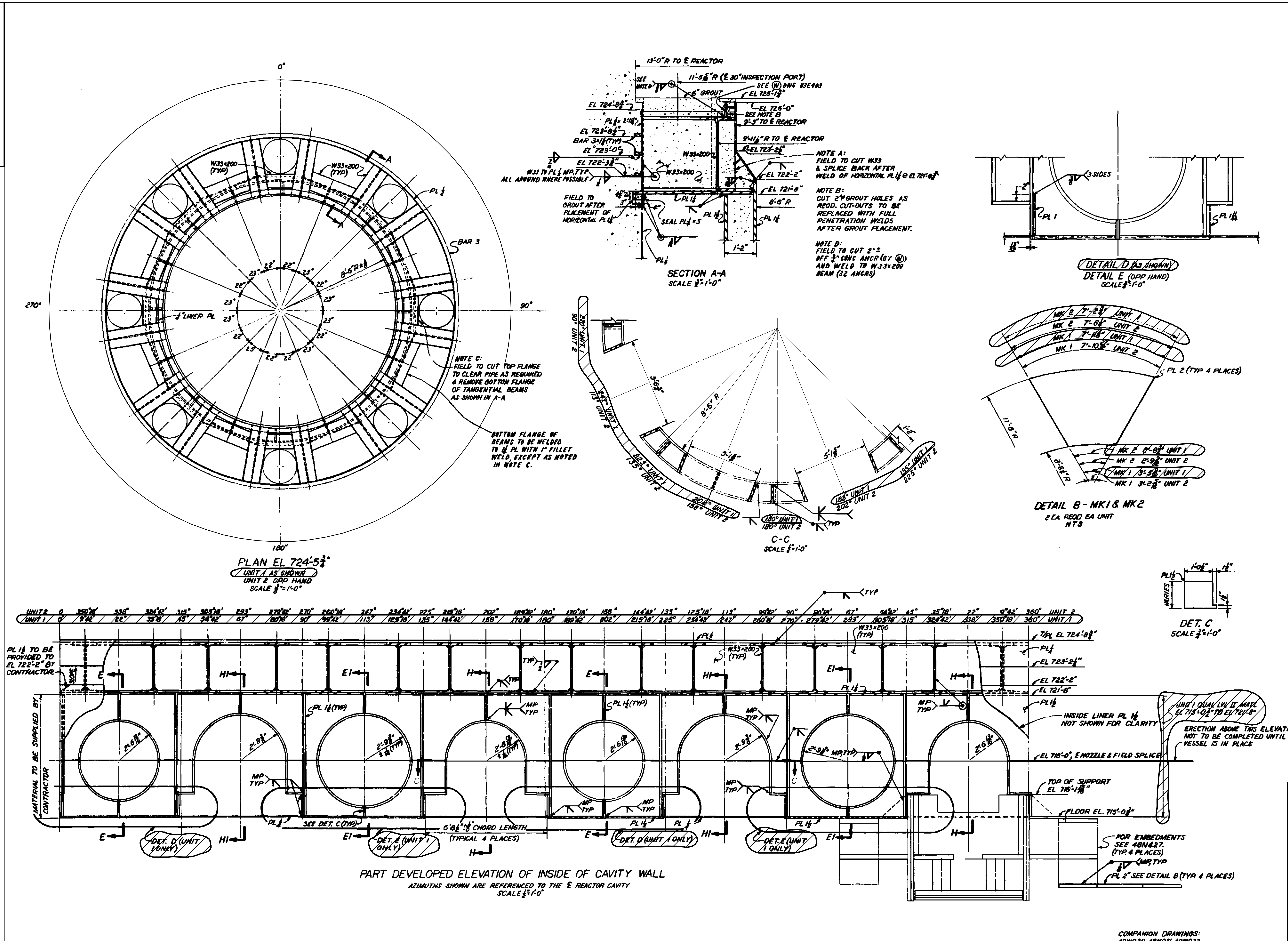
NOTES:

1. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
2. Q.A. FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONSTRUCTION SPECIFICATION N3G-881.
3. ALL MATERIAL AND FABRICATION BY TVA FIELD, EXCEPT AS NOTED.
4. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
5. ALL WELDING TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFCATION G-29C.
6. WELDING DOCUMENTATION AND INSPECTION TO BE IN ACCORDANCE WITH CONSTRUCTION SPECECIFICATION N3G-881.
7. ALL CARBON STEEL WELDING ELECTRODES TO BE AWS A5.1 E70 SERIES.
8. ALL MATERIAL TO BE ASTM A36, EXCEPT AS NOTED.
9. FOR SECTIONS NOT SHOWN SEE 48W932.
10. ALL SPLICES TO BE FULL PENETRATION WELDS.
11. BACK-UP BARS & FIELD SPLICES EXCEPTAS NOTED NOT SHOWN.
12. MISMATCH OF VERTICAL PL'S AT HORIZ FIELD SPLICE EL 718'-0" TO BE 1/8" MAX.
13. PROCUREMENT, FABRICATION AND DOCUMENTATION OF MATERIAL BY CONTRACTOR SHALL CONFORM TO THE TVA QA PLAN.
14. CONC. ANCRS. TO BE ASTM A108 AND ONLY THAT THEY CONFORM TO THE APPROPRIATE ASTM SPECIFICATION.

SCALE AS NOTED
COMPANION DRAWINGS: 48W930, 48N931, 48W932

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BLDG
UNIT 1
MISCELLANEOUS STEEL
REACTOR CAVITY
EMBEDDED PARTS SH 4
TVA DWG NO. 48W933 RB
FIGURE 3.8.3-7G



NOTES:
1. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
2. Q.A FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONSTRUCTION SPECIFICATION N3G-881.
3. ALL MATERIAL AND FABRICATION BY TVA FIELD, EXCEPT AS NOTED.
4. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
5. ALL WELDING TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.
6. WELDING DOCUMENTATION AND INSPECTION TO BE IN ACCORDANCE WITH CONSTRUCTION SPECECIFICATION N3G-881.
7. ALL CARBON STEEL WELDING ELECTRODES TO BE AWS A5.1 E70 SERIES.
8. ALL MATERIAL TO BE ASTM A36, EXCEPT AS NOTED.
9. FOR SECTIONS NOT SHOWN SEE 48W932.
10. ALL SPLICES TO BE FULL PENETRATION WELDS.
11. BACK-UP BARS & FIELD SPLICES EXCEPT AS NOTED NOT SHOWN.
12. MISMATCH OF VERTICAL PL'S AT HORIZ. FIELD SPLICE EL 716'-0" TO BE 1/4" MAX.
13. PROCUREMENT, FABRICATION AND DOCUMENTATION OF MATERIAL BY CONTRACTOR SHALL CONFORM TO THE TVA QA PLAN.
14. CONC. ANCRS. TO BE ASTM A108 AND REQUIRE ONLY THAT THEY CONFORM TO THE APPROPRIATE ASTM SPECIFICATION.

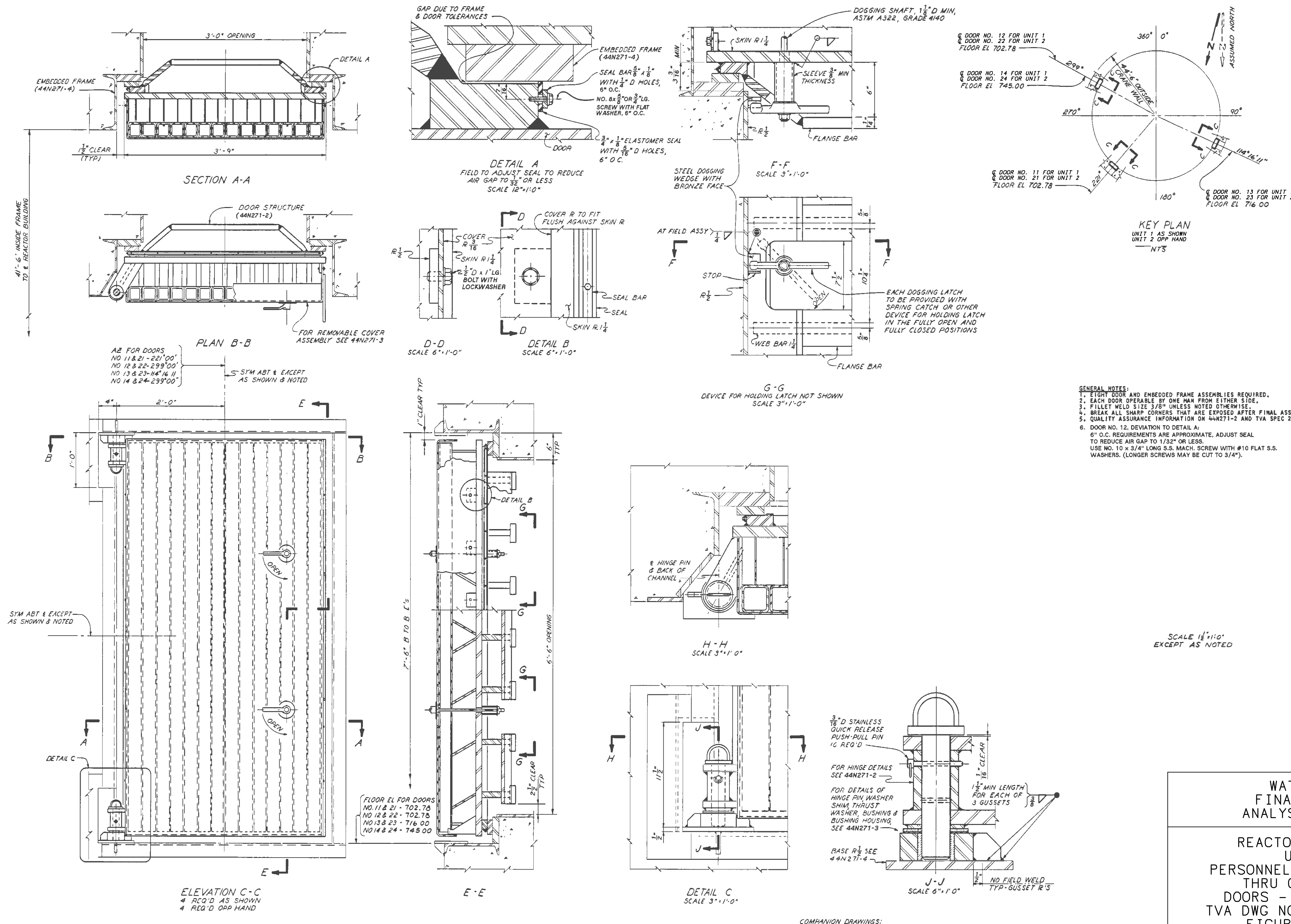
FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 48W933

COMPANION DRAWINGS:
48W930, 48N931, 48W932

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BLDG
UNIT 2
MISCELLANEOUS STEEL
REACTOR CAVITY
EMBEDDED PARTS SH 4
TVA DWG NO. 2-48W933 R0
FIGURE 3.8.3-7G(U2)

CAD MAINTAINED DRAWING

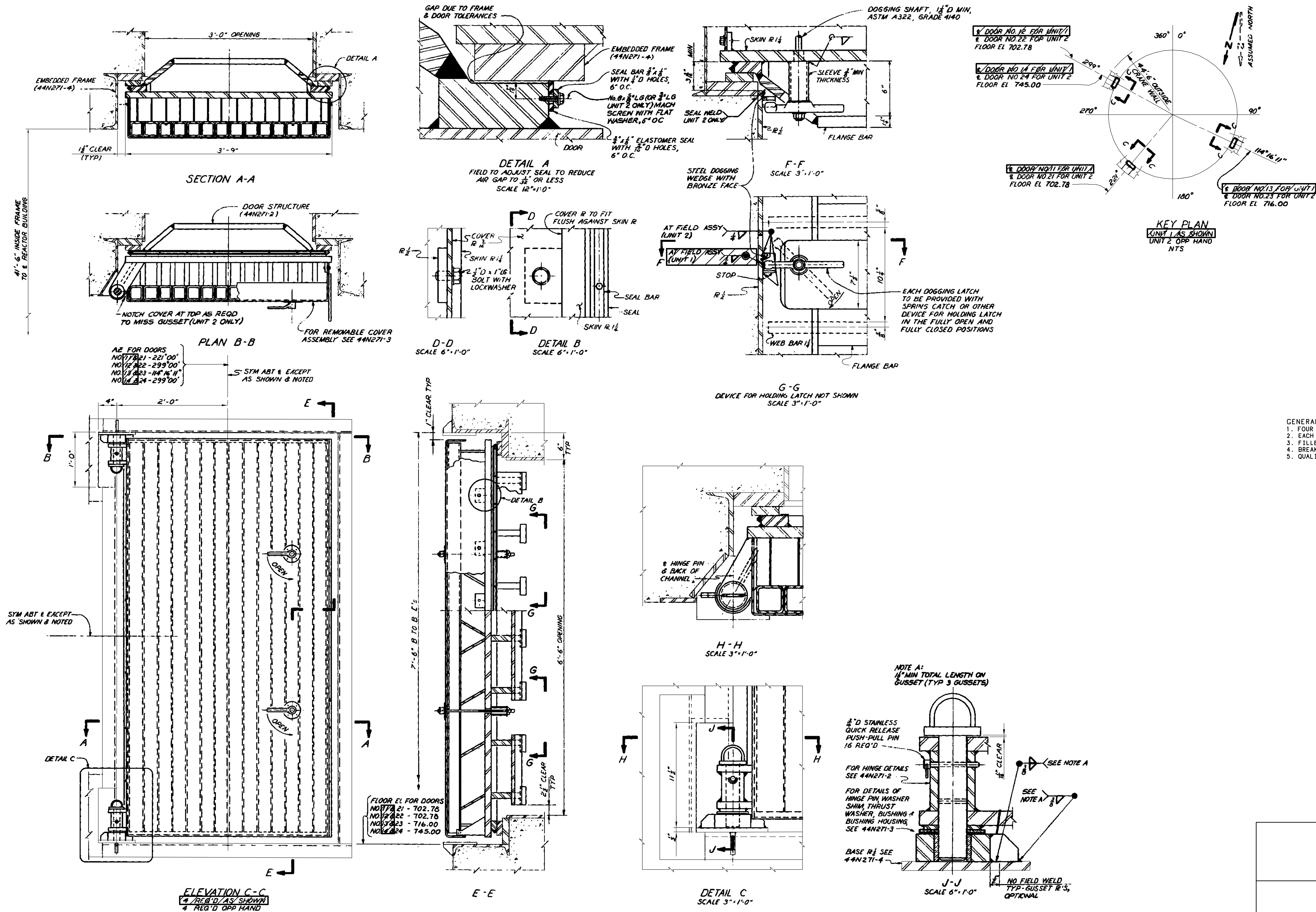


GENERAL NOTES:
1. EIGHT DOOR AND EMBEDDED FRAME ASSEMBLIES REQUIRED.
2. EACH DOOR OPERABLE BY ONE MAN FROM EITHER SIDE.
3. FILLET WELD SIZE 3/8" UNLESS NOTED OTHERWISE.
4. BREAK ALL SHARP CORNERS THAT ARE EXPOSED AFTER FINAL ASSEMBLY.
5. QUALITY ASSURANCE INFORMATION ON 44N271-2 AND TVA SPEC 2284.
6. DOOR NO. 12, DEVIATION TO DETAIL A:
   6" O.C. REQUIREMENTS ARE APPROXIMATE, ADJUST SEAL TO REDUCE AIR GAP TO 1/32" OR LESS.
   USE NO. 10 x 3/4" LONG S.S. MACH. SCREW WITH #10 FLAT S.S. WASHERS. (LONGER SCREWS MAY BE CUT TO 3/4").

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
PERSONNEL ACCESS DOORS
THRU CRANE WALL
DOORS - ARRANGEMENT
TVA DWG NO. 44N271-1 RC
FIGURE 3.8.3-8

CAD MAINTAINED DRAWING



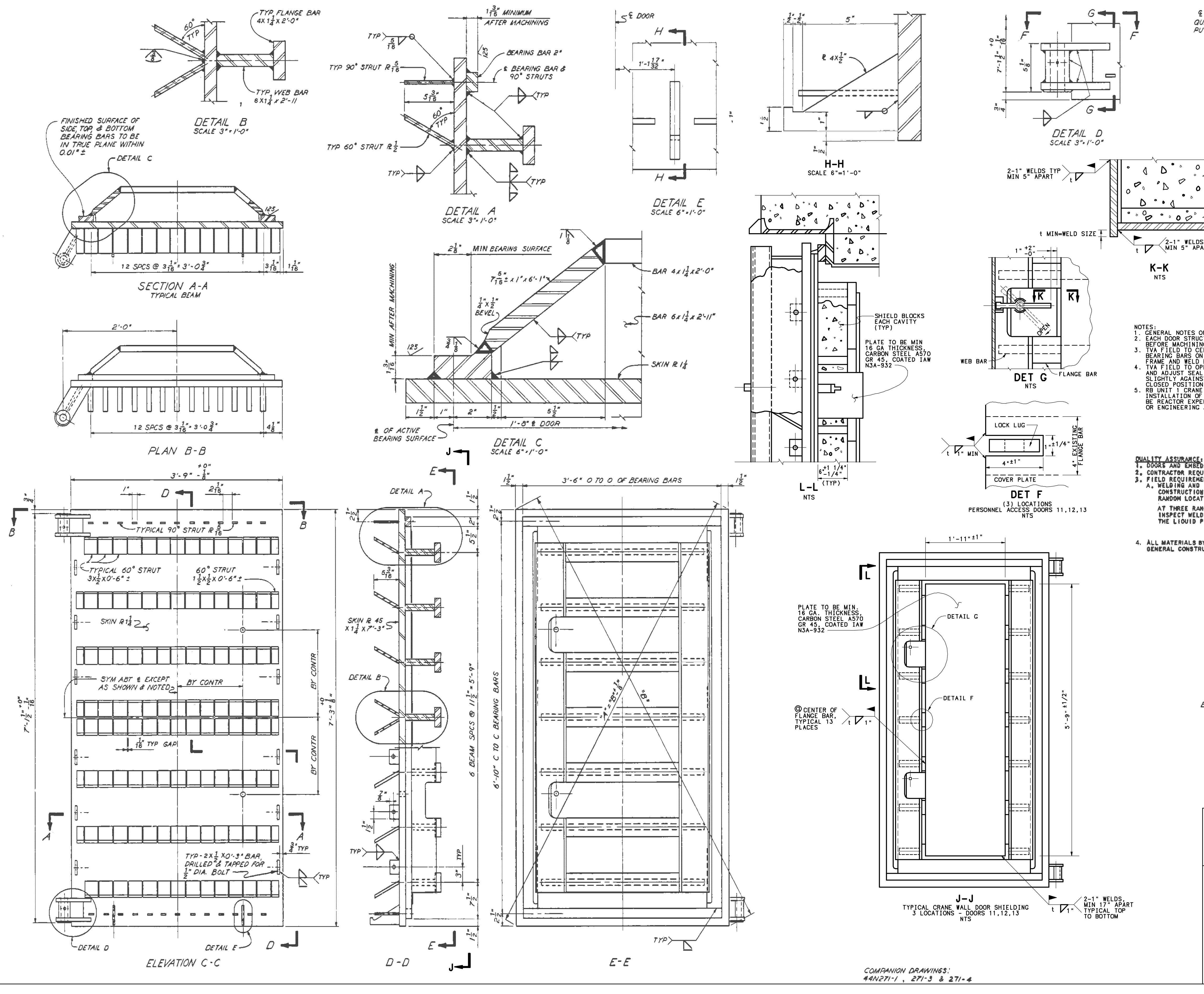
GENERAL NOTES:
1. FOUR DOOR AND EMBEDDED FRAME ASSEMBLIES REQUIRED.
2. EACH DOOR OPERABLE BY ONE MAN FROM EITHER SIDE.
3. FILLET WELD SIZE 3/8" UNLESS NOTED OTEHRWISE.
4. BREAK ALL SHARP CORNERS THAT ARE EXPOSED AFTER FINAL ASSEMBLY.
5. QUALITY ASSURANCE INFORMATION ON 44N271-2 AND TVA SPEC 2284.

FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 44N271-1

SCALE 1-1/2" = 1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
PERSONNEL ACCESS DOORS
THRU CRANE WALL
DOORS - ARRANGEMENT
TVA DWG NO. 2-44N271-1 R0
FIGURE 3.8.3-8(U2)



NOTES:
1. GENERAL NOTES ON 44N271-1.
2. EACH DOOR STRUCTURE STRESS RELIEVED AFTER SHOP WELDING AND BEFORE MACHINING IN ACCORDANCE WITH TVA SPEC 2284.
3. TVA FIELD TO CENTER EACH DOOR IN EMBEDDED FRAME OPENING WITH BEARING BARS ON DOOR IN CONTACT WITH BEARING BARS ON EMBEDDED FRAME AND WELD BUSHING HOUSINGS TO BASE PLATES.
4. TVA FIELD TO OPERATE DOOR THROUGH FULL TRAVEL SEVERAL TIMES AND ADJUST SEAL ON DOOR SO THAT SEAL CONTACTS OR COMPRESSES SLIGHTLY AGAINST THE EMBEDDED FRAME WHEN DOOR IS DOGGED IN THE CLOSED POSITION.
5. RB UNIT 1 CRANE WALL DOORS 11, 12 & 13 MODIFIED FOR INSTALLATION OF SHIELDING MATERIAL. SHIELDING MATERIAL SHALL BE REACTOR EXPERIMENTS, INC. CATALOG #277 SHIELD MATERIAL OR ENGINEERING APPROVED EQUAL TO FILL CAVITIES IN DOORS.

QUALITY ASSURANCE:
1. DOORS AND EMBEDDED FRAMES ARE CATEGORY 1 EQUIPMENT.
2. CONTRACTOR REQUIREMENTS IN ACCORDANCE WITH TVA SPEC 2284.
3. FIELD REQUIREMENTS:
   A. WELDING AND INSPECTION OF WELDS IN ACCORDANCE WITH GENERAL CONSTRUCTION SPEC G-29C, EXCEPT TEST 10 PERCENT OF STUDS AT RANDOM LOCATIONS, BEND TO 30°.

      AT THREE RANDOM LOCATIONS ON EACH SIDE OF EMBEDDED FRAME, INSPECT WELDS OF MK 44N271-4-11 TO MK 44N271-4-4 BY EITHER THE LIQUID PENETRANT OR MAGNETIC PARTICLE METHODS.

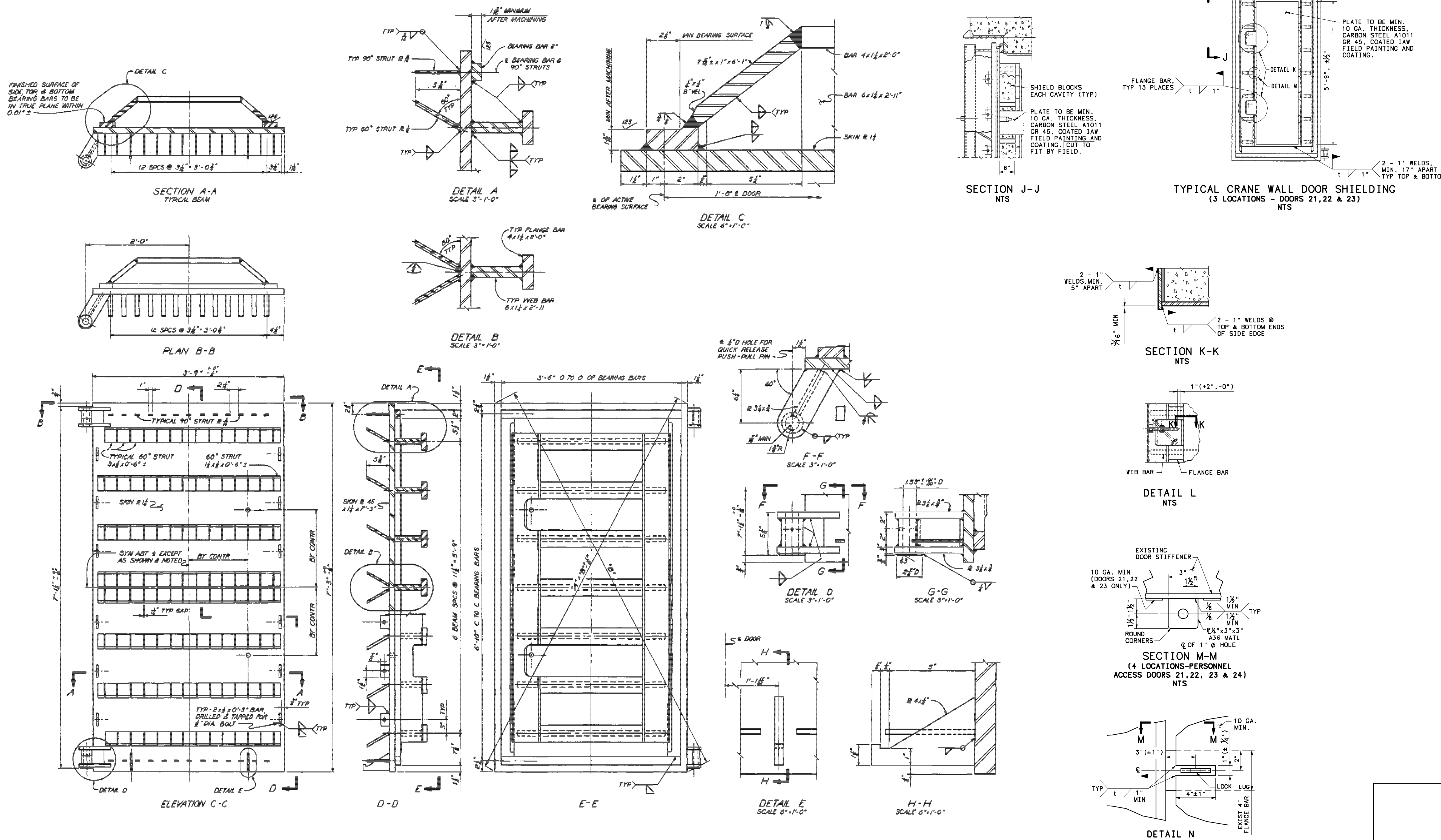
4. ALL MATERIALS BY TVA FIELD & FIELD WORK SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

SCALE $1\frac{1}{2}$" = 1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
PERSONNEL ACCESS DOORS
THRU CRANE WALL DOORS
STRUCTURAL DETAILS
TVA DWG NO. 44N271-2 RC
FIGURE 3.8.3-9

CAD MAINTAINED DRAWING



NOTES:
1. GENERAL NOTES ON 2-44N271-1.
2. EACH DOOR STRUCTURE STRESS RELIEVED AFTER SHOP WELDING AND BEFORE MACHINING IN ACCORDANCE WITH TVA SPEC 2284.
3. TVA FIELD TO CENTER EACH DOOR IN EMBEDDED FRAME OPENING WITH BEARING BARS ON DOOR IN CONTACT WITH BEARING BARS ON EMBEDDED FRAME AND WELD BUSHING HOUSING TO BASE PLATES.
4. TVA FIELD TO OPERATE DOOR THROUGH FULL TRAVEL SEVERAL TIMES AND ADJUST SEAL ON DOOR SO THAT SEAL CONTACTS OR COMPRESSES SLIGHTLY AGAINST THE EMBEDDED FRAME WHEN DOOR IS DOGGED IN THE CLOSED POSITION.
5. RB UNIT 2 CRANE WALL DOORS MODIFIED FOR INSTALLATION OF SHIELDING MATERIAL. SHIELDING MATERIAL SHALL BE SHIELWERX CATALOG # SWX-277 MATERIAL OR ENGINEERING APPROVED EQUAL TO FILL CAVITIES IN DOORS.
6. SHIELD MATERIAL SHALL BE MIXED & FORMED IN ACCORDANCE WITH MANUFACTURERS' RECOMMENDATIONS. SHIELD MATERIAL SHALL BE CAST, TRIMMED TO FIT & INSERTED AS HARDENED BLOCKS INSIDE CAVITIES IN DOORS AS SHOWN IN TYPICAL SECTIONS. TO ENSURE SNUG FIT-UP WITHIN THE DOOR CAVITIES , FIELD MAY CAST CENTER BLOCKS @ EACH DOOR CAVITY LOCATION UP TO 7¼" IN THICKNESS VERSUS 6".
7. FOR WELDABILITY PURPOSES, EXISTING DOOR MATERIAL IS A-36.
8. NOTCH COVER PLATES AS REQUIRED TO ALLOW INSTALLATION OVER DOOR LOCK LUG.

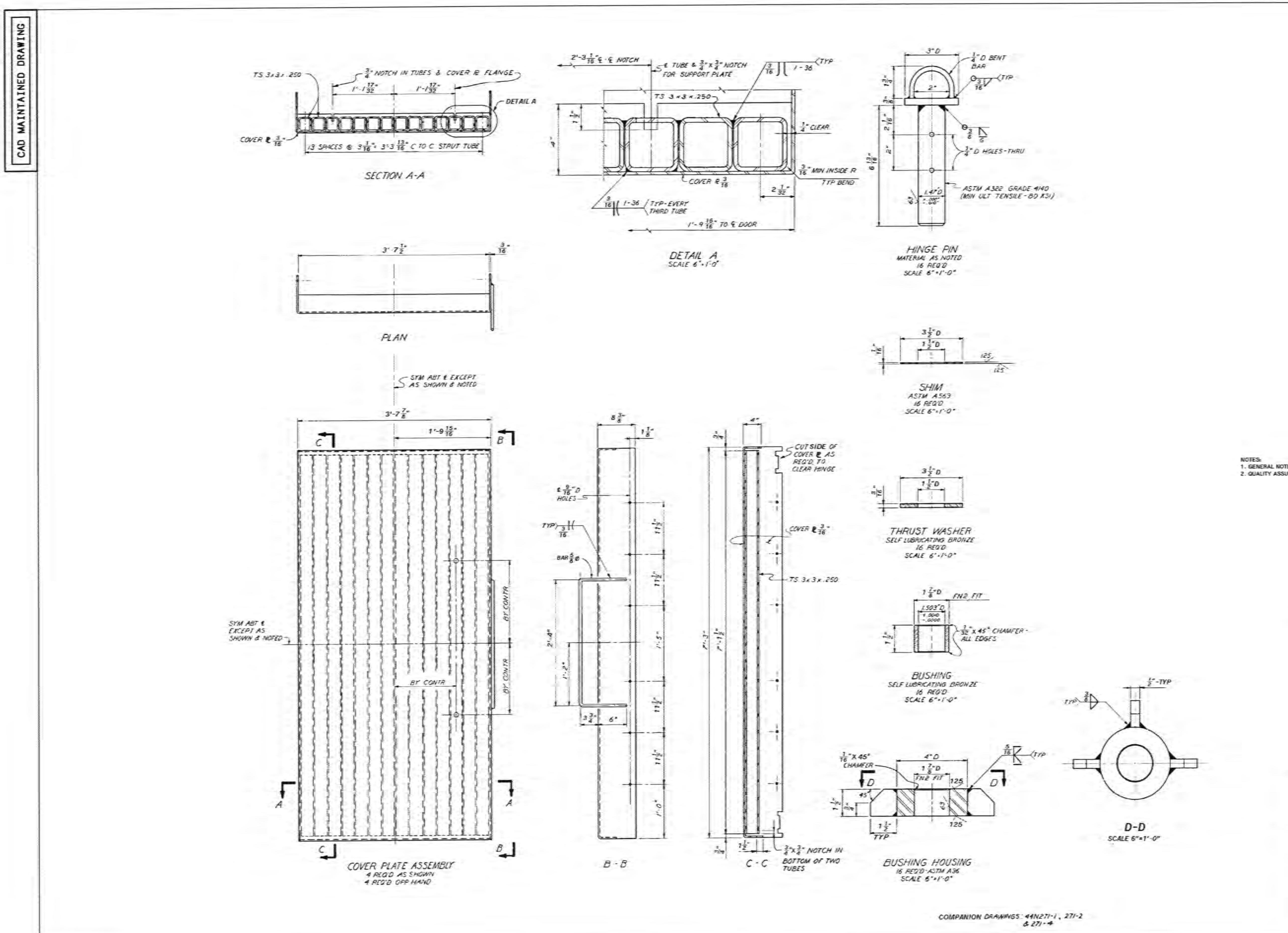
QUALITY ASSURANCE:
1. DOORS AND EMBEDDED FRAMES ARE CATEGORY 1 EQUIPMENT.
2. CONTRACTOR REQUIREMENTS IN ACCORDANCE WITH TVA SPEC 2284.
3. FIELD REQUIREMENTS:
   A. WELDING AND INSPECTION OF WELDS IN ACCORDANCE WITH GENERAL CONSTRUCTION SPEC G-29C EXCEPT TEST 10 PERCENT OF STUDS AT RANDOM LOCATIONS, BEND TO 30 DEGREES. AT THREE RANDOM LOCATIONS ON EACH SIDE OF EMBEDDED FRAME, INSPECT WELDS OF MK 44N271-4-11 TO MK 44N271-4-4 BY EITHER THE LIQUID PENETRANT OR MAGNETIC PARTICLE METHODS.
4. ALL MATERIALS BY TVA FIELD & FIELD WORK SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

SCALE 1-1/2" = 1'-0"
EXCEPT AS NOTED

COMPANION DWGS:
2-44N271-1, 271-3, & 271-4

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
PERSONNEL ACCESS DOORS
THRU CRANE WALL DOORS
STRUCTURAL DETAILS
TVA DWG NO. 2-44N271-2 R2
FIGURE 3.8.3-9(U2)



NOTES:
1. GENERAL NOTES ON 44N271-1.
2. QUALITY ASSURANCE INFORMATION ON 44N271-2 TVA SPEC 2284.

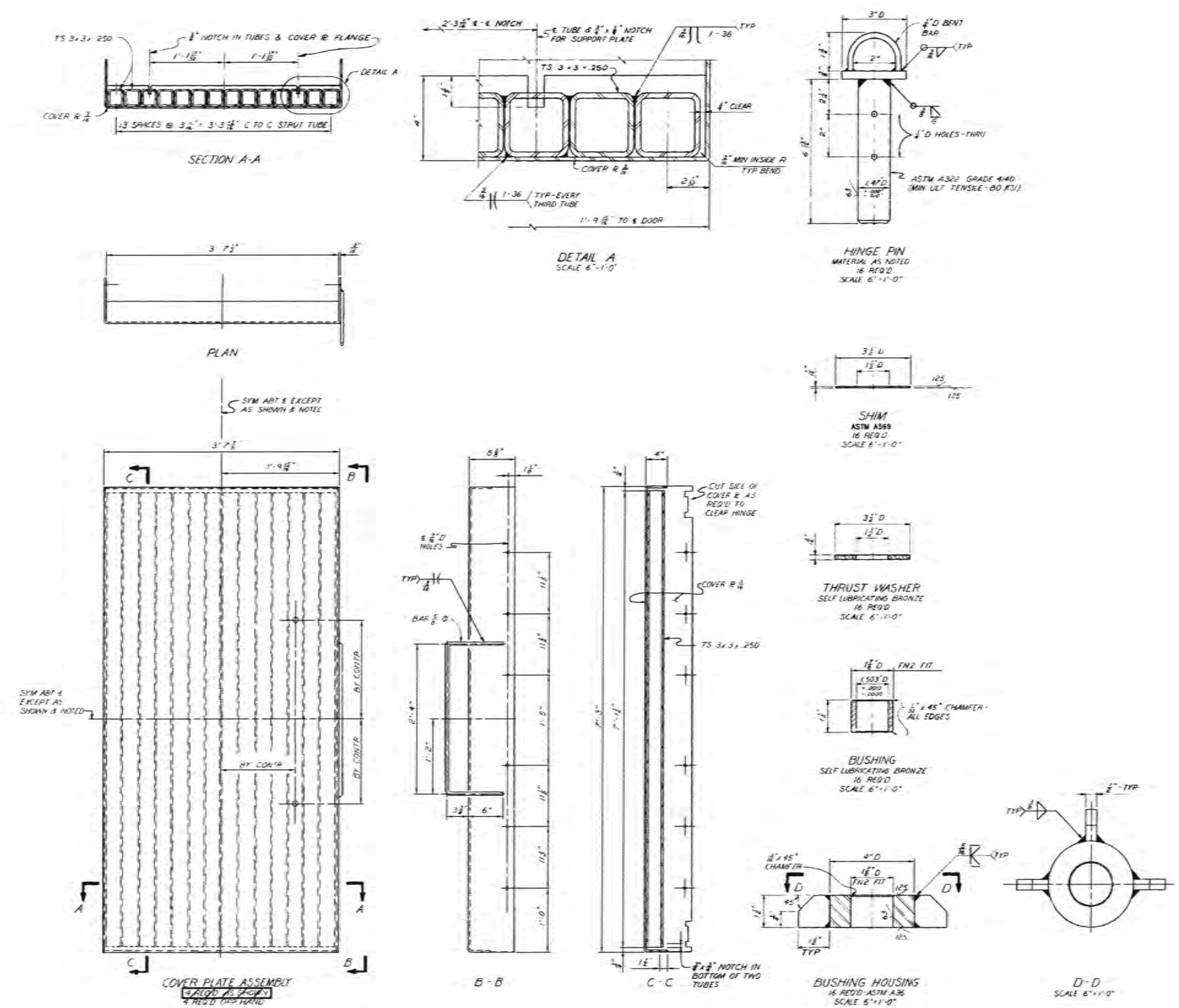
SCALE 1½"=1'-0" EXCEPT AS NOTED

COMPANION DRAWINGS: 44N271-1, 271-2 & 271-4

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
PERSONNEL ACCESS DOORS
THRU CRANE WALL DOORS
MISCELLANEOUS DETAILS
TVA DWG NO. 44N271-3 RB
FIGURE 3.8.3-10

CAD MAINTAINED DRAWING



NOTES:
1. GENERAL NOTES ON 44N271-1.
2. QUALITY ASSURANCE INFORMATION ON 44N271-2 AND TVA SPEC 2284.

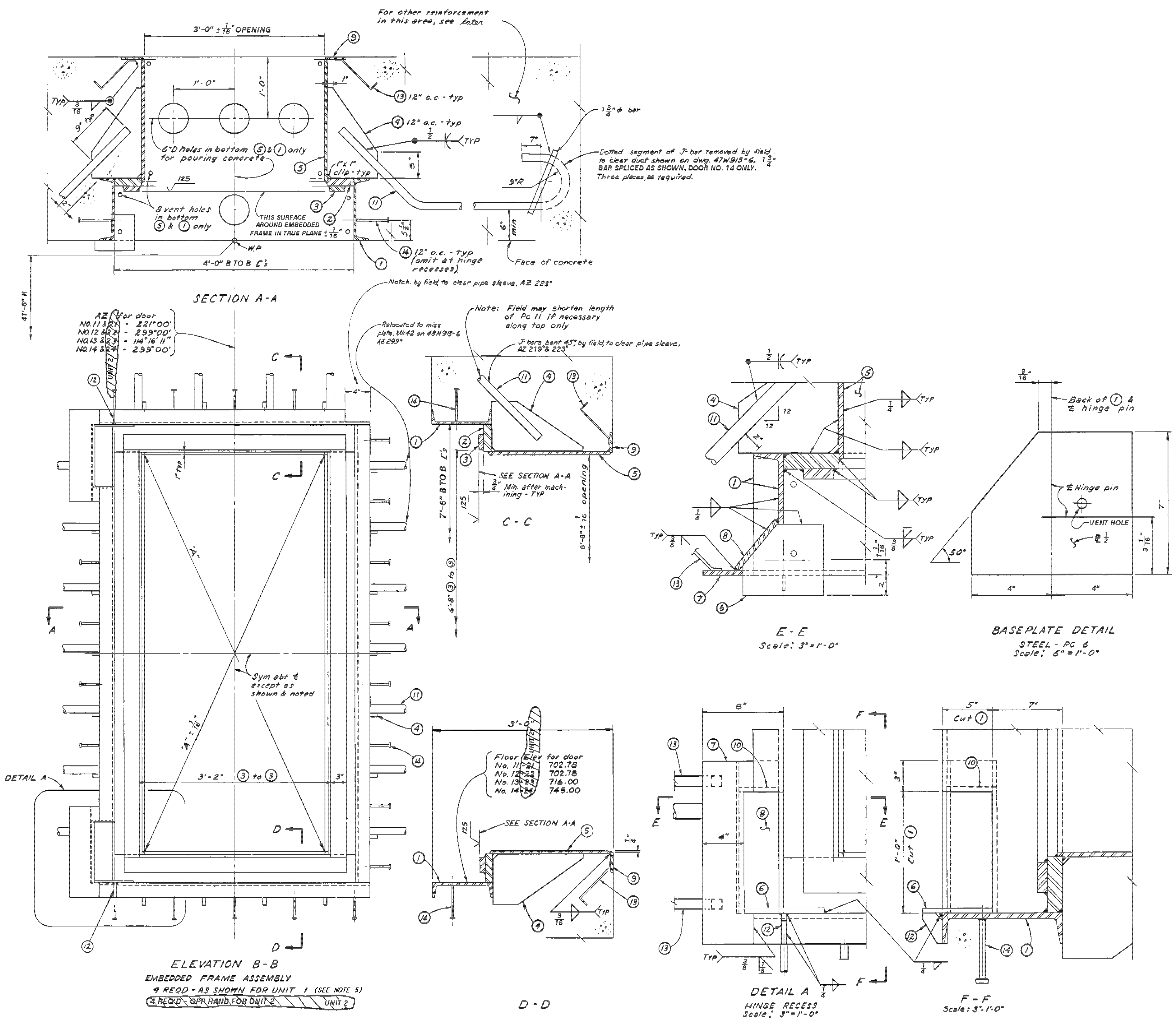
SCALE 1½"=1'-0"
EXCEPT AS NOTED

FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 44N271-3

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
PERSONNEL ACCESS DOORS
THRU CRANE WALL DOORS
MISCELLANEOUS DETAILS
TVA DWG NO. 2-44N271-3 R0
FIGURE 3.8.3-10(U2)

CAD MAINTAINED DRAWING



NOTES:
1. GENERAL NOTES ON 44N271-1.
2. CIRCLED NUMBERS ARE ITEM NUMBERS. SEE BILL OF MATERIAL BELOW.
3. EACH EMBEDDED FRAME TO BE STRESS RELIEVED AFTER SHOP WELDING AND BEFORE MACHINING IN ACCORDANCE WITH TVA SPEC 2284.
4. QUALITY ASSURANCE INFORMATION ON 44N271-2 AND TVA SPEC 2284.
5. ADDITIONAL STRUCTURAL STEEL IS ADDED TO DOOR NUMBERS 11 & 12 FOR LEAD SHIELDING ATTACHMENT. SEE DRAWING NO. 1-44N271-5 FOR DETAILS.

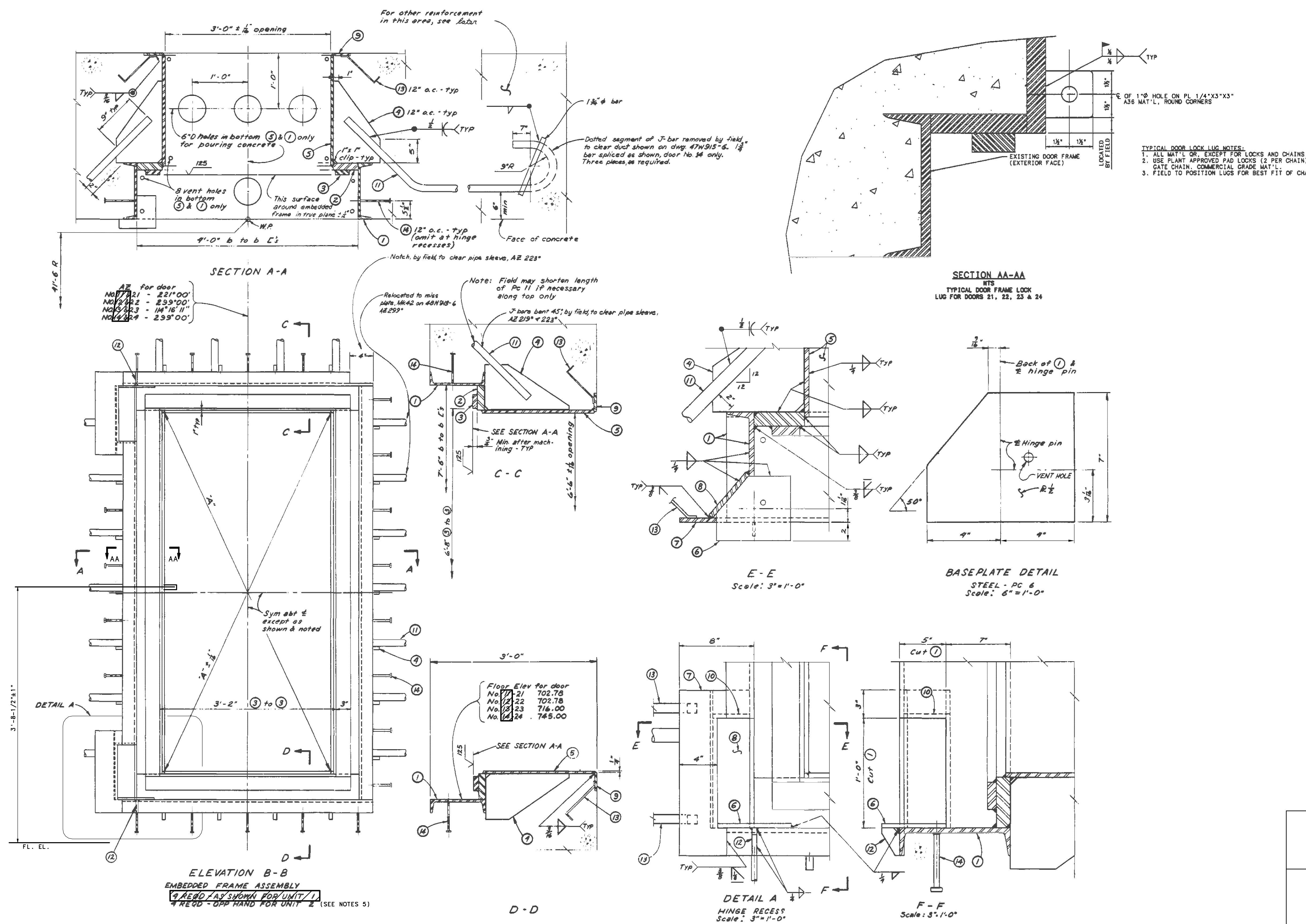
BILL OF MATERIAL

ITEM NO.	DESCRIPTION
1	12 CHANNEL 30.0
2	PLATE 5-1/2" X 1-1/2"
3	PLATE 3" X 3/4" MIN (AFTER MACHINING)
4	PLATE 10" X 3/4" X 1'-6" (CUT)
5	PLATE 24" X 1/2"
6	PLATE 7" X 1/2" X 0'-8" (CUT)
7	PLATE 5" X 1/2" X 1'-6" (CUT)
8	PLATE 6" X 1/2" X 1'-3"
9	PLATE 4" X 1/2"
10	PLATE 4"± X 1/2" X 0'-4"± (CUT)
11	1-3/4" D X 6'-8" BENT ROD
12	PLATE 2" X 1/2" X 0'-3"
	THESE ITEMS SUPPLIED BY TVA FIELD
13	1" X 1/4" X 0'-10" STRAP ANCHOR, STEEL. (208 REQD TOTAL FOR 8 EMBEDDED FRAMES)
14	5/8" D X 6-3/8" HEADED CONCRETE ANCHOR STUDS, ASTM A 108. (176 REQD TOTAL FOR 8 EMBEDDED FRAMES)

SCALE 1 1/2" = 1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
PERSONNEL ACCESS DOORS
THRU CRANE WALL
EMBEDDED FRAMES
TVA DWG NO. 44N271-4 RC
FIGURE 3.8.3-11



NOTES:

1. GENERAL NOTES ON 44N271-1.
2. CIRCLED NUMBERS ARE ITEM NUMBERS. SEE BILL OF MATERIAL BELOW.
3. EACH EMBEDDED FRAME TO BE STRESS RELIEVED AFTER SHOP WELDING AND BEFORE MACHINING IN ACCORDANCE WITH TVA SPEC 2284.
4. QUALITY ASSURANCE INFORMATION ON 44N271-2 AND TVA SPEC 2284.
5. ADDITIONAL STRUCTURAL STEEL IS ADDED TO DOOR NUMBERS 21 & 22 FOR LEAD SHIELDING ATTACHMENT. SEE DRAWING NO. 2-44N271-5-1 FOR DETAILS.

BILL OF MATERIAL

ITEM NO.	DESCRIPTION
1	12 CHANNEL 30.0
2	PLATE 5-1/2" X 1-1/2"
3	PLATE 3" X 3/4" MIN (AFTER MACHINING)
4	PLATE 10" X 3/4" X 1'-6" (CUT)
5	PLATE 24" X 1/2"
6	PLATE 7" X 1/2" X 0'-8" (CUT)
7	PLATE 5" X 1/2" X 1'-6" (CUT)
8	PLATE 6" X 1/2" X 1'-3"
9	PLATE 4" X 1/2"
10	PLATE 4"± X 1/2" X 0'-4"± (CUT)
11	1-3/4" D X 6'-8" BENT ROD
12	PLATE 2" X 1/2" X 0'-3"
	THESE ITEMS SUPPLIED BY TVA FIELD
13	1" X 1/4" X 0'-10" STRAP ANCHOR, STEEL. (208 REQD TOTAL FOR 8 EMBEDDED FRAMES)
14	5/8" D X 6-3/8" HEADED CONCRETE ANCHOR STUDS, ASTM A 108. (176 REQD TOTAL FOR 8 EMBEDDED FRAMES)

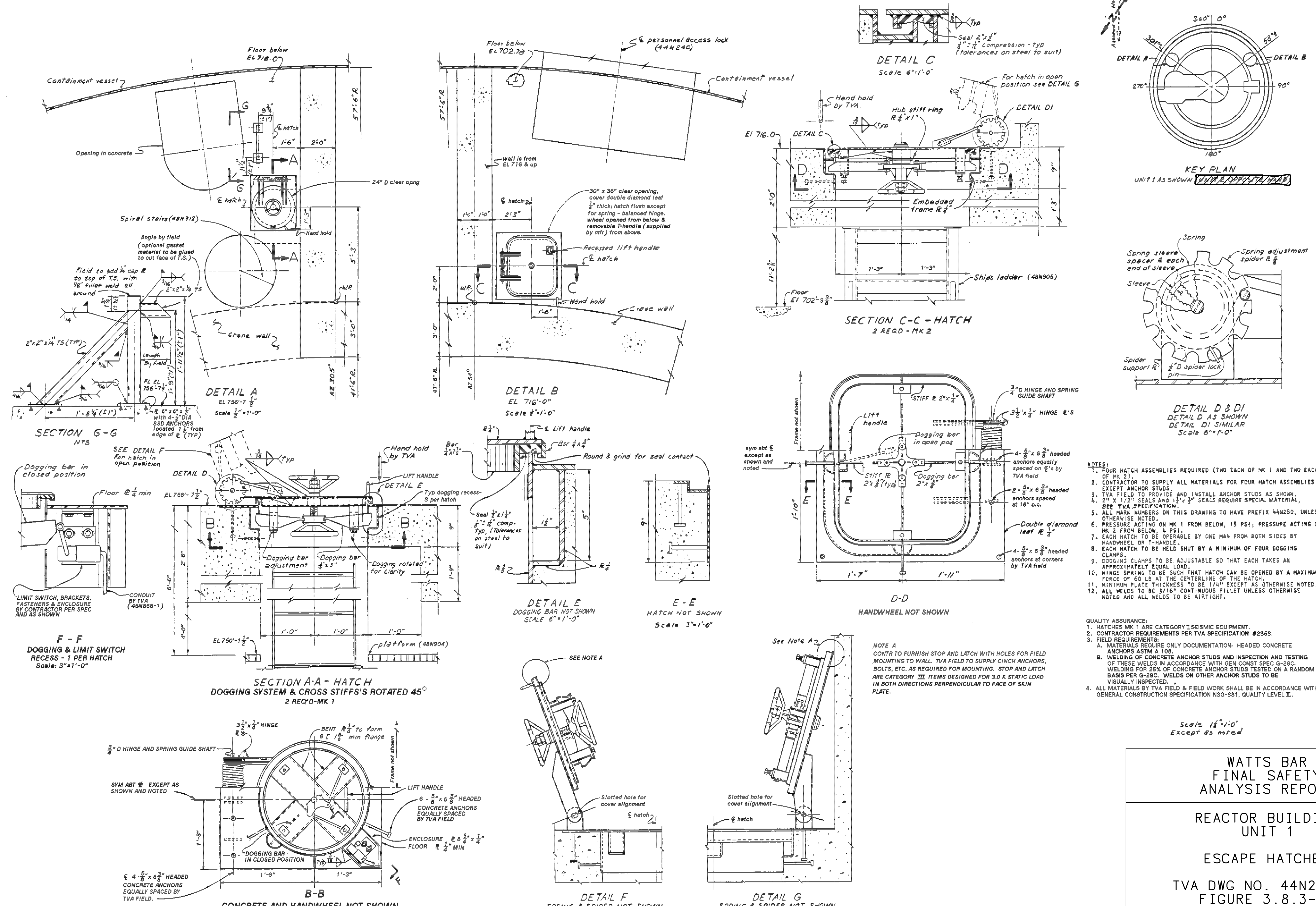
FOR HATCHED AREAS SEE UNIT 1 AC DRAWING 44N271-4

SCALE 1-1/2" = 1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
PERSONNEL ACCESS DOORS
THRU CRANE WALL
EMBEDDED FRAMES
TVA DWG NO. 2-44N271-4 R2
FIGURE 3.8.3-11(U2)

CAD MAINTAINED DRAWING



NOTES:

1. FOUR HATCH ASSEMBLIES REQUIRED (TWO EACH OF MK 1 AND TWO EACH OF MK 2).
2. CONTRACTOR TO SUPPLY ALL MATERIALS FOR FOUR HATCH ASSEMBLIES EXCEPT ANCHOR STUDS.
3. TVA FIELD TO PROVIDE AND INSTALL ANCHOR STUDS AS SHOWN.
4. 2" X 1/2" SEALS AND 1 1/4" x 1/2" SEALS REQUIRE SPECIAL MATERIAL, SEE TVA SPECIFICATION.
5. ALL MARK NUMBERS ON THIS DRAWING TO HAVE PREFIX 44N280, UNLESS OTHERWISE NOTED.
6. PRESSURE ACTING ON MK 1 FROM BELOW, 15 PSI; PRESSURE ACTING ON MK 2 FROM BELOW, 4 PSI.
7. EACH HATCH TO BE OPERABLE BY ONE MAN FROM BOTH SIDES BY HANDWHEEL OR T-HANDLE.
8. EACH HATCH TO BE HELD SHUT BY A MINIMUM OF FOUR DOGGING CLAMPS.
9. DOGGING CLAMPS TO BE ADJUSTABLE SO THAT EACH TAKES AN APPROXIMATELY EQUAL LOAD.
10. HINGE SPRING TO BE SUCH THAT HATCH CAN BE OPENED BY A MAXIMUM FORCE OF 60 LB AT THE CENTERLINE OF THE HATCH.
11. MINIMUM PLATE THICKNESS TO BE 1/4" EXCEPT AS OTHERWISE NOTED.
12. ALL WELDS TO BE 3/16" CONTINUOUS FILLET UNLESS OTHERWISE NOTED AND ALL WELDS TO BE AIRTIGHT.

QUALITY ASSURANCE:

1. HATCHES MK 1 ARE CATEGORY I SEISMIC EQUIPMENT.
2. CONTRACTOR REQUIREMENTS PER TVA SPECIFICATION #2353.
3. FIELD REQUIREMENTS:
   A. MATERIALS REQUIRE ONLY DOCUMENTATION: HEADED CONCRETE ANCHORS ASTM A 108.
   B. WELDING OF CONCRETE ANCHOR STUDS AND INSPECTION AND TESTING OF THESE WELDS IN ACCORDANCE WITH GEN CONST SPEC G-29C. WELDING FOR 25% OF CONCRETE ANCHOR STUDS TESTED ON A RANDOM BASIS PER G-29C. WELDS ON OTHER ANCHOR STUDS TO BE VISUALLY INSPECTED.
4. ALL MATERIALS BY TVA FIELD & FIELD WORK SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

Scale 1 1/2"=1'-0"
Except as noted

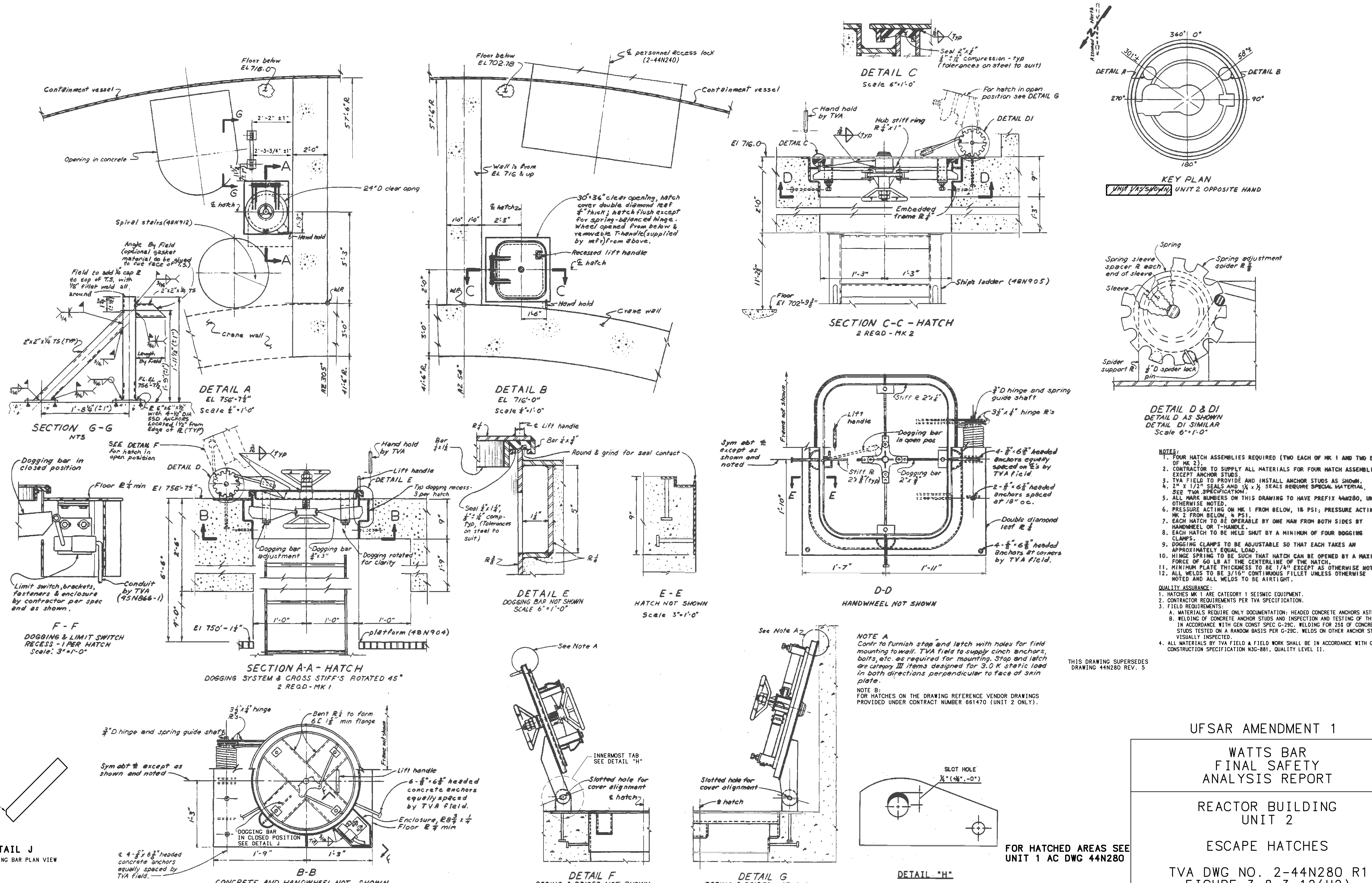
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1

ESCAPE HATCHES

TVA DWG NO. 44N280 RC
FIGURE 3.8.3-12

CAD MAINTAINED DRAWING



NOTES:
1. FOUR HATCH ASSEMBLIES REQUIRED (TWO EACH OF MK 1 AND TWO EACH OF MK 2).
2. CONTRACTOR TO SUPPLY ALL MATERIALS FOR FOUR HATCH ASSEMBLIES EXCEPT ANCHOR STUDS.
3. TVA FIELD TO PROVIDE AND INSTALL ANCHOR STUDS AS SHOWN.
4. 2" X 1/2" SEALS AND 1¼ x ½ SEALS REQUIRE SPECIAL MATERIAL, SEE TVA SPECIFICATION.
5. ALL MARK NUMBERS ON THIS DRAWING TO HAVE PREFIX 44N280, UNLESS OTHERWISE NOTED.
6. PRESSURE ACTING ON MK 1 FROM BELOW, 15 PSI; PRESSURE ACTING ON MK 2 FROM BELOW, 4 PSI.
7. EACH HATCH TO BE OPERABLE BY ONE MAN FROM BOTH SIDES BY HANDWHEEL OR T-HANDLE.
8. EACH HATCH TO BE HELD SHUT BY A MINIMUM OF FOUR DOGGING CLAMPS.
9. DOGGING CLAMPS TO BE ADJUSTABLE SO THAT EACH TAKES AN APPROXIMATELY EQUAL LOAD.
10. HINGE SPRING TO BE SUCH THAT HATCH CAN BE OPENED BY A MAXIMUM FORCE OF 60 LB AT THE CENTERLINE OF THE HATCH.
11. MINIMUM PLATE THICKNESS TO BE 1/4" EXCEPT AS OTHERWISE NOTED.
12. ALL WELDS TO BE 3/16" CONTINUOUS FILLET UNLESS OTHERWISE NOTED AND ALL WELDS TO BE AIRTIGHT.

QUALITY ASSURANCE:
1. HATCHES MK 1 ARE CATEGORY 1 SEISMIC EQUIPMENT.
2. CONTRACTOR REQUIREMENTS PER TVA SPECIFICATION.
3. FIELD REQUIREMENTS:
   A. MATERIALS REQUIRE ONLY DOCUMENTATION: HEADED CONCRETE ANCHORS ASTM A108.
   B. WELDING OF CONCRETE ANCHOR STUDS AND INSPECTION AND TESTING OF THESE WELDS IN ACCORDANCE WITH GEN CONST SPEC G-29C. WELDING FOR 25% OF CONCRETE ANCHOR STUDS TESTED ON A RANDOM BASIS PER G-29C. WELDS ON OTHER ANCHOR STUDS TO BE VISUALLY INSPECTED.
4. ALL MATERIALS BY TVA FIELD & FIELD WORK SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

NOTE A
Contr to furnish stop and latch with holes for field mounting to wall. TVA field to supply cinch anchors, bolts, etc. as required for mounting. Stop and latch are category III items designed for 3.0 K static load in both directions perpendicular to face of skin plate.

NOTE B:
FOR HATCHES ON THE DRAWING REFERENCE VENDOR DRAWINGS PROVIDED UNDER CONTRACT NUMBER 661470 (UNIT 2 ONLY).

THIS DRAWING SUPERSEDES DRAWING 44N280 REV. 5

FOR HATCHED AREAS SEE UNIT 1 AC DWG 44N280

UFSAR AMENDMENT 1

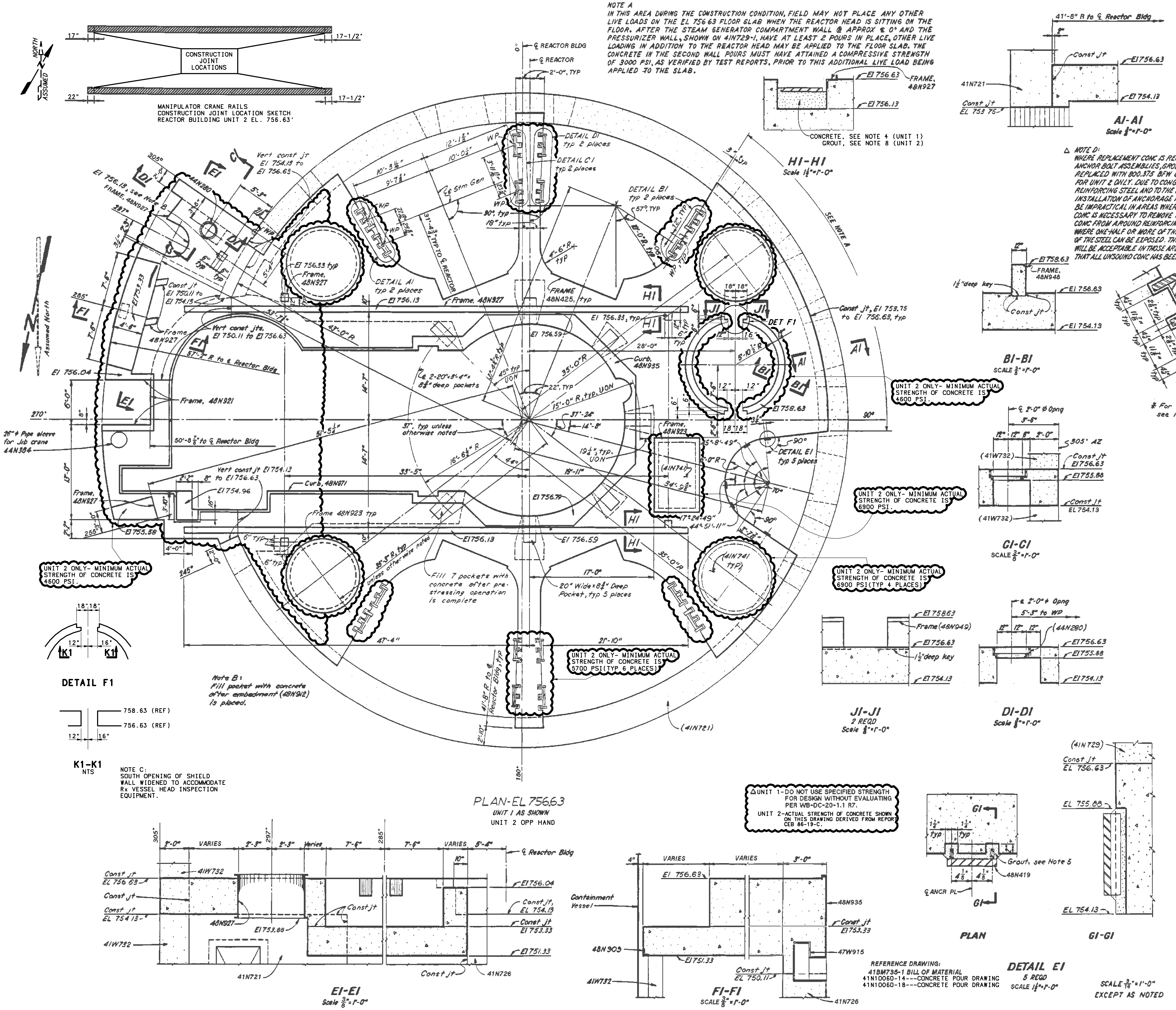
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2

ESCAPE HATCHES

TVA DWG NO. 2-44N280 R1
FIGURE 3.8.3-12(U2)

CAD MAINTAINED DRAWING



NOTE A
IN THIS AREA DURING THE CONSTRUCTION CONDITION, FIELD MAY NOT PLACE ANY OTHER LIVE LOADS ON THE EL 756.63 FLOOR SLAB WHEN THE REACTOR HEAD IS SITTING ON THE FLOOR. AFTER THE STEAM GENERATOR COMPARTMENT WALL @ APPROX ℄ 0° AND THE PRESSURIZER WALL, SHOWN ON 41N729-1, HAVE AT LEAST 2 POURS IN PLACE, OTHER LIVE LOADING IN ADDITION TO THE REACTOR HEAD MAY BE APPLIED TO THE FLOOR SLAB. THE CONCRETE IN THE SECOND WALL POURS MUST HAVE ATTAINED A COMPRESSIVE STRENGTH OF 3000 PSI, AS VERIFIED BY TEST REPORTS, PRIOR TO THIS ADDITIONAL LIVE LOAD BEING APPLIED TO THE SLAB.

CONCRETE. SEE NOTE 4 (UNIT 1)
GROUT. SEE NOTE 8 (UNIT 2)

△ NOTE D:
WHERE REPLACEMENT CONC IS REQD AROUND ANCHOR BOLT ASSEMBLIES, GROUT MAY BE REPLACED WITH 800.375 BFW CONC MIX FOR UNIT 2 ONLY. DUE TO CONGESTION OF REINFORCING STEEL AND TO THE SIZE AND INSTALLATION OF ANCHORAGE PL, IT MAY BE IMPRACTICAL IN AREAS WHERE REPAIR OF CONC IS NECESSARY TO REMOVE ADDITIONAL CONC FROM AROUND REINFORCING STEEL TO WHERE ONE-HALF OR MORE OF THE PERIMETER OF THE STEEL CAN BE EXPOSED. THIS SITUATION WILL BE ACCEPTABLE IN THOSE AREAS PROVIDED THAT ALL UNSOUND CONC HAS BEEN REMOVED.

#IN LIEU OF GROUT FOR UNIT 2, SEE NOTE D.

NOTE C:
SOUTH OPENING OF SHIELD WALL WIDENED TO ACCOMMODATE Rx VESSEL HEAD INSPECTION EQUIPMENT.

NOTES:
1. The Reactor Building is a Class 1 structure, and quality assurance is required.
2. For general notes, see dwg 41N716-1.
3. For embedded parts, pipes, and conduits, see Structural Steel, Mechanical, and Electrical drawings.
△ 4. Concrete shall be Class 500.75A FW. In addition to normal requirements, compressive strength tests of this concrete are required at 90 & 180 days to verify ultimate design strength in excess of 6,750 psi. The floor shall reach 2,000 psi compressive strength and shall be in place a minimum of 7 days before shoring is removed.
△ 5. Grout shall attain a minimum compressive strength of 8000 psi in 90 days. A trial mix shall be tested in accordance with Corps of Engineers test method CRD-C88 to assure the required strength. "Five Star" non-shrink non-metallic grout by U.S. Grout or approved equal may be used in accordance with the manufacturers recommendations.
6. WHERE CONCRETE CANNOT BE PROPERLY COMPACTED IN FORMS DUE TO HIGH CONCENTRATIONS OF EMBEDMENTS, THE PROJECT MATERIALS ENGINEER MAY SUBSTITUTE A CONCRETE MIX OF EQUAL OR GREATER STRENGTH AND WITH SMALLER SIZE COARSE AGGREGATE. THIS ALTERNATE MIX SHALL BE APPROVED BY DED.
7. SEE CALCULATION WCG-1-1370 FOR ENGINEERING REVIEW AND ASSESSMENT OF THE REACTOR BUILDING CONCRETE FEATURES.
8. GROUT SHALL BE TESTED AND COMPLY WITH MAI-5.4 SECTION 6.18.1, ASTM C-942 AND GENERAL SPECIFICATION G-51. A MINIMUM COMPRESSIVE STRENGTH SHALL BE 8000 PSI IN 90 DAYS.

△UNIT 1-DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.
UNIT 2-ACTUAL STRENGTH OF CONCRETE SHOWN ON THIS DRAWING DERIVED FROM REPORT CEB 86-19-C.

REFERENCE DRAWING:
41BM735-1 BILL OF MATERIAL
41N1060-14---CONCRETE POUR DRAWING
41N1060-18---CONCRETE POUR DRAWING

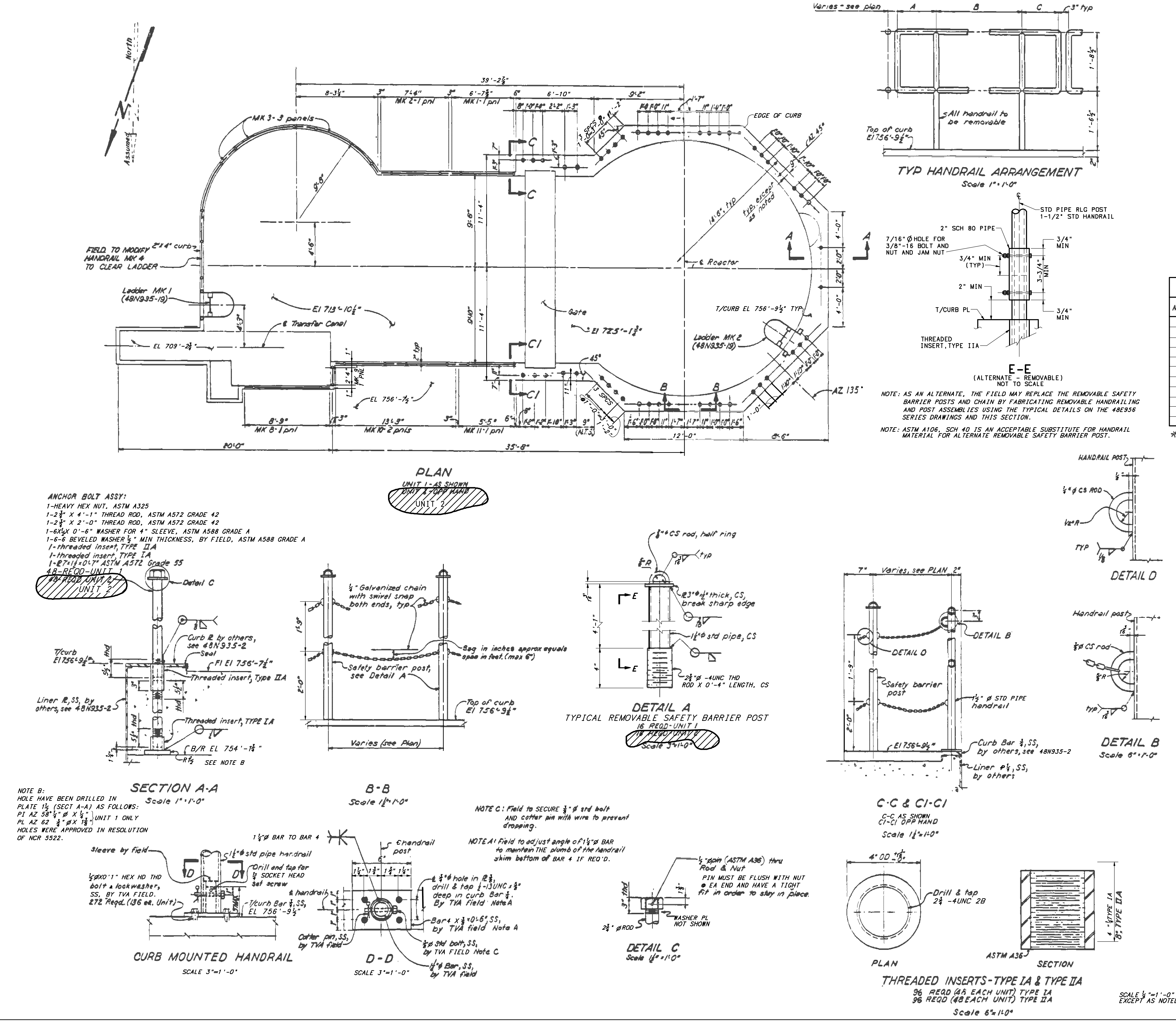
SCALE 3/16" = 1'-0"
EXCEPT AS NOTED

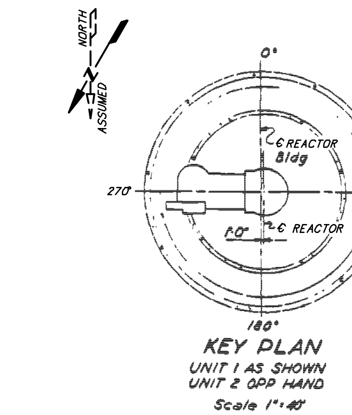
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
FLOOR-EL 756.63
OUTLINE
TVA DWG NO. 41N735-1 RG
FIGURE 3.8.3-13

CAD MAINTAINED DRAWING





SCHEDULE - HANDRAIL PANEL-CURB MOUNTED					
MARK NO.	* NO. REQD	A	B	C	REMARKS
1	1	0"	5'-8⅝"	11"	
2	1	11"	5'-6"	11"	
3	3	11"	6'-3½"	11"	Arc length-bend to 9'-8" ℄ R
4	2	11"	3'-5"	11"	
7	2	11"	2'-7½"	11"	
8	1	11"	6'-11"	11"	
9	1	0"	2'-4"	0"	
10	2	11"	4'-8"	11"	
11	1	0"	4'-6"	11"	

* Number required for each Unit

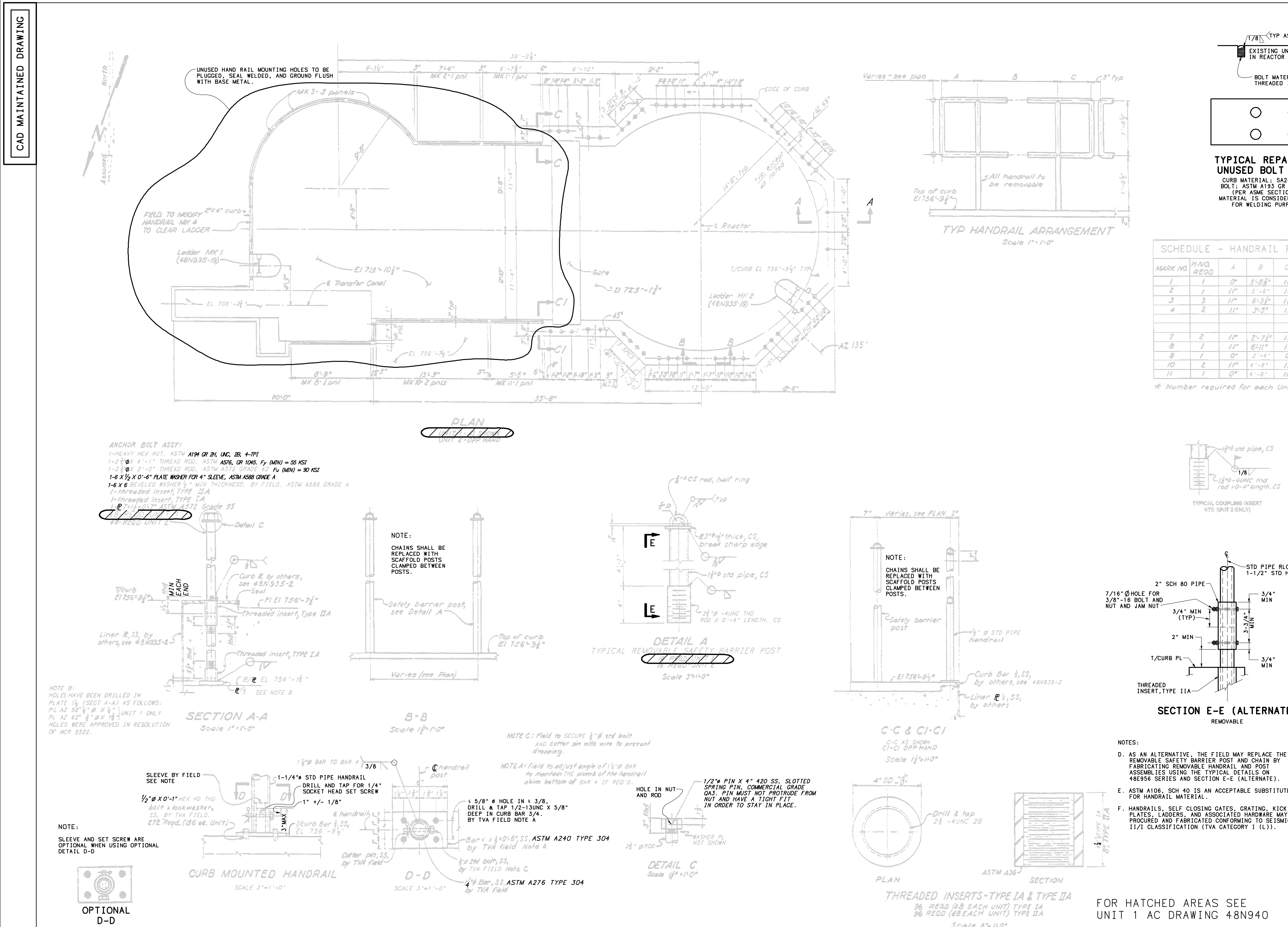
ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
NO Q.A. DOCUMENTION REQUIRED FOR HANDRAIL. Q.A. FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONSTRUCTION SPECIFICATION N36-881, U.O.N. ALL WELDING BY TVA FIELD TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C
WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N36-881, U.O.N.
PROCUREMENT, FABRICATION AND DOCUMENTION OF MATERIAL BY CONTRACTOR SHALL CONFORM TO THE TVA Q.A. PLAN.

ALL MATERAIL AND FABRICATION BY TVA FIELD, EXCEPT AS NOTED.
ALL HANDRAIL SHALL BE 1½" ØSTD PIPE, ASTM 120 OR ASTM A36.
SHORT RADIUS WELDING ALL, OR PARTS THEREOF, SHALL BE USED FOR ALL HANDRAIL BENDS.
ALL HANDRAIL PANELS TO BE REMOVEABLE.
EACH REMOVABLE SECTION OF HANDRAIL SHALL HAVE A DIFFERENT AND PERMANENT IDENTIFICATION MARKING.
SS-DENOTES STAINLESS STEEL TYPE 304.
ALL CARBON STEEL (CS) TO BE ASTM A36, EXCEPT AS NOTED.
● -DENOTES LOCATION WHERE SAFTEY BARRIER POST (DETAIL A) IS INSERTED INTO ANCHOR BOLT INSERT AFTER MISSILE SHIELD HAS BEEN REMOVED.
QA REQUIRED FOR ANCHOR BOLT ASSEMBLY: QUALITY LEVEL I IN ACCORDANCE WITH WBNP CONSTRUCTION SPECIFICATION N36-881, EXCEPT AS NOTED, NOTED, NUTS AND WASHERS TO BE QUALITY LEVEL II.
WELDING ELECTRODES;
- WELDING SS TO SS-AWS A5.4, E308 CLASSIFICATION.
- WELDING CS TO SS-AWS A5.4, E309 CLASSIFICATION.
- WELDING CS TO CS-AWS A5.1, E70 CLASSIFICATION.

SCALE ½"=1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL
REACTOR WELL HANDRAIL & MISSILE
SHIELD ANCHOR BOLTS-EL. 756.63
TVA DWG NO. 48N940 RD
FIGURE 3.8.3-14

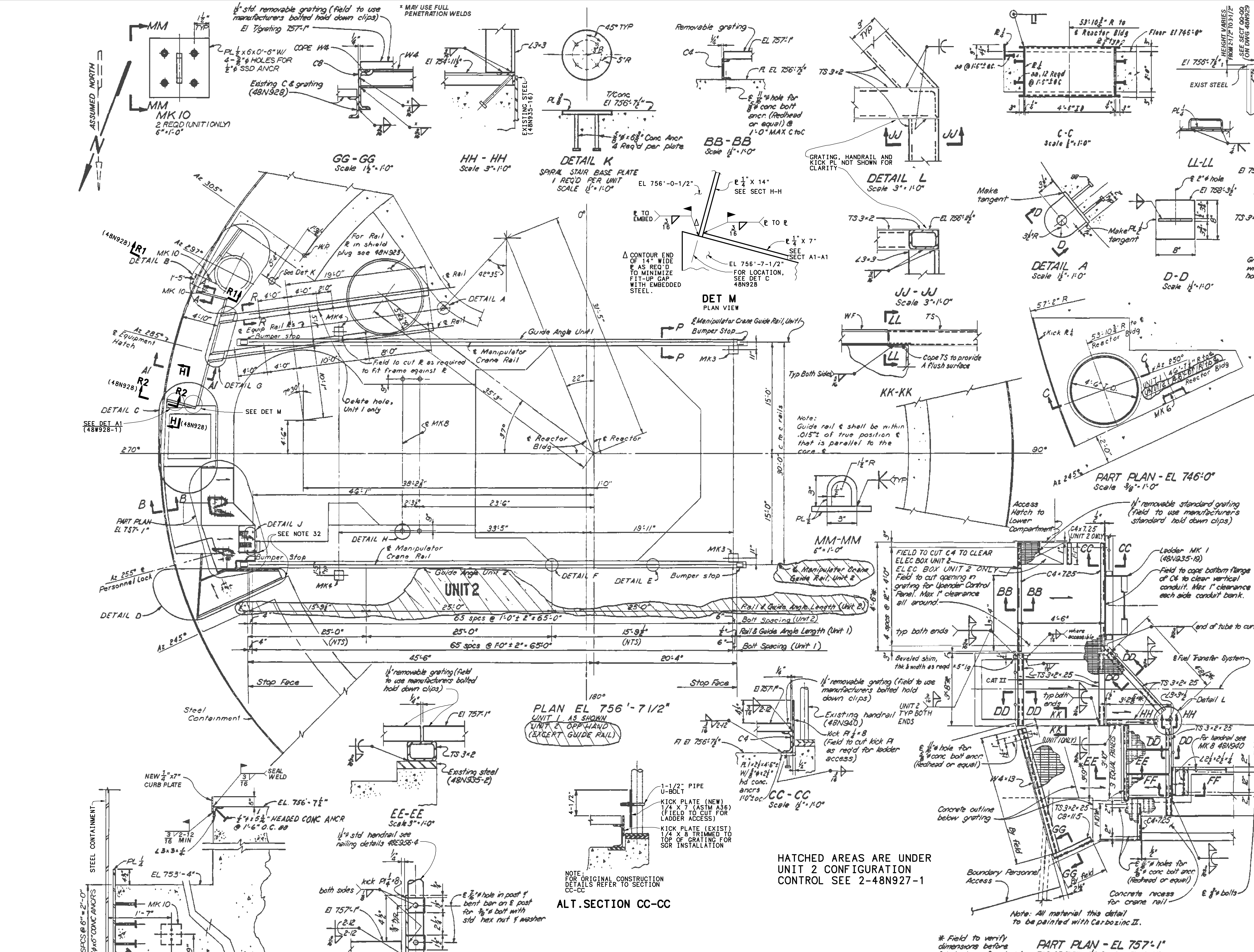


UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL
REACTOR WELL HANDRAIL & MISSILE
SHIELD ANCHOR BOLTS-EL. 756.63
TVA DWG NO. 2-48N940 R3
FIGURE 3.8.3-14(U2)

CAD MAINTAINED DRAWING



THIS DRAWING IS A DUPLICATE OF SEQUOYAH NUCLEAR PLANT DRAWING 48N927 R5 EXCEPT AS NOTED BY ENCIRCLEMENTS ON R0 ISSUE. ELEVATIONS NOT ENCIRCLED ON R0 ISSUE ARE SEQUOYAH REACTOR BUILDING ELEVATIONS RAISED BY 23'-0". ENCIRCLED ELEVATIONS ON R0 ISSUE HAVE BEEN OTHERWISE CHANGED TO SATISFY DESIGN REQUIREMENTS

NOTES:

1. ALL GRATING TO BE PAINTED GRATING EXCEPT AS NOTED.
2. GRATING PURCHASED ON REQ № 821414 IN STOCK SIZES EXCEPT GRATING ON PART PLAN-EL 756'-9 1/2".
3. ALL HOT DIP GALVANIZED GRATING TO HAVE LOAD CARRYING BARS CUT SQUARE AND THE CLEARANCE TO THE BANDING BARS TO BE SUCH AS TO PERMIT COMPLETE GALVANIZED COATING.
4. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
5. QA FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONSTRUCTION SPECIFICATION N3C-881, U.O.N. G29C.
6. WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION
7. PROCUREMENT, FABRICATION AND DOCUMENTATION OF MATERIAL BY CONTRACTOR SHALL CONFORM TO THE TVA Q.A. PLAN
8. FOR SECTIONS AND DETAILS NOT SHOWN SEE 48N928 AND 48N929.
9. ALL MATERIAL AND FABRICATION, EXCEPT GRATING, CRANE RAILS, RAIL CLIPS, AND BUMPER STOPS, BY TVA FIELD.
10. STAINLESS STEEL NUTS SHALL BE ASTM A194 GRADE 8.
11. ALL STRUCTURAL SHAPES AND PLATES TO BE ASTM A36.
12. CRANE RAIL AND RAIL CLIPS FURNISHED BY WESTINGHOUSE.
13. ALL HEADED CONCRETE ANCHORS TO BE ASTM A108.
14. STAINLESS STEEL THREADED RODS SHALL BE ASTM A320 GR B8, OR A276, TYPE 304 EXCEPT AS NOTED & 30,000 PSI YIELD STRENGTH MIN.
15. ALL GRATING TO BE RECTANGULAR TYPE WITH LOAD BARS 1 1/2 x 3/16 AND SPACED AT 1 3/16" O.C. BANDING BARS TO BE 1 1/2 x 3/16 AND WELDED CONTINUOUS TO EVERY FOURTH LOAD CARRYING BAR.
16. ALL GRATING SHALL BE TACK WELDED TO SUPPORTING STEEL UNLESS OTHERWISE SHOWN OR NOTED.
17. ARROW (⟵⟶) INDICATES DIRECTION OF BEARING BARS.
18. HANDRAILING SYMBOL SHOWN ON PLAN THUS —o—.
19. HANDRAILING TO BE 1 1/2"ϕ STD. PIPE, ASTM A120 OR A36.
20. ALL TVA WELDING TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G29C.
21. * INDICATES WELDS TO BE EXAMINED BY LIQUID PENETRANT OR MAGNETIC PARTICLE METHOD.
22. W.P. DENOTES WORK POINT.
23. NUMBER SHOWN IS FOR ONE UNIT.
24. FOR DETAILS OF MKS NOT SHOWN SEE COMPANION DWGS.
25. ALL FILLET WELDS TO BE 3/16" EXCEPT AS NOTED.
26. GRATING FABRICATION BY TVA FIELD.
27. NO Q.A. DOCUMENTATION REQUIRED FOR HANDRAIL.
28. FIELD WELDING OF MK 10 TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G29-M.
29. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
30. THIS PLATFORM WAS EVALUATED DURING THE SEISMIC/CIVIL VALIDATION PROGRAM (WCG-1-1419). ACCEPTANCE IS PROVIDED IN CALCULATION WCG-1-1316.

NOTES CONT:

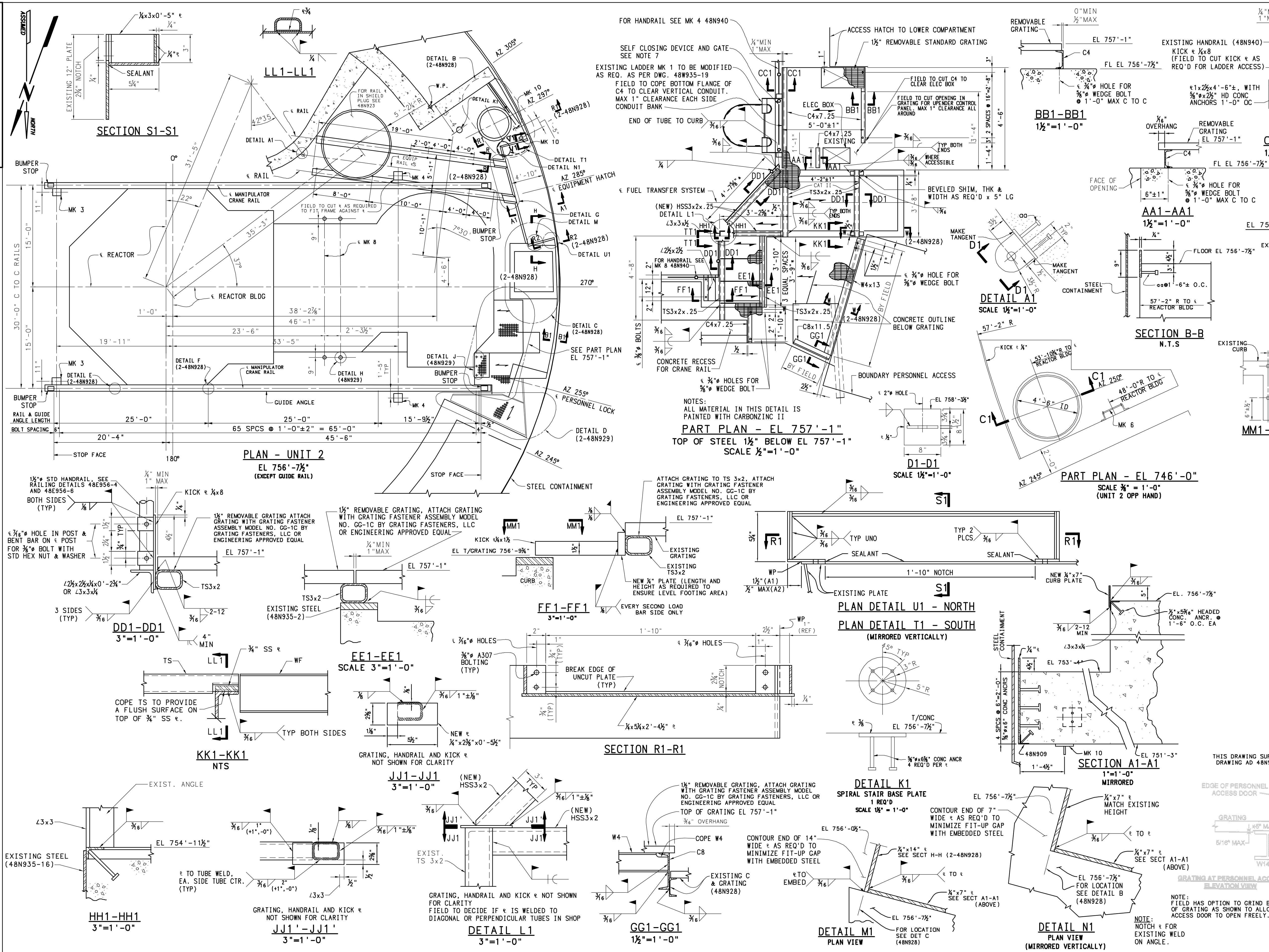
31. CATEGORY II MEMBERS DO NOT PROVIDE SUPPORT TO, ARE NOT IN THE LOAD PATH OF, OR WILL NOT COME IN CONTRACT WITH ANY CATEGORY I STRUCTURE, SYSTEM OR COMPONENT. ATTACHING SAFETY RELATED FEATURES TO OR LOCATING SAFETY RELATED FEATURES IN THE PROXIMITY OF THESE MEMBERS IS NOT PERMITTED WITHOUT ENGINEERING APPROVAL. SEE CALCULATION WCG-1-1336 AND SPECIFICATION N3C-941.
32. THE FLOOR COATING IS TO BE REPLACED, THEREFORE ANY SMALL GAPS ALONG THE CONCRETE/CARBON STEEL INTERFACE WILL BE FILLED AT THE TIME IN ACCORDANCE WITH ENGINEERING SPECIFICATION N3A-932.

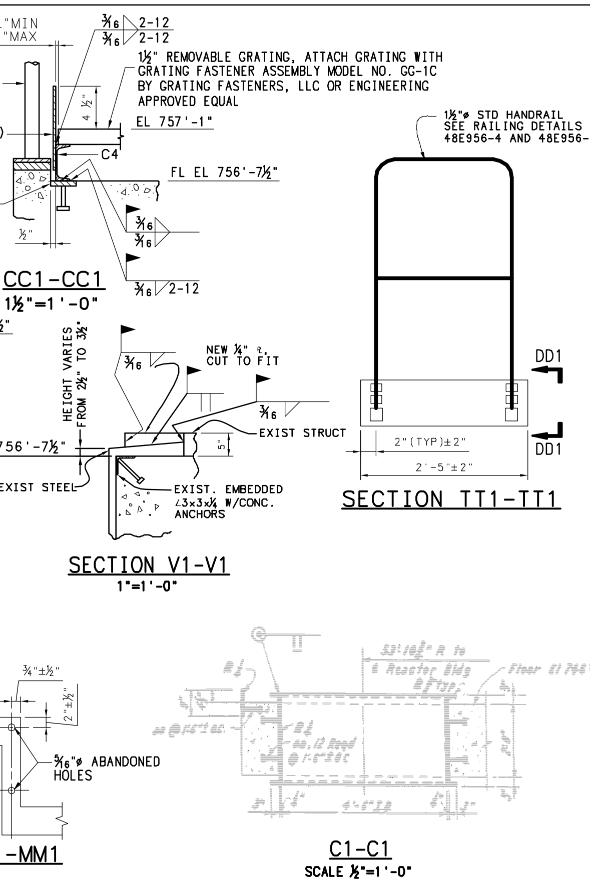
Scale 3/16" = 1'-0"
Except as noted
Companion drawing 2-48N927-1, 2-48W928, 48N929

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL FRAMES,
GRATING AND EMB PARTS,
EL 756.63 - SHEET 1
TVA DWG NO. 48N927 RK
FIGURE 3.8.3-15





NOTES:

1. ALL MATERIAL AND FABRICATION BY BECHTEL FIELD.
2. GRATING COATING IS SL-1-104, CARBOZINC IISG W/ GRATING LOAD CARRYING BARS CUT SQUARE. CLEARANCE TO THE BENDING BARS TO BE SUCH AS TO PERMIT COMPLETE CARBOZINC IISG COATING.
3. TOUCH-UP WITH COATING SL-1-104, CARBOZINC IISG AT ALL WELDING AND BOLT HOLE SURFACES IN THE GRATING.
4. ALL GRATING TO BE RECTANGULAR TYPE WITH LOAD BARS 1½"x 3⁄16" AND SPACE AT 1 3⁄16"O.C. BANDING BARS TO BE 1½"X 3⁄16" AND WELDED CONTINUOUS TO EVERY 4th LOAD CARRYING BAR.
5. HAND RAILING TO BE 1½"ø STD. PIPE FIELD PAINTED.
6. MATERIALS SHALL CONFORM TO THE FOLLOWING:
   - A. STRUCTURAL SHAPES AND PLATES: ASTM A36 (Fy = 36 ksi) OR ASTM A572, GRADE 50.
   - B. PIPES FOR HANDRAILS: ASTM A53, GRADE B SCHEDULE 40.
   - C. TUBE STEEL: ASTM A500, GRADE B.
   - D. BOLTS: ASTM A325.
   - E. NUTS: ASTM A563, GRADE DH.
   - F. SHORT RADIUS ELBOWS: ASTM A234, GRADE WPB SCHEDULE 40.
   - G. HANDRAIL BOLTS: ASTM A307, AS SPECIFIED ON NOTE 13 OF DWG. 48E956-4.
   - H. ⅛" BENT BAR: ASTM A1011, GRADE 36 TYPE B.
   - I. GRATING: ASTM A1011.
7. BECHTEL FIELD FABRICATED SELF-CLOSING SAFETY DEVICES ARE DETAILED ON DRAWING 48E956-5.
8. PREFABRICATED SELF-CLOSING SAFETY DEVICES APPROVED BY SITE ENGINEERING INCLUDE FabEnCo's SELF-CLOSING SAFETY GATE OR INTREPID INDUSTRIES BAR TYPE GATE. OTHER SELF-CLOSING SAFETY DEVICES APPROVED BY SITE ENGINEERING OR THE SAFETY ENGINEER MAY BE USED.
9. COPE TS OR ADD SHIMS/BEVELED SHIMS WHERE NEEDED IN ORDER TO INSURE THE GRATING IS LEVEL.
10. HANDRAILING AND GRATING ARE QUALITY RELATED. STRUCTURAL STEEL SHAPES, TUBE STEEL, BOLTS AND NUTS ARE SAFETY RELATED.
11. GRATING FASTENER ASSEMBLY MODEL NO. GT-W WITH SELF TAPPING SCREWS BY GRATING FASTENERS LLC MAY BE USED IN PLACE OF MODEL NO. GG-1C, IN AREAS WHERE GRATING IS TO BE ATTACHED TO TUBE STEEL.
12. GLIDELOC VERTICAL HEIGHT ACCESS LADDER SYSTEM KIT (STAINLESS STEEL) MAY BE USED AS THE SAFETY CLIMBING DEVICE.
13. CATEGORY II MEMBERS (DESIGNED BY CAT II) DO NOT PROVIDE SUPPORT TO, ARE NOT IN THE LOAD PATH OF, AND SHALL NOT COME INTO CONTACT WITH ANY CATEGORY I STRUCTURE, SYSTEM OR COMPONENT. ATTACHING SAFETY RELATED FEATURES TO OR LOCATING SAFETY FEATURES IN THE PROXIMITY OF THESE MEMBERS IS NOT PERMITTED WITHOUT ENGINEERING APPROVAL. SEE CALCULATION WCG-2-359 AND SPECIFICATION N3C-941.
14. FOR SECTIONS AND DETAILS NOT SHOWN SEE 48N928 AND 48N929.
15. CRANE RAIL AND RAIL CLIPS FURNISHED BY WESTINGHOUSE.
16. ARROW (⟵⟶) INDICATES DIRECTION OF BEARING BARS.
17. ALL FILLET WELDS TO BE 3⁄16" EXCEPT AS NOTED.

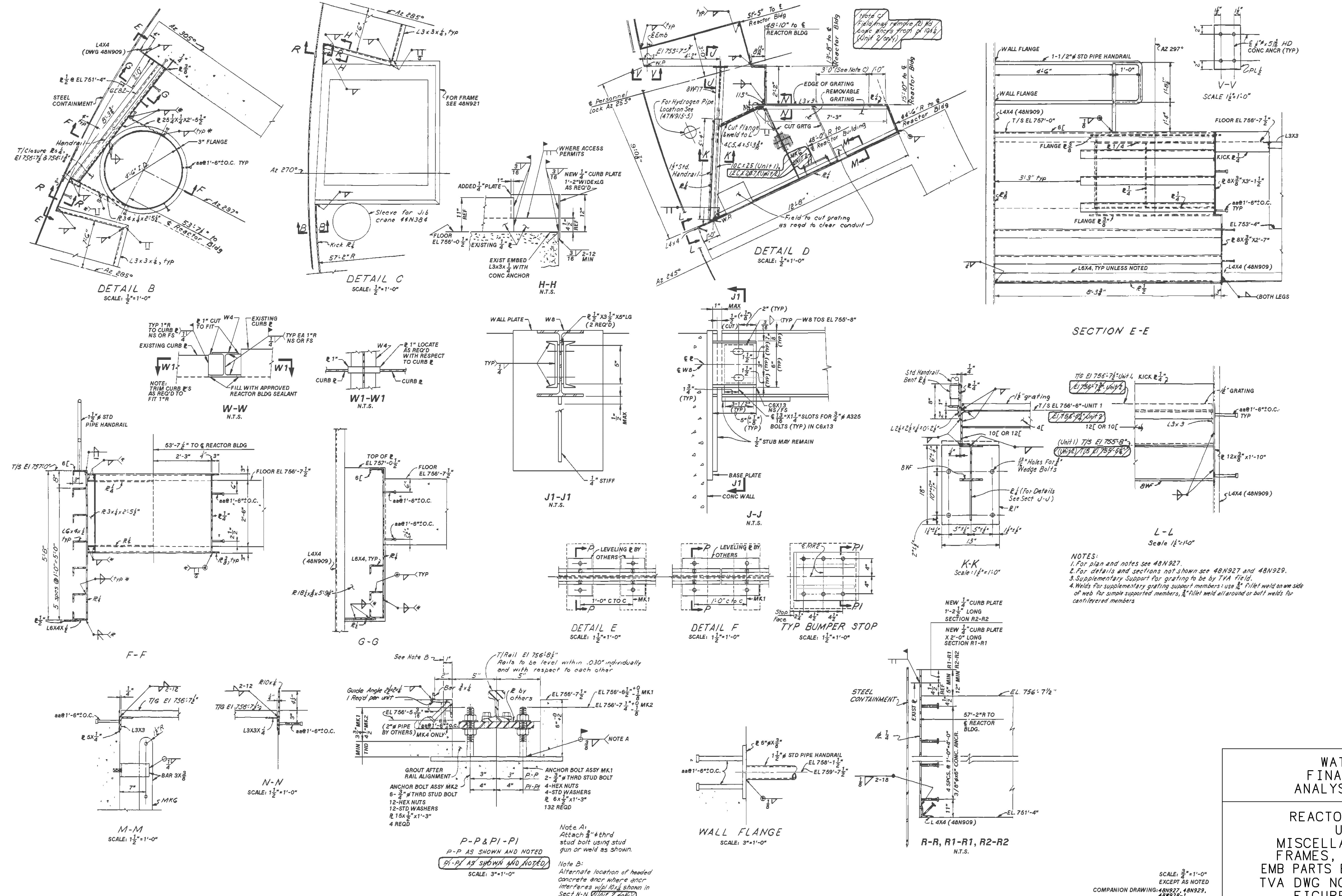
THIS DRAWING SUPERSEDES
DRAWING AD 48N927 R19

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

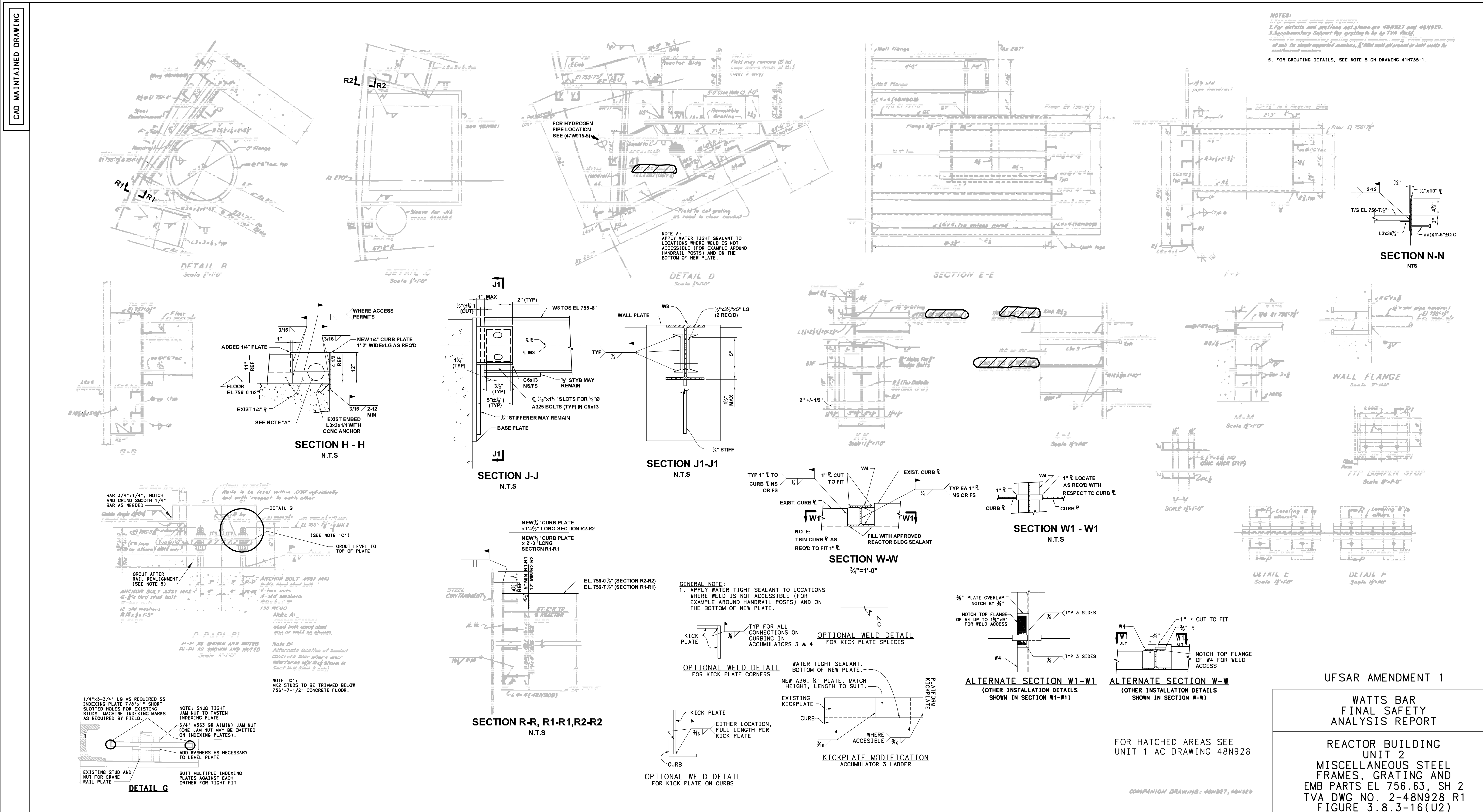
REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL FRAMES,
GRATING AND EMB PARTS,
EL 756.63
TVA DWG NO. 2-48N927-1 R0
FIGURE 3.8.3-15(U2)

CAD MAINTAINED DRAWING



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL
FRAMES, GRATING AND
EMB PARTS EL 756.63, SH 2
TVA DWG NO. 48N928 RE
FIGURE 3.8.3-16

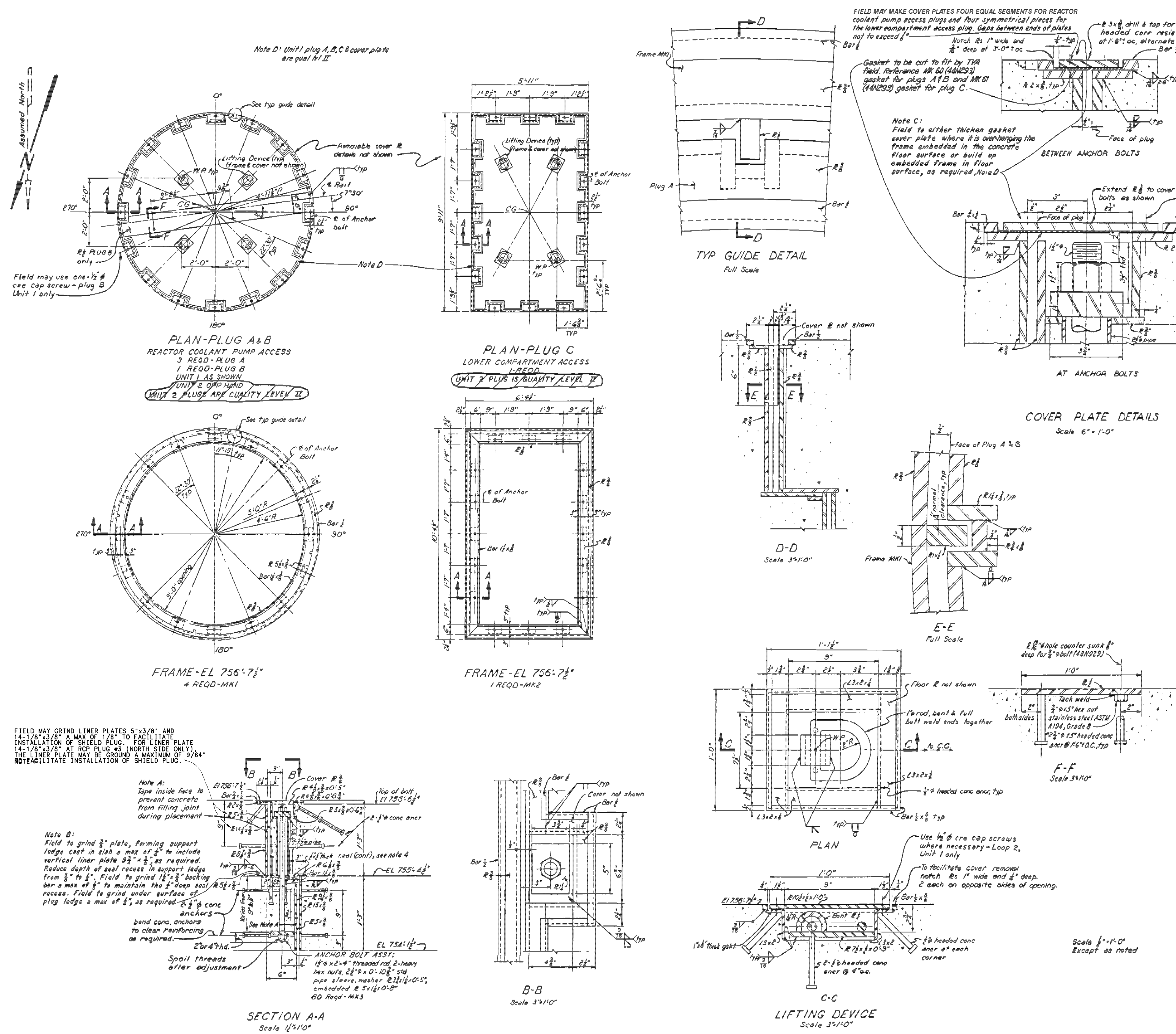


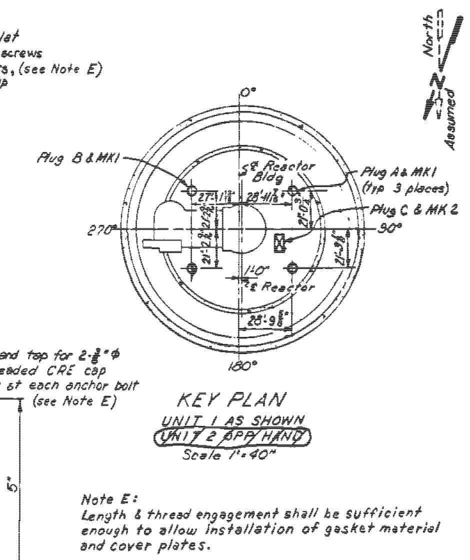
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL
FRAMES, GRATING AND
EMB PARTS EL 756.63, SH 2
TVA DWG NO. 2-48N928 R1
FIGURE 3.8.3-16(U2)

CAD MAINTAINED DRAWING



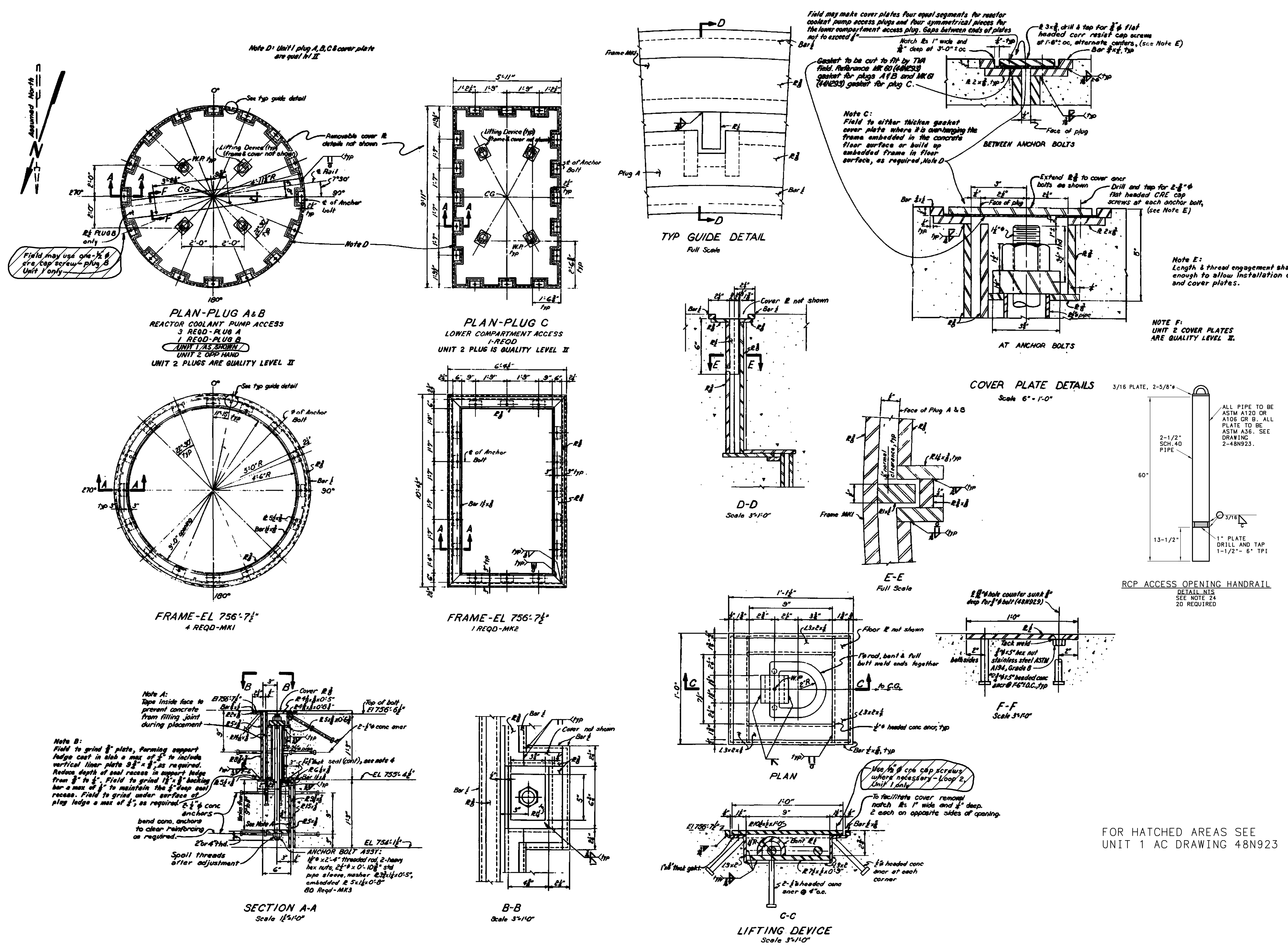


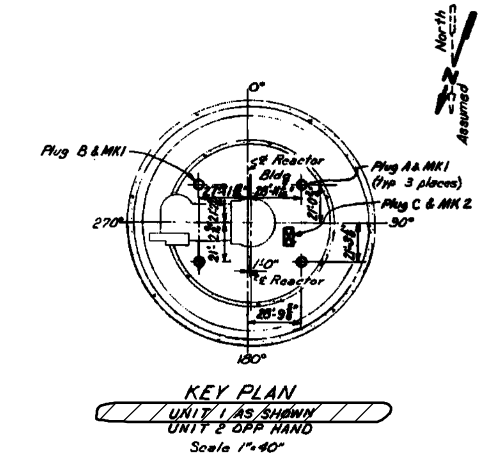
NOTE F:
UNIT 2 COVER PLATES
ARE QUALITY LEVEL II.

1. FIELD MAY CUT, PLUG WELD, AND REDRILL REMOVABLE COVER PLATES FOR REACTOR SHIELD PLUGS A,B,&C AS REQD TO OBTAIN MAX FIT WITH FRAMES IN PLUG & FL SLAB. REJOIN COVER PLS W/SINGLE BEV FULL PEN WELD.
2. FIELD TO ADD SHIMS AS REQUIRED TO THE LEDGES OF PLUG RECESSES TO OBTAIN MINIMUM BEARING SURFACE REQUIRED. SHIMS TO VARY IN LENGTH AND THICKNESS AS REQUIRED. SHIMS MAYBE ASTM A36 OR A570 GRADE D.
3. PLUG GUIDES MAY BE LOCATED TO ACCOMMODATE ROTATION(REQUIRED FOR MAX BEARING SURFACE)OF THE PLUGS FOR LOOPS 1,2,&3. PLUG GUIDES TO REMAIN LOCATED AT 90° INCR BUT AT VARYING ANGLES MEASURED FROM 0°.
4. FIELD TO MAINTAIN A ¼" RECESS DEPTH FOR ⅛"-¼" SEAL IN LIEU OF SHIMS ON LEDGES, FIELD MAY BUTTER WELD EITHER THE ⅜" PL IN THE SLAB LEDGES OR THE UNDERNEATH PORTION OF THE ⅜" PL IN THE PLUGS.
5. PROCUREMENT, FABRICATION AND DOCUMENTATION OF MATERIAL BY CONTRACTOR SHALL CONFORM TO THE TVA Q.A. PLAN.
6. WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3G-881.
7. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
8. QA FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONSTRUCTION SPECIFICATION N3G-881, EXCEPT AS NOTED.
9. ALL MATERIAL TO BE ASTM A36, EXCEPT AS NOTED.
10. ALL NUTS TO BE ASTM A307, EXCEPT AS NOTED.
11. ALL MATERIAL AND FABRICATION BY TVA FIELD, EXCEPT PLUGS A,B & C.
12. PIPE SLEEVES TO BE ASTM A53.
13. NUMBER REQUIRED IS FOR ONE UNIT.
14. ALL HEADED CONCRETE ANCHORS TO BE ASTM A108, 1/2"φ x 5 3/16"± SPACING @ 1'-6" O.C., EXCEPT AS NOTED.
15. ALL CORROSION RESISTANT SCREWS AND BOLTS TO BE ASTM A193-GRADE B8.
16. LIFTING DEVICE FRAMES TO BE DRILLED AND TAPPED TO FASTEN COVERS WITH FOUR 3/8"φ FLATHEAD CORROSION RESISTANT CAP SCREWS.
17. FOR LOCATION OF SIELD PLUGS SEE 41735-1.
18. ALL WELDING BY TVA FIELD TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.
19. ALL WELDS TO BE LIQUID PENETRANT OF MAGENETIC PARTICLE INSPECTED.
20. WELDING ELECTRODES TO BE 4WS A5.1, E70 SERIES.
21. SEAL AND GASKET MATERIAL TO BE FURNISHED BY HEAVY EQUIPMENT GROUP.
22. ALL 11D CONC ANCRS, PIPE SLEEVES, NUTS AND WASHERS THAT ARE BEING FURNISHED BY TVA FIELD REQUIRE ONLY CERTIFICATION THAT CONFORM TO THE APPROCIATE ASTM SPECIFICATION.
23. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL
SHIELD PLUGS & FRAMES
EL. 756.63
TVA DWG NO. 48N923 RE
FIGURE 3.8.3-17





1. FIELD MAY CUT, PLUG WELD, AND REDRILL REMOVABLE COVER PLATES FOR REACTOR SHIELD PLUGS A,B,&C AS REQD TO OBTAIN MAX FIT WITH FRAMES IN PLUG & FL SLAB. REJOIN COVER PLS W/SINGLE BEV FULL PEN WELD.
2. FIELD TO ADD SHIMS AS REQUIRED TO THE LEDGES OF PLUG RECESSES TO OBTAIN MINIMUM BEARING SURFACE REQUIRED. SHIMS TO VARY IN LENGTH AND THICKNESS AS REQUIRED. SHIMS MAYBE ASTM A36 OR A570 GRADE D.
3. PLUG GUIDES MAY BE LOCATED TO ACCOMMODATE ROTATION (REQUIRED FOR MAX BEARING SURFACE) OF THE PLUGS FOR LOOPS 1,2,&3. PLUG GUIDES TO REMAIN LOCATED AT 90° INCR BUT AT VARYING ANGLES MEASURED FROM 0°.
4. FIELD TO MAINTAIN A ¾" RECESS DEPTH FOR ¾"×¾" SEAL. IN LIEU OF SHIMS ON LEDGES, FIELD MAY BUTTER WELD EITHER THE ¾"PL IN THE SLAB LEDGES OR THE UNDERNEATH PORTION OF THE ¾"PL IN THE PLUGS.
5. PROCUREMENT, FABRICATION AND DOCUMENTATION OF MATERIAL BY CONTRACTOR SHALL CONFORM TO THE TVA QA. PLAN.
6. WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3C-881.
7. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
8. QA FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONSTRUCTION SPECIFICATION N3C-881, EXCEPT AS NOTED.
9. ALL MATERIAL TO BE ASTM A36, EXCEPT AS NOTED.
10. ALL NUTS TO BE ASTM A307, EXCEPT AS NOTED.
11. ALL MATERIAL AND FABRICATION BY TVA FIELD, EXCEPT PLUGS A,B & C.
12. PIPE SLEEVES TO BE ASTM A53.
13. NUMBER REQUIRED IS FOR ONE UNIT.
14. ALL HEADED CONCRETE ANCHORS TO BE ASTM A108, 1/2"Φ x 5 3/16"± SPACING @ 1'-6" O.C., EXCEPT AS NOTED.
15. ALL CORROSION RESISTANT SCREWS AND BOLTS TO BE ASTM A193-GRADE B8.
16. LIFTING DEVICE FRAMES TO BE DRILLED AND TAPPED TO FASTEN COVERS WITH FOUR 3/8"Φ FLATHEAD CORROSION RESISTANT CAP SCREWS.
17. FOR LOCATION OF SHIELD PLUGS SEE 41N735-1.
18. ALL WELDING BY TVA FIELD TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.
19. ALL WELDS TO BE LIQUID PENETRANT OR MAGNETIC PARTICLE INSPECTED.
20. WELDING ELECTRODES TO BE AWS A5.1, E70 SERIES.
21. SEAL AND GASKET MATERIAL TO BE FURNISHED BY HEAVY EQUIPMENT GROUP.
22. ALL HD CONC ANCRS, PIPE SLEEVES, NUTS AND WASHERS THAT ARE BEING FURNISHED BY TVA FIELD REQUIRE ONLY CERTIFICATION THAT THEY CONFORM TO THE APPROPRIATE ASTM SPECIFICATION.
23. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
24. FOR DETAILS OF HANDRAIL POSTS AROUND RCP ACCESS OPENING, SEE FCR 62126A. FOR ADDITIONAL DETAILS REFER TO DRAWING 2-48N940.
25. FOR LIFTING DEVICE FRAMES WITH DAMAGED 3/8" THREADS, FIELD MAY DRILL, TAP, AND INSTALL 3/8"ø×16 THREADS PER INCH × 9/16"OD HELICOIL CAT ID# CLH064X USING DRILL AND TAP PROVIDED IN KIT. USE LOKTITE 242 ON COIL OD, STAKE AT LOCATION AND GRIND COIL FLUSH.

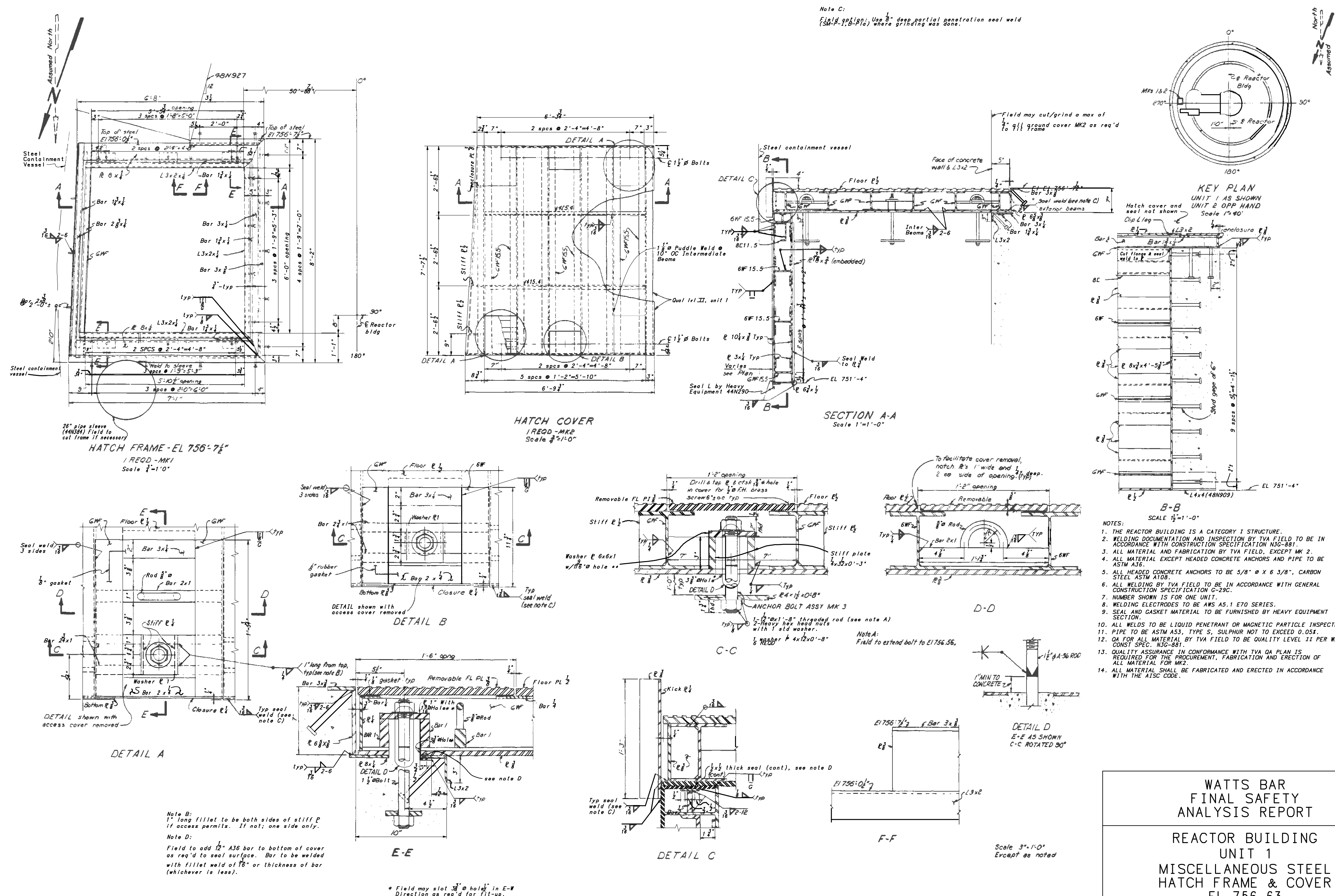
FOR HATCHED AREAS SEE UNIT 1 AC DRAWING 48N923

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL
SHIELD PLUGS & FRAMES
EL. 756.63
TVA DWG NO. 2-48N923 R1
FIGURE 3.8.3-17(U2)

CAD MAINTAINED DRAWING



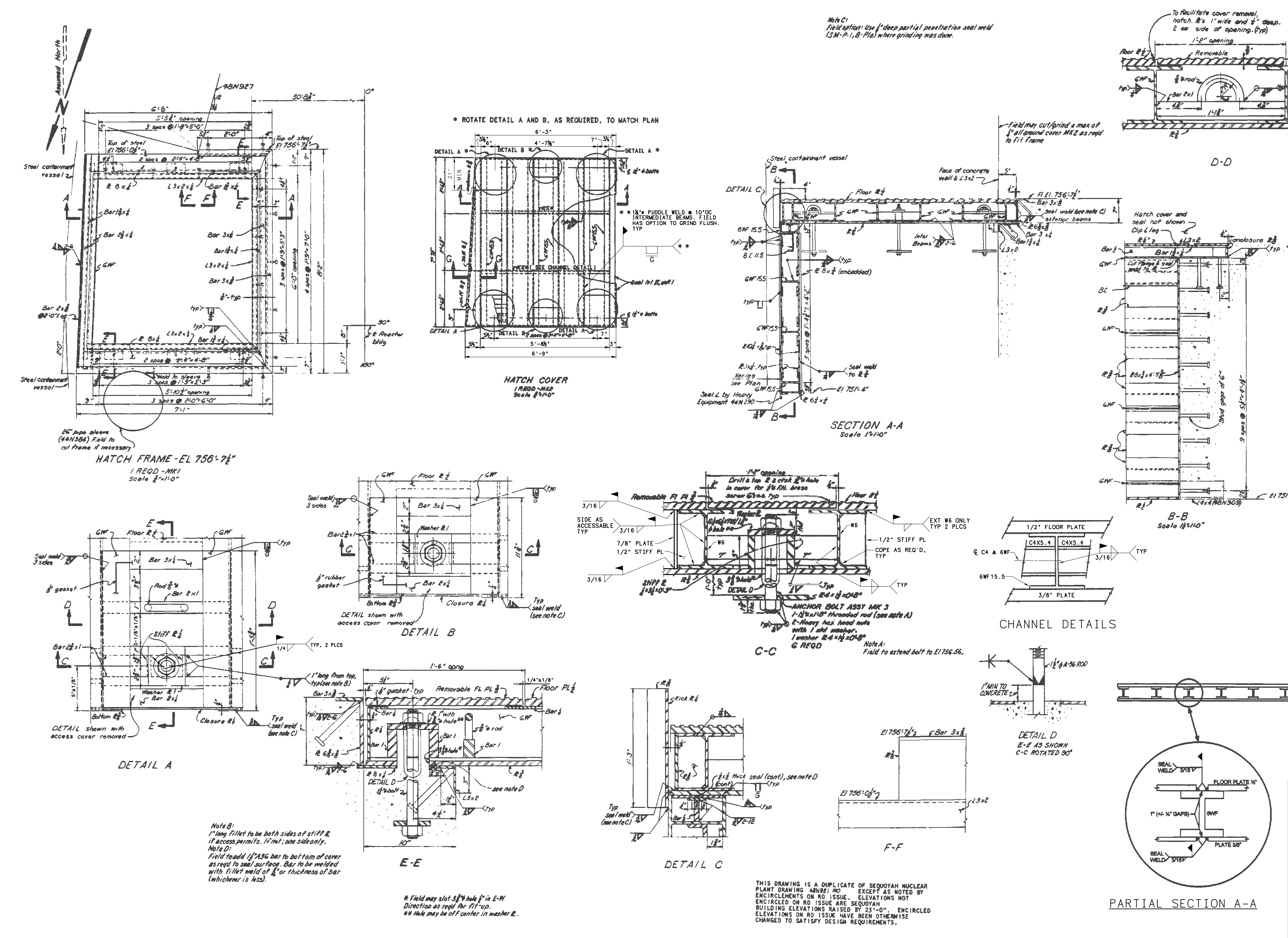
NOTES:

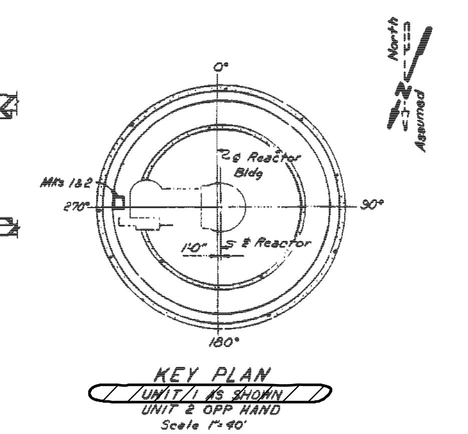
1. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
2. WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3G-881.
3. ALL MATERIAL AND FABRICATION BY TVA FIELD, EXCEPT MK 2.
4. ALL MATERIAL EXCEPT HEADED CONCRETE ANCHORS AND PIPE TO BE ASTM A36.
5. ALL HEADED CONCRETE ANCHORS TO BE 5/8" Ø X 6 3/8", CARBON STEEL ASTM A108.
6. ALL WELDING BY TVA FIELD TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.
7. NUMBER SHOWN IS FOR ONE UNIT.
8. WELDING ELECTRODES TO BE AWS A5.1 E70 SERIES.
9. SEAL AND GASKET MATERIAL TO BE FURNISHED BY HEAVY EQUIPMENT SECTION.
10. ALL WELDS TO BE LIQUID PENETRANT OR MAGNETIC PARTICLE INSPECTED.
11. PIPE TO BE ASTM A53, TYPE S, SULPHUR NOT TO EXCEED 0.05%.
12. QA FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONST SPEC. N3G-881.
13. QUALITY ASSURANCE IN CONFORMANCE WITH TVA QA PLAN IS REQUIRED FOR THE PROCUREMENT, FABRICATION AND ERECTION OF ALL MATERIAL FOR MK2.
14. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
MISCELLANEOUS STEEL
HATCH FRAME & COVER
EL 756.63
TVA DWG NO. 48N921 RB
FIGURE 3.8.3-18

CAD MAINTAINED DRAWING





NOTES:

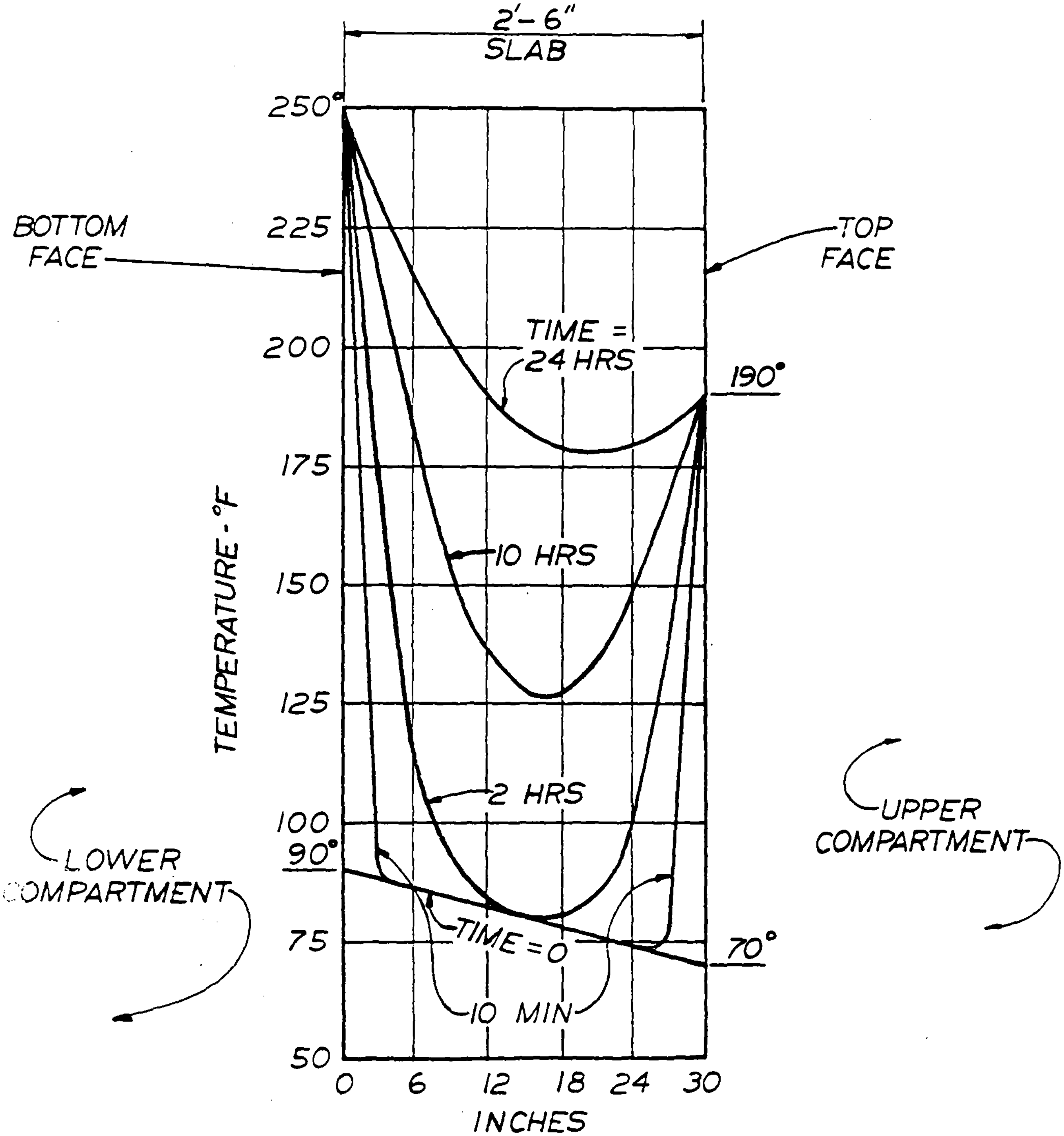
1. THE REACTOR BUILDING IS A CATEGORY I STRUCTURE.
2. WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3G-881.
3. ALL MATERIAL AND FABRICATION BY TVA FIELD, EXCEPT MK 2.
4. ALL MATERIAL EXCEPT HEADED CONCRETE ANCHORS AND PIPE TO BE ASTM A36.
5. ALL HEADED CONCRETE ANCHORS TO BE 5/8"⌀ x 6 3/8", CARBON STEEL ASTM A108.
6. ALL WELDING BY TVA FIELD TO BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.
7. NUMBER SHOWN IS FOR ONE UNIT.
8. WELDING ELECTRODES TO BE AWS A5.1 E70 SERIES.
9. SEAL AND GASKET MATERIAL TO BE FURNISHED BY HEAVY EQUIPMENT SECTION.
10. ALL WELDS TO BE LIQUID PENETRANT OR MAGNETIC PARTICLE INSPECTED.
11. PIPE TO BE ASTM A53, TYPE S, SULPHUR NOT TO EXCEED 0.05%.
12. QA FOR ALL MATERIAL BY TVA FIELD TO BE QUALITY LEVEL II PER WBNP CONST SPEC N3G-881
13. QUALITY ASSURANCE IN CONFORMANCE WITH TVA QA PLAN IS REQUIRED FOR THE PROCUREMENT, FABRICATION AND ERECTION OF ALL MATERIAL FOR MK 2.
14. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH THE AISC CODE.
15. SEE FCR 61500, FCR 62126 AND FCR 61187 FOR EQUIPMENT HATCH REMOVABLE HANDRAIL DETAILS AND KICKER PLATE.
16. SEE FCR 61500 AA-01 FOR WELDING DETAILS FOR HAND RAIL.

FOR HATCHED AREAS SEE UNIT 1 AC DRAWING 48N921

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

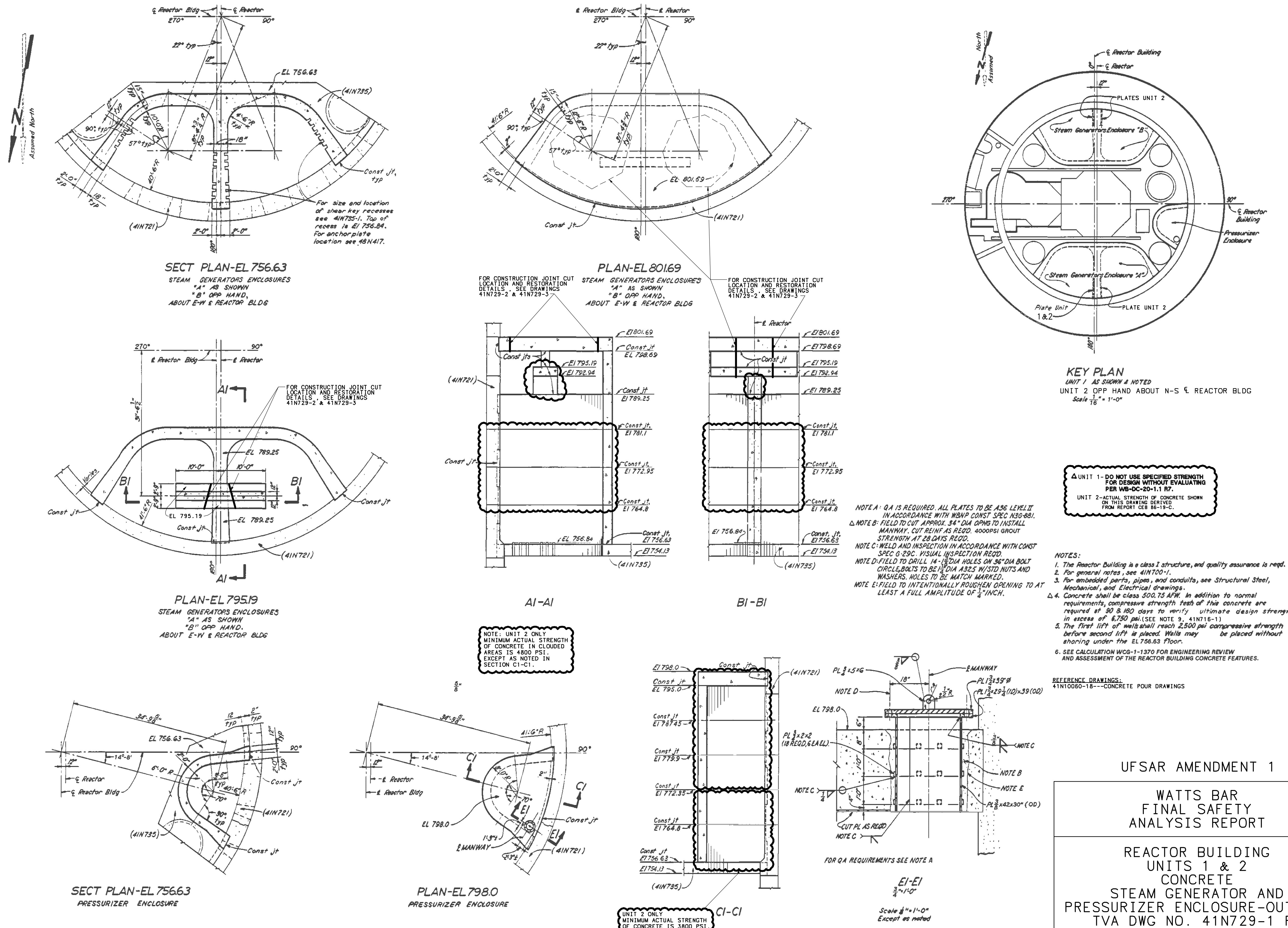
REACTOR BUILDING
UNIT 2
MISCELLANEOUS STEEL
HATCH FRAME & COVER
EL 756.63
TVA DWG NO. 2-48N921 R2
FIGURE 3.8.3-18(U2)



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Divider Floor
LOCA Temperature Gradients

FIGURE 3.8.3-19



NOTE A: QA IS REQUIRED. ALL PLATES TO BE A36 LEVEL II IN ACCORDANCE WITH WBNP CONST SPEC N3G-881.

△ NOTE B: FIELD TO CUT APPROX. 34" DIA OPNG TO INSTALL MANWAY. CUT REINF AS REQD. 4000PSI GROUT STRENGTH AT 28 DAYS REQD.

NOTE C: WELD AND INSPECTION IN ACCORDANCE WITH CONST SPEC G-29C. VISUAL INSPECTION REQD.

NOTE D: FIELD TO DRILL 14-1 3/8" DIA HOLES ON 36" DIA BOLT CIRCLE, BOLTS TO BE 1 1/4" DIA A325 W/STD NUTS AND WASHERS. HOLES TO BE MATCH MARKED.

NOTE E: FIELD TO INTENTIONALLY ROUGHEN OPENING TO AT LEAST A FULL AMPLITUDE OF 1/4" INCH.

△ UNIT 1- DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

UNIT 2-ACTUAL STRENGTH OF CONCRETE SHOWN ON THIS DRAWING DERIVED FROM REPORT CEB 86-19-C.

NOTES:

1. The Reactor Building is a class I structure, and quality assurance is reqd.
2. For general notes, see 41N700-1.
3. For embedded parts, pipes, and conduits, see Structural Steel, Mechanical, and Electrical drawings.
4. △ Concrete shall be class 500.75 AFW. In addition to normal requirements, compressive strength tests of this concrete are required at 90 & 180 days to verify ultimate design strength in excess of 6,750 psi. (SEE NOTE 9, 41N716-1)
5. The first lift of walls shall reach 2,500 psi compressive strength before second lift is placed. Walls may be placed without shoring under the EL 756.63 floor.
6. SEE CALCULATION WCG-1-1370 FOR ENGINEERING REVIEW AND ASSESSMENT OF THE REACTOR BUILDING CONCRETE FEATURES.

REFERENCE DRAWINGS:
41N10060-18---CONCRETE POUR DRAWINGS

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNITS 1 & 2
CONCRETE
STEAM GENERATOR AND
PRESSURIZER ENCLOSURE-OUTLINE
TVA DWG NO. 41N729-1 RF
FIGURE 3.8.3-20

3.8.4 Other Category I Structures

The Category I structures other than the primary containment, interior structures, and Shield Building are listed as follows:

1. Auxiliary-Control Building and Associated Structures

   a. Control Bay Portion

   b. Auxiliary Building Portion

   c. Additional Equipment Building Portion

   d. Waste Packaging Structure

   e.. Condensate Demineralizer Waste Evaporator Structure Portion

2. Diesel Generator Building

3. Category I Water Tanks and Pipe Tunnels

4. Class 1E Electrical Systems Structures

5. North Steam Valve Room

6. Intake Pumping Station and Retaining Walls

7. Miscellaneous ERCW Structures

8. Additional Diesel Generator Building

9. South Steam Valve Room

The structures are designed as described in Sections 3.8.4.1 through 3.8.4.7.

Evaluations and modifications of the reinforced concrete structures are optionally done using ultimate strength design methods in accordance with the codes, load definitions and load combinations specified in Appendix 3.8E.

Evaluations and modifications of existing steel structures and miscellaneous steel and design of new steel members added after July 1979, are done in accordance with the codes, load definitions and load combinations specified in Appendix 3.8E.

3.8.4.1 Description of the Structures

3.8.4.1.1 Auxiliary-Control Building

This building and associate structures are multistory reinforced concrete structures which provide housing for the Engineered Safety Feature Systems, etc., which are necessary for the two reactor units. Certain floors in the control bay, the Condensate Demineralizer Waste Evaporator Structure, and the roof of the fuel handling bay are supported by structural steel framing. Refer to Figures 3.8.4-1 through 3.8.4-9 for the general layout and configuration of the structure.

Control Bay Portion

Structure

The control bay portion is a multistory reinforced concrete structure that is built integrally with the Auxiliary Building portion as shown in Figure 3.8.4-8. The structure is separated from the Turbine Building by a 2-inch expansion joint filled with fiberglass insulation which prevents interaction of the two buildings when subjected to seismic motion. The structure was built in one stage, in advance of the Auxiliary Building, and was used initially as a foundation for the construction gantry crane.

Structural steel framing in the control bay consists of four steel framed bays of 25.0 feet by 45.0 feet at Elevation 729.0 and the entire floor at Elevations 741.0 and 755.0, with the exception of the two exterior bays on both ends of the building.

At Elevation 729.0 the two exterior bays on both ends of the building are pipe run areas and consequently the floor is 1-1/2-inch deep steel grating on steel beams. Reinforced concrete columns were used not only for structural requirements but also to reduce the beam size required.

The floor at Elevation 741.0 is a 1-1/2-inch-deep steel grating on steel beams. The floor acts as both the structural support and the horizontal restraint for the cable tray support systems between Elevation 729.0 and Elevation 755.0. An extensive horizontal bracing system is used at Elevation 741.0 to ensure that the floor maintains maximum horizontal restraint capacity with minimum member sizes.

The floor at Elevation 755.0, except in the two exterior bays on both ends of the building, is an 8-inch concrete slab supported on steel beams with intermediate columns on the building longitudinal centerline. The main control boards and instrument racks are located on this floor and the cable trays are supported below it.

Control Room Shield Doors

Two doors located inside the Main Control Room at floor Elevation 755.0 at doorways C-36 and C-54 provide radiation shielding for personnel inside the Main Control Room during a post-LOCA period. The doors normally remain open and are closed only in the event of a LOCA. The doors are manually operated from inside the control room and require no seals as they do not serve any pressure-confining function.

The two doors are identical except opposite hand and operate in opposite hand directions. Each door is a rectangular, structural steel frame with a skinplate on each side, thus forming a hollow box which is filled with lead shot to provide the required shielding.

Each door is suspended from above by two monorail-type trolleys operating on a standard structural I beam. The trolley closest to the leading edge of each door is of the geared, hand chain-driven type for opening and closing the door.

Manually operated turnbuckle dogging linkages are provided at the top and bottom of each door for firmly securing the doors in either the open or closed positions.

Equipment Hatch Covers

There are two equipment hatches provided at Elevation 708.0. The covers will normally remain closed at all times during plant operation. Each cover consists of an assembly of three welded structural steel sections held together by rows of steel screws with gaskets provided at each joint in the cover. In the closed position, the covers are secured by steel screws around the periphery. In the event that the hatch covers are removed, supports are provided around the hatch opening for the installation of handrails. These hatches were originally designed to be watertight, however this is no longer required.

Auxiliary Building Portion

Structure

The building is a multistory reinforced concrete structure that provides housing for the Engineered Safety Feature Systems, which are necessary to the two reactor units. The Auxiliary Building structure is attached to the Control Building and located between the Reactor Buildings as shown between column lines q and y, and A-1 and A-15, in Figure 3.8.4-8. In the final constructed state, the Control Building portion will act integrally with the Auxiliary Building portion. The Auxiliary Building is separated from the Reactor Buildings by a 1-inch expansion joint filled with fiberglass insulation that prevents interaction of the buildings when subjected to seismic motion. Seals are provided in the expansion joint to prevent the inleakage of either water or air since the Auxiliary Building, at times, serves as secondary containment. Below grade the seals, which consist of a polyvinyl chloride (PVC) material, are designed to withstand external water pressure, possible detrimental effects of the environment, the anticipated horizontal seismic movement of the buildings, and an assumed differential settlement of 1 inch between the buildings without loss of integrity.

The spent fuel pit and fuel transfer canal is housed within the Auxiliary Building. The massive reinforced concrete walls and slab are built integrally with the Auxiliary Building as illustrated by Figures 3.8.4-3 and 3.8.4-5. The spent fuel pit is equipped with two gates. The fuel transfer canal gate (0-GATE-079-0005) is used for dewatering the fuel transfer canal for maintenance. The cask loading pit gate (0-GATE-079-0006) is currently not utilized. The cask loading pit gate will be installed in its stored position only. The fuel transfer canal gate is installed or removed under balanced load. The spent fuel gates are illustrated by Figures 3.8.4-69 through 3.8.4-71.

Structural steel framing was used to support the Auxiliary Building roof over the area serviced by the main building crane because of the clear span requirements. This area is approximately 223.0 feet long and 80.0 feet wide. The roof is a reinforced concrete slab constructed on metal roof decking that is supported by steel purlins on welded steel trusses.

Railway Access Hatch Covers

Six hinged covers shown in Figures 3.8.4-10 and 3.8.4-11 combine to close the railroad access hatch opening in the floor of the Auxiliary Building at floor Elevation 757.0 with the six covers in the raised position, a clear opening of approximately 16-feet, 6-inches by 68-feet, 3-inches is provided over the railroad tracks. Spent fuel casks, new fuel shipments, and major items of equipment entering or leaving the Auxiliary Building above the Elevation 757.0 floor must go through this hatchway.

The hatch covers and their embedded frame provide a semi-airtight closure and operate in conjunction with the railroad access door to provide an airlock for the Auxiliary Building against a pressure differential of 1/4-inch of water.

An electrical interlock system is provided to interlock the operation of the access hatch covers with the railroad access door. Two limit switches, connected in series to provide redundancy, are provided with each hatch cover and arranged to trip when a hatch cover begins to open. The interlocking of these switches with switches on the door prevents the door from being opened when any hatch cover is open or partially open. In like manner, switches on the door prevent opening of any hatch cover when the door is open or partially open.

The hatch covers are required to maintain their integrity and Category I function only when closed. When closed, there is no load on the operating machinery and it has no function to perform. Therefore, the operating machinery is not considered as Category I.

Railroad Access Door

The railroad access door shown in Figures 3.8.4-12 through 3.8.4-15 for the Auxiliary Building provides closure for the access opening in the east wall at the railroad tracks which are at Elevation 729.0. The door and its embedded frame provide a semi-airtight closure and operate in conjunction with the railroad access hatch covers to provide an airlock for the Auxiliary Building previously described. (This door functions as an ABSCE airlock boundary - Door A112).

With the door fully opened, the clear opening in the wall is 16.0 feet wide and 20.0 feet high. All new or spent fuel shipments and major equipment entering or leaving the Auxiliary Building by truck or other means passes through this door.

The door and door track are constructed of welded steel. The door, rectangular in cross section, is constructed of horizontal and vertical members with diagonal bracing as required for strength and rigidity. The exterior side of the door is covered with a steel skin plate. The embedded frame for the door is constructed of welded steel and is anchored to the concrete.

The door seals in the closed position with the side and top seals compressed against sealing surfaces on the embedded frame and the bottom seal compressed against an embedded sill plate. A sloped track guides the door rollers and positions the door so that the top and side seals contact the sealing surfaces only when the door is in or near the closed position.

An electric hoist unit opens and closes the door by lifting and lowering it vertically through a slot in the Elevation 757.0 floor. The hoist unit is mounted on the inside wall above the door slot. The door passes through this slot, and extensions of the frame act as guides for the door in the raised position.

The area above the floor at Elevation 757.0, occupied by the hoist and the door in its raised position, is enclosed with an airtight structural steel enclosure with gaskets provided on the access covers necessary for servicing the hoist unit and door.

The access door and its frame are required to maintain their integrity and Category I function only when closed. When closed there is no load on the hoist unit, and it has no function to perform. Also, the hoist unit has no function to perform relative to the air tightness of the steel enclosure at Elevation 757.0. Therefore, the hoist unit is not considered as Category 1.

Manways in RHR Sump Valve Room

Two 54-inch-diameter manways, shown in Figure 3.8.4-16, and located at Elevation 698 feet 1 inch in the walls of the residual heat removal (RHR) sump valve room are provided for each reactor unit. The manways provide passageways through the walls of the sump valve room for workmen, tools, and equipment. The doors will normally be closed during plant operation, when reactor containment integrity is required, unless the doors are open to allow normal maintenance or monitoring activities. Although not required for containment integrity, the manways were designed to remain intact when the doors are open during an earthquake to prevent damage to other equipment in the vicinity of the manways.

Each manway consists of an embedded steel frame and a welded steel door. The door is secured in the closed position by bolts. The door is provided with slotted hinges to facilitate opening and closing and to allow for compression of the seals when the door is closed.

Pressure Confining Personnel Doors

This section covers the following pressure confining personnel access control doors located in the Auxiliary-Control Building. Door numbers listed for the doors are the designations used in Figures 3.8.4-17 through 3.8.4-20. The door details for specific heavy equipment type doors are shown in Figures 3.8.4-21 through 3.8.4-23. The door details for the remaining doors are shown on Reference [1].

1. The doors for stairs 7 and 8 penthouses at Elevation 772.0, doors A184 and A191.

2. The double doors to the personnel and equipment access rooms, Elevation 757.0 (one for each unit) doors A152* and A159*.

3. The double doors at the ice condenser equipment room, Elevation 757.0, door A155.

4. The double doors to the emergency gas treatment filter room, Elevation 757.0, door A158.

5. The doors to the Reactor Building access room at Elevation 757.0 (one for each unit), doors A156 and A157.

6. The doors for stairs 3 and 4 penthouses at Elevation 757.0, doors A154 and A173.

7. The double doors to the elevator shaft at Elevation 757.0, door A153.

8. The N-line control bay doors at Elevation 755.0 (two double doors with bidirectional pressure requirements), doors C36 and C54, and Elevation 729.0 (two double doors with bidirectional pressure requirements), doors C29 and C34.

9. The N-line instrument rooms access door at Elevation 708.0 (single door with bidirectional pressure requirements), door C26.

10. The double doors to the heating and ventilating spaces at Elevation 737.0 (one for each unit), doors A123* and A132*.

11. The door separating the Additional Equipment Building and the airlock at Elevation 737.0 (one for each unit, bidirectional pressure requirements), doors A183*, A192*, A214*, and A215*.

12. The door to the cask decontamination room, Elevation 729.0, door A115.

13. The doors in the X-line wall of the cask loading area at Elevation 729.0 (one single door A113*, and one double door, A114*).

14. Doors A161*, A162*, A64, A77, A216, A217, A94, A95, A96, A97, A98, A99, A164, A165, A166, and A167.

15. The doors to the main steam and feedwater valve rooms at Elevation 729.0 (one for each unit), doors A101* and A105*.

16. The double doors at main entrance from Service Building, Elevation 713.0, door A57*.

17. Annulus access door, Door A65,* and door to the Reactor Building access room, Door A64 at Elevation 713.0.

18. The airlock door to the radiochemical laboratory at Elevation 713.0, door A55*.

19. The door in the C-3 line wall leading to the instrument room at Elevation 708.0, door C20 (water tight and pressure confining).

20. The exterior double doors at the entrance to the Unit 1 Additional Equipment Building at Elevation 729.0, door A117.

21. The Auxiliary Building door that separates stairwell no. 11 from the Unit 1 ventilation and purge air room on Elevation 737.0, door A125*.

22. The Auxiliary Building door that separates stairwell no. 10 from the Unit 2 ventilation and purge air room on Elevation 737.0, door A130*.

23. The Auxiliary Building door that separates shutdown board room A from the personnel & equipment access airlock (that leads to the refueling room floor) on Elevation 757.0, door A151.

24. The Auxiliary Building door that separates shutdown board room B from the personnel & equipment access airlock (that leads to the refueling room floor) on Elevation 757.0, door A160.

25. The Auxiliary Building doors that separate the mechanical equipment room from the HEPA filter plenum room at Elevation 772.0, doors A212* and A213*.

26. The Auxiliary Building door that separates the upper portion of the refueling room from the airlock that leads to the Auxiliary Building roof at Elevation 786.0, door A206*.

27. The Auxiliary Building exterior door that separates the Auxiliary Building roof from the airlock that leads to the upper portion of the refueling room at Elevation 786.0, door A207*.

28. The Auxiliary Building door that separates the upper portion of the refueling room floor from the airlock that leads to the Auxiliary Building roof at Elevation 814.75, A208*.

29. The Auxiliary Building exterior door that separates Auxiliary Building roof from the airlock that leads to the upper portion of the refueling room floor at Elevation 814.75, door A209.*

30. The Control Building door that separates stairwell C1 from the corridor on the west side of the main control room at Elevation 755.0, door C37.

31. The Control Building doors that separate the main control room from the Auxiliary Building at Elevation 755.0, doors C49 and C50.

32. The Control Building door that separates stairwell C2 from the east side corridor at Elevation 755.0, door C-53.

33. The Control Building door that separates stairwell C2 from the corridor leading to the Technical Support Center at Elevation 755.0, door C60.

34. The exterior door at the entrance to the Condensate Demineralizer Waste Evaporator (CDWE) Building at Elevation 729.0, door DE1.*

35. The Auxiliary Building double doors at the entrance to the Service Building at Elevation 713.0, door A56.*

36. The Auxiliary Building door that separates the Auxiliary Building from the airlock that leads to door A55 at Elevation 729.0, door A60.*

37. The Auxiliary Building double doors that separate the heating and ventilating equipment rooms from the airlock that leads to door A123 at Elevation 737.0, door A122.*

38. The Auxiliary Building double doors that separate the heating and ventilating equipment rooms from the airlock that leads to door A132 at Elevation 737.0, door A133.*

39. The interior CDWE Building doors that combine with door DE1 to establish the necessary airlock at the exterior entrance to the CDWE Building, doors DE4* and DE5.*

40. The waste packaging room door that separates the waste packaging room from the railroad access room, double door A111.*

* These doors function as an ABSCE boundary. For airlocks, See Table 3.8.4-7b.

The doors are hinged, manually operated type metal doors, complete with frames and closers. The frames are either welded to plates, bolted to the concrete walls, or welded to embedded plates. Both single and double doors are involved. Double doors consist of an active and inactive leaf, with the active leaf being used for normal traffic. Doors A65, A55, C20 and C26 have a single skin plate with horizontal stiffeners. Door A57 is a double skinned door with horizontal and vertical stiffeners. All other doors are the flush type. All doors except A55, A57, A65, C20, and C26 are secured for tornado, annulus pressure drop or flood by means of a normal latching mechanism. Door A65 is secured by use of hand-operated dogs and doors A55, A57, C20, and C26 are secured by a dogging mechanism which is manually operated by a handwheel. All doors affected by tornadoes are secured during tornado watches and door A65 is secured during flood warnings.

During normal operation, the doors provide personnel and equipment access. Doors A55, A56, A57, A60, A101, A105, A111, A112, A113, A114, A122, A123, A125, A130, A132, A133, A183, A192, A206, A207, A208, A209, A214$^{\pm}$, and A215 are also components of the building airlocks which serve to maintain a slight negative pressure in the Auxiliary and Reactor Buildings. These doors are equipped with electrical interlocks to assure that one of each pair of interlocked doors is always closed except when under administrative control. Doors A161, A162, DE1, DE4, and DE5 are also components of the building airlocks; however, they are not electrically interlocked.

Doors A55, A56, A57, A60, A101, A105, A111, A112, A113, A114, A122, A123, A125, A130, A132, A133, A152, A159, A161, A162, A183, A192, A206, A207, A208, A209, A212, A213, A214, A215, DE1, DE4, and DE5 are components of the Auxiliary Building Secondary Containment Enclosure (ABSCE) boundary. These doors will be subjected to the slight pressure differential (1/2" water gauge) needed to establish the ABSCE.

Doors C36, C37, C49, C50, C53, C54, and C60 are components of the Main Control Room Habitability Zone (MCRHZ) boundary. These doors will be subjected to the slight pressure differential (1/8" water gauge) needed to establish the MCRHZ.

Waste Packaging Structure

The waste packaging area is a one-story reinforced concrete structure supported on crushed stone backfill placed in four-inch layers and compacted to a minimum of 70% relative density and is located on the north end of the Auxiliary Building as shown in Figures 3.8.4-3 and 3.8.4-5. The roof of the structure slopes about 24° and consists of a series of precast beams topped by 4 inches of poured-in-place concrete. The structure is separated from the Auxiliary Building by a 2-inch expansion joint filled with fiberglass insulation which prevents interaction of the two buildings when subjected to seismic motion.

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Condensate Demineralizer Waste Evaporator Structure

The Condensate Demineralizer Waste Evaporator Building portion is a two-story reinforced concrete structure that houses equipment necessary for processing condensate demineralizer wastes and for serving as a backup in processing floor drain wastes. The structure is supported on H-bearing piles and is located on the northeast side of the Auxiliary Building as shown in Figures 3.8.4-2 through 3.8.4-9. An access tunnel to the waste packaging area is separated from that structure by a 2-inch expansion joint filled with fiberglass material which prevents interaction of the buildings if subjected to seismic motion.

Additional Equipment Building Portion

The Additional Equipment Building portion consists of multi-story reinforced concrete structures, one for each unit, which accommodate accumulators for each unit and for the transfer of ice condenser equipment. The structures are located adjacent to the Reactor Buildings and near the north end of the main Auxiliary Building as shown in Figures 3.8.4-57 through 3.8.4-59. Each building is founded on sound rock and is separated from the Reactor Building by one inch of expansion joint material which prevents interaction of the building when subjected to seismic motion.

South Steam Valve Room

The South Steam Valve Room is an integral compartment of the Auxiliary Building portion of the Auxiliary-Control Building. The room is shown on Figures 3.8.4-3, 3.8.4-4, 3.8.4-8, and 3.8.4-49 through 3.8.4-49c. This compartment protects the isolation valves of the main steam lines, and other safety-related equipment, from the effects of tornados and earthquakes, as well as providing support for the main steam and feedwater pipes that exit from the Shield Building. The room is designed in accordance with the loads, load combinations, load factors, and allowable stresses given in Table 3.8.4-1.

Structural steel framing is used to support the roofing and roof decking of the valve room. The metal roof decking is designed to blow off to relieve pressure in the room.

Protection of the safety-related components within the room from horizontal tornado missiles is provided by the exterior walls of the Auxiliary-Control Building which includes one wall of the valve room. The other walls forming the room are interior to the building and are not subject to impact from horizontal tornado missiles.

Protection from vertical missiles is provided by the Reactor Building shield wall and by multiple levels of structural steel beams. The adjacent Reactor Building wall restricts the angles of possible missile entry. Since the roof of the steam valve room is more than 30 feet above plant grade, protection is required for the 1-inch diameter rod, missile A5 of Spectrum A (see Table 3.5-7 and Section 3.5.1.4). The multiple levels of structural steel beams partially screen safety-related components by further restricting possible missile entry angles. Small slender missiles such as the 1-inch diameter rod are known to be aerodynamically unstable and, therefore, tumble in flight. It is highly unlikely that a tumbling missile could strike safety-related equipment due to the limited pathways through the multiple levels of steel support structures. Therefore, adequate protection from vertical missiles is provided.

3.8.4.1.2 Diesel Generator Building

The building is a two-story rectangular reinforced concrete box-type structure that houses the diesel generators and associated auxiliary equipment. Interior walls of reinforced concrete separate the diesel generators into four compartments. The diesel fuel storage tanks are embedded in the base slab. The structure is supported on crushed stone backfill placed in four-inch layers and compacted to a minimum of 70% relative density. For general layout and configuration of the structure see Figures 3.8.4-24 through 3.8.4-29.

Diesel Generator Building Doors and Bulkheads

The four doors shown in Figures 3.8.4-30 through 3.8.4-32 at Elevation 742.0 in the north wall of the Diesel Generator Building along with removable bulkheads above the doors provide closures for the 11 feet - 10 inches high by 8 feet - 8 inches wide access openings to the diesel generator units. They provide for passage of large tools and repair parts for the diesel generators. The doors are normally closed and latched. The bulkheads are bolted in position and are removed only for major repair of the diesel generators. The doors and bulkheads are covered on the outside of the Diesel Generator Building by precast concrete bulkheads as shown in Figures 3.8.4-27 and 3.8.4-33. Together they protect the generators from damage by tornadoes, missiles, wind, snow, ice, and rain and form part of the security to prevent entry into the Diesel Generator Building by unauthorized persons. See Section 3.8.4.5.5.

Each bulkhead above the door is a structural steel frame 4 feet - 5-1/2 inches high by 9 feet - 5 inches wide. It is covered on both sides with a steel skin plate and provided with a crushable strip on the inner side along the top and sides. Turnbuckles support the bulkheads vertically, and they are held horizontally by bolted clamps at the sides and top.

Each door is 7 feet - 9-1/2 inches high and consists of two leaves that are manually operated and hinged at the outer sides to an embedded steel frame. The two leaves bear against steel bars at the outer sides, against an embedded angle at the bottom, against each other at the center, and against a steel angle at the top. The bars are welded to the embedded frame and the angle to the bulkhead above the door.

Each door leaf is a structural steel frame covered on both sides with a steel skin plate and provided with a crushable strip around its periphery where it bears against lateral support. Both leaves are provided with latches that are operated from the inside only.

The steel doors and bulkheads were provided in the original design of the Diesel Generator Building to protect the diesel generators from missiles B1, B2, and B3 of missile spectrum B in Table 3.5-8. In a review of the tornado protection criteria by the NRC in 1975 a determination was made that the level of protection provided by the doors and bulkheads should be upgraded to resist three additional missiles (B4, B5, and B6). The existing steel doors and bulkheads were found to be inadequate for the additional missiles. Therefore, precast concrete bulkheads were placed in front of the door openings to provide the additional missile protection. The precast concrete bulkheads consist of several individual sections stacked into place and bolted in position to the concrete walls. The precast concrete bulkheads are required to be in place when the diesel generators are operable.

The precast concrete bulkheads are 14 inches thick which is adequate to prevent penetration from missiles B4, B5, and B6. The 14-inch thickness is not sufficient to prevent some scabbing. However, the steel doors prevent the scabbed particles from entering the generator compartments. In the event the steel doors are open the scabbed particles will not reach the diesel generators due to the separation of the generators and doors. This protective scheme of preventing penetration but not scabbing is necessary due to the desire to keep the weight of the precast sections low to facilitate removal by field personnel.

3.8.4.1.3 Category I Water Tanks and Pipe Tunnels

There is one refueling water storage tank for each unit at Watts Bar Nuclear Plant. (The functional requirements for these tanks are discussed in Chapter 6.) Pipes extending from these tanks to the Auxiliary Building are housed in reinforced concrete tunnels which vary in width and height.

Refueling Water Storage Tanks (RWST)

As noted in Tables 3.2-1 and 3.2-2a, the RWST foundation is classified as Category I. The RWST is a Seismic Category I structure. A storage basin is provided around the tank to retain sufficient borated water in the event the tank is ruptured by a tornado missile or other initiating event. Details of the storage basin and the technical basis for it are discussed below. The RWST has a minimum required capacity of 370,000 gallons. The tank is a cylindrical vessel whose longitudinal axis is oriented in the vertical direction.

The end of the cylinder which forms the base or bottom of the tanks is completely enclosed with a 5/16-inch thick flat plate. The base of the tanks sits on a support structure consisting of a concrete slab 57 feet -0 inches in diameter and 3 feet -6 inches thick to which the tanks are attached at 48 anchor points. The perimeter ring plate is grout supported by the slab foundation. The interior is filled with approximately 6 inches of sand on top of the concrete slab, which supports the entire surface of the tank base or bottom plate. The top of the cylindrical section of the tank is sealed at the sidewall roof intersection using conical-shaped roofs whose apexes coincide with the tank's longitudinal axis.

A barrier wall is located around the RWST to protect the bottom three feet of the tank. This provides storage for borated water after a postulated rupture of the tank. A steamline break in one of the lines outside of the Reactor Building in close proximity to the RWST is the most demanding event that could require suction from the RWST and also credibly be associated with a rupture of the RWST. For this event a maximum of 20,000 gallons of borated water was initially analyzed by the NSSS vendor to provide the negative reactivity to keep the reactor subcritical. A lesser amount has been analyzed to maintain departure from nucleate boiling ratio (DNBR). The barrier wall, however, is designed to retain a volume in excess of this amount while supplying adequate net positive suction head to all ESF pumps. Figure 3.8.4-36a shows the distance (greater than 3 feet) between the top of the wall and the suction intake elevation. The tank is equipped with an atmospheric vent located at the peak or cone apex of the roofs. The vent is designed to pass a volume flow rate of air that is at least equal to the maximum withdrawal rate from the tank. Necessary precautions have been taken in the design of the vent to assure birds, animals, and/or other foreign objects, including, rain cannot enter the tanks. Tank physical dimensions and other parameters are given in Section 9.2.7. The foundations are shown in Figures 3.8.4-35 through 3.8.4-36C. Load combinations for the foundations are shown in Table 3.8.4-20.

Pipe Tunnels

The pipe tunnels housing the piping extending from the primary and refueling water tanks to the Auxiliary Building are concrete box-type structures that vary in width and height. Protection against flooding of the Auxiliary Building in case of a tank or pipe rupture is provided by walls that separate the tanks from the main tunnel. The layout and configuration of the tunnels are shown in Figures 3.8.4-35 and 3.8.4-36.

3.8.4.1.4 Class 1E Electrical System Manholes and Duct Runs

The manholes and duct runs shown in Figures 3.8.4-37 through 3.8.4-46 house the electrical cables that must remain in operation when flood levels rise above the plant grade and emergency power is required for safe shutdown of the plant. The manholes and duct runs lie in soil overburden that varies in depth from 30 to 35 feet.

The Category 1E manholes are rectangular box-type structures of reinforced concrete built below plant grade with an access shaft projecting above the surrounding soil. Category 1E manholes are equipped with watertight covers and sump pumps. A concrete cover is provided for protection from vertical missiles.

The duct runs connecting the manholes are continuous reinforced concrete beams with embedded electrical conduits. Duct runs are buried with a minimum of 18 inches of soil cover above them, except near the intake pumping station where they are exposed. Duct runs enter the manholes through sealed openings in manhole walls. A minimum of 6-1/2 inches of reinforced concrete is provided above the embedded conduits for protection from vertical missiles.

3.8.4.1.5 North Steam Valve Room

The structure, shown in Figures 3.8.4-47 through 3.8.4-49 is designed to protect the isolation valves of the main steam lines from the effects of tornadoes and earthquakes, as well as provide support for the valves and main steam pipes and feed water pipes that exit from the Shield Building. The structure consists principally of several high reinforced concrete walls anchored into a 7-foot-thick base slab that rests on a grillage of reinforced concrete foundation walls supported to rock. A 2-inch expansion joint separates the valve room from the Shield Building.

Structural steel framing is used to support roof decking of the valve room. The metal roof decking is designed to blow off when the internal pressure at the roof reaches 72 pounds per square foot.

Protection of the components from horizontal tornado missiles is provided by the walls of the steam valve room. Protection from vertical tornado missiles is provided by a 24-inch-thick concrete awning which covers approximately one-third of the roof, by the Reactor Building shield wall, and by multiple levels of structural beams. The concrete awning and the adjacent Reactor Building wall, which extends 89 feet above the decking, restrict the angles of possible missile entry. Since the roof of the steam valve room is more than 30 feet above plant grade, protection is required for small, slender missiles, such as the 1-inch-diameter roof missile A5 of Spectrum A (see Table 3.5-7 and Section 3.5.1.4). (See also Figures 3.8.4-49B through 3.8.4-49C).

The main steam safety and relief valves are located about 21 and 17 feet, respectively, below the decking. Four 30-inch-wide flange beams and numerous 8-inch steel channels serve to partially screen the safety and relief valves from tornado missiles by further restricting possible entry angles. The largest pathway through the wide flange beams and the channels is approximately 1.5 feet by 2.5 feet (two such pathways) in plan area. Small slender missiles such as the 1-inch diameter rod are known to be aerodynamically unstable in flight and, therefore, tumble during flight. It is highly unlikely that a tumbling missile could follow the pathways discussed above without being deflected. Therefore, the main steam safety and relief valves are adequately protected from vertical tornado missiles. (See also Figures 3.8.4-49B through 3.8.4-49C).

The main steam, main feedwater, and feedwater bypass isolation valves are located below the safety and relief valves and are further protected from missile damage by five levels of wide flange beams (33-inch to 8-inch size) provided for pipe break restraint and support functions. There are no practical pathways by which tornado missiles could reach these valves. (See Figures 3.8.4-48B through 3.8.4-49C).

3.8.4.1.6 Intake Pumping Station and Retaining Walls

Pumping Station

The intake pumping station is a cellular box-type, reinforced concrete, waterfront structure founded on bedrock and partially backfilled on three sides. On the land side, retaining walls hold back the fill to Elevation 710.0. Permanent openings are provided in the reservoir side of the pumping station to allow flooding of any unwatered pump wells when the reservoir level exceeds Elevation 690.0. The essential raw cooling water (ERCW) pumps, fire protection pumps, and screen wash pumps are located on the upper deck at Elevation 741.0 above the maximum possible flood and are covered by a roof. This deck is completely enclosed by a concrete wall 13 feet high. A wall also supports the structural steel grillage system, shown in Figure 3.8.4-68, which provides tornado missile protection to the equipment below. The raw cooling water pumps are located on the deck at Elevation 728.0, which is below the maximum probable flood, but are not required for maintenance of plant safety. The mechanical and electrical equipment are located on the floors at Elevation 722.0 and 711.0, respectively. A permanent pedestal crane is mounted above the upper deck at Elevation 754.0 for handling of equipment. The structural outline is shown in Figures 3.8.4-50 and 3.8.4-51.

Traveling Water Screens

As shown in Figure 3.8.4-52, the screens are of the single or through flow, automatic cleaning type with a nominal basket width of 4.0 feet.

The capacity of each screen, with a head loss of 6 inches for a clean screen and minimum water depth, is approximately 25,000 gallons per minute at a water velocity of 2.0 feet per second. Basket travel speed is about 10 feet per minute. Removal of trash and refuse from the basket screens is by water sprays located in the head frame.

The drive motor for each screen is sized to start the screen with water at Elevation 737.5 and a head loss of 2-feet, 6-inches. All drive components are rated for continuous duty and are suitable for outdoor service.

Timers provided in the control circuits for the screens function to operate the screens for 18 minutes every 60 hours to prevent "freezing" of the machinery parts from nonuse. This provides assurance that the screens are in an operable condition at all times.

The heads of the screens, including drive components, are located above the maximum possible flood level. The screens are designed to operate during any flood, including a maximum possible flood, with water to Elevation 739.2 and 5' 0" head loss.

The four screens are identical with two screens provided for each of the two supply trains at the intake station. Each of the two screens on each supply train has sufficient capacity to screen the total water required for one train. The capacity of one supply train is sufficient to supply all water required for the ERCW during a LOCA.

Starting of the screens by pressure switches on the spray water assures that adequate spray water for removal of trash is available when the screens are started. This greatly reduces the possibility of carrying trash over the screens and into the screened water.

Watertight Doors

There are two watertight doors provided on Elevation 722.0, identified as W10A and W10B. These doors prevent water from the room containing one train of the ERCW System from entering the room containing the other train.

Concrete Retaining Wall

The earthfill is hold back by two concrete retaining walls from the pumping station to a point 32 feet from the pumping station. The concrete walls are keyed into rock and are separated from the pumping station by expansion joints. For outline of walls, see Figure 3.8.4-53.

Sheet Pile Retaining Walls

The sheet pile walls are parallel and extend from each end of the back of the pumping station toward the main plant. These parallel walls contain earthfill to Elevation 710.0 and project above the sloping grade a maximum of 29 feet at the pumping station. For layout of walls, see Figures 3.8.4-54 and 3.8.4-55.

3.8.4.1.7 Miscellaneous Essential Raw Cooling Water (ERCW) Structures

Slabs and Beams Supporting ERCW Pipes

At the Intake Pumping Station, the ERCW pipes are supported on a reinforced concrete slab. The slab is approximately 8 feet below grade and 50 feet above bedrock. The slab is supported by a bracket on the pumping station wall, bearing piles, and undisturbed earth. Structural separation from the pumping station is provided by 1/2-inch of expansion joint material. The slab is shown in Figure 3.8.4-56.

The ERCW pipes at the Diesel Generator Building are encased in concrete beams for support. The pipes are separated from the beams by insulation and the beams are separated from the Diesel Generator Building by expansion joint material. The beams are supported by brackets on the Diesel Generator Building and by Class A backfill. The beams are shown in Figure 3.8.4-56b.

Discharge Overflow Structure

The discharge overflow structure is a reinforced concrete box-type structure supported on granular fill material placed over basal gravel. The function of the discharge overflow structure is to provide for the normal flow rate discharge of the ERCW System without unacceptable back pressure if the downstream pipes are blocked and to permit flow to the holding pond under normal conditions. The structural outline is shown in Figure 3.8.4-46a.

Standpipe Structures

The two standpipe structures are mass reinforced concrete structures placed on firm granular material. The structures have backfill on four sides for the first 8 feet of height and extend 17 feet above grade. The function of these structures is to protect the standpipes from tornado-generated missiles. The structures are shown in Figure 3.8.4-56a.

Valve Covers

These structures consist of reinforced concrete slabs covering the valves in the ERCW pipes. The slabs are located at grade above the pipes and are supported by either the missile protection slab and/or backfill. The slabs have small openings with precast concrete covers above each valve stem. The openings in the missile protecting valve covers provide immediate access to the valves in an emergency. The structures are shown in Figure 3.8.4-56c.

3.8.4.1.8 Additional Diesel Generator Building

The Additional Diesel Generator Building is located 349.25 feet north of the centerline of the Reactor Buildings and 54.5 feet west of the centerline of the Unit 1 Reactor Building. It is a two-story rectangular, reinforced concrete, box-type structure which houses the additional diesel generator unit and its auxiliary equipment. The building is 96 feet long by 53 feet wide and is supported entirely on end bearing structural steel H-Piles as shown in Figure 3.8.4-72. The base slab is 12 feet thick with the finished floor at Elevation 742.0. The diesel fuel storage tanks are embedded in the base slab. For general layout and configuration of the building see Figures 3.8.4-73 through 3.8.4-80.

Additional Diesel Generator Building Doors and Bulkheads

The two large door openings, shown in Figures 3.8.4-74 and 3.8.4-75, in the north and east exterior walls of the building at Elevation 742.0, provide for passage of large tools and repair parts for the additional diesel generator unit and its auxiliary equipment. Removable missile barriers of precast, stackable concrete sections are installed and bolted into position in front of these doorways to protect safety-related equipment from tornado wind and missiles. These missile barriers also form part of the Security System by preventing unauthorized entry into the building through these doors. Due to the presence of the precast concrete missile barriers in front of the doorways, the equipment doors do not need to function as missile barriers and therefore standard double doors are used. The precast concrete missile barriers will be removed only for major repair of the diesel generator.

3.8.4.2 Applicable Codes, Standards, and Specifications

Unless otherwise indicated in the UFSAR, the design and construction of the Category I structures other than the primary containment and interior structures are based upon the appropriate sections of the following codes, standards, and specifications. Modifications to these codes, standards, and specifications are made where necessary to meet the specific requirements of the structures. These modifications are noted in Sections 3.8.3.2, 3.8.4.3, 3.8.4.4, and 3.8.4.6. Where date of edition, copyright, or addendum is specified, earlier versions of the listed documents were not used. In some instances, later revisions of the listed documents were used where design safety was not compromised.

3.8.4.2.1 List of Documents

1. American Concrete Institute (ACI)

ACI 214-77	Recommended Practice for Evaluation of Strength Results of Concrete
ACI 318-63	Building Code Requirements for Reinforced Concrete. (See Section 3.8.4.2.2 for basis for use of this section.)
ACI 318-71	Building Code Requirements for Reinforced Concrete
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting
ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete

ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI 349-76	Code Requirements for Nuclear Safety Related Concrete Structures, Appendix C only
ACI 531-79	Building Code Requirements for Concrete Masonry Structures

2. American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," adopted February 12, 1969, as amended through June 12, 1974, except welded construction is in accordance with Item 5 below.

3. Steel Structures Painting Council, Surface Preparation Specification No. 2, 'Hand Tool Cleaning.'

4. American Society for Testing and Materials (ASTM), 1971 Annual Book of ASTM Annual Standards.

5. American Welding Society (AWS)

   'Structural Welding Code,' AWS D1.1-72, with revisions 1-73 and 2-74 except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders.

   Visual inspection of structural welds will meet the minimum requirements of Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 as specified on the design drawings or other engineering design output. See Item 18 below.

   'Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Connections,' AWS D12.1-61.

6. American Gear Manufacturers Association, Standards for Helical and Herringbone Gears.

7. Uniform Building Code, International Conference of Building Officials, Los Angeles, 1970 Edition.

8. Southern Standard Building Code, 1969 Edition, 1971 Rev.

9. 'Nuclear Reactors and Earthquakes,' USAEC Report TID-7024, August 1963.

10. American Society of Civil Engineering Transactions,
Vol. 126, Part II, Paper No. 3269, 'Wind Forces on Structures,' 1961.

11. Code of Federal Regulations, Title 29, Chapter XVII, "Occupational Safety and Health Administration, Dept. of Labor", Part 1910, 'Occupational Safety and Health Standards.'

12. Regulatory Guides (RG)

RG 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures
RG 1.13	Fuel Storage Facility Design
RG 1.15	Testing of Reinforcing Bars of Category I Concrete Structures
RG 1.31	Control of Stainless Steel Welding
RG 1.55	Concrete Placement in Category I Structures

13. Section deleted in initial UFSAR

14. TVA Reports

TVA-TR-1, Topical Report, TVA-TR-1 Protection Against Pipe Whip Resulting From Piping Ruptures, 1973.
TVA-TR-78-4 Design of Structures for Missile Impact, 1978.
CEB-86-12 Study of Long Term Concrete Strength at Sequoyah and Watts Bar Nuclear Plants
CEB-86-19-C Concrete Quality Evaluation

15. National Electrical Manufacturers Association, Motor and Generator Standards MG-1, 1970 Edition.

16. Structural Engineers Association of California, "Recommended Lateral Force Requirements and Commentary," 1968 Edition.

17. National Fire Protection Association (NFPA) 30.

18. Nuclear Construction Issues Group (NCIG)

NCIG-01, Revision 2	-	Visual Weld Acceptance Criteria (VWAC) for Structural Welding At Nuclear Power Plants
NCIG-02, Revision 0	-	Sampling Plan for Visual Reinspection of Welds

The referenced NCIG documents may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS.

NCIG-02, Revision 0, was used as the original basis for the Department of Energy (DOE) Weld Evaluation Project (WEP) EG&G Idaho, Incorporated, statistical assessment of TVA performed welding at WBNP. Any further sampling reinspections of structural welds subsequent to issuance of NCIG-02, Revision 2, are performed in accordance with NCIG-02, Revision 2 requirements.

The applicability of the NCIG documents is specified in controlled design output documents such as drawings and construction specifications. Inspectors performing visual weld examination to the criteria of NCIG-01 are trained in the subject criteria.

3.8.4.2.2 Basis for Use of the 1963 Edition of ACI 318

The reason for using the 1963 edition of the ACI 318 Code was that much of the Watts Bar Plant was a duplicate of the Sequoyah Plant, for which structures were designed using the 1963 edition. On that basis, design computations for the Sequoyah Plant were the initial design computations for the Watts Bar Plant.

In some instances, structures could not be duplicated and new design computations were prepared for these structures with the designs in accordance with the ACI 318-71 Code. Within duplicate structures, where loading changes required investigation of the Sequoyah design for an element of the structure, and the result was a change in member size or reinforcement requirement, the redesign for the member was in accordance with the ACI 318-71 Code.

The duplicate structures are the Auxiliary-Control Building and the Diesel Generator Building.

The differences between the two code editions for working stress design were examined and the conclusion was that none of these differences significantly affect the safety of the plant.

The following are comparisons of the important parts of the code which affect safety:

Flexure	ACI 318-63 W. S. Design	ACI 318-71 Alt. Sect 8.10
Concrete Stress	0.45 f'_C	0.45 f''_C
Steel Stress	24,000 psi	24,000 psi

<u>Tied Columns</u>		
Maximum P	$0.212\ f'_C A_g$	$0.268\ f'_C A_g$
Balanced P	$+20{,}400\ A_s$ $\sim 0.15\ f''_C A_g$	$+18{,}900\ A_s$ $\sim 0.16\ f''_C A_g$
<u>Concrete Shear Stress</u>		
Beam V_C	$1.1\ \sqrt{f'_C}$ 1	$1.1\ \sqrt{f'_C}$ 2
Slab V_C	$2\ \sqrt{f'_C}$ 3	$2\ \sqrt{f'_C}$ 4
<u>Bearing Stress</u>		
Concrete	0.25 to 0.375 f''_C	0.3 to 0.6 f''_C

One of the apparent major differences between the two codes is the method by which rebar contact splice lengths are calculated. The 1971 code does require longer splice lengths where bars are spaced closer than 6 inches, but it is TVA practice to specify bar spacing of 6 inches or greater for bars which must be lapped by a contact splice. Listed below is a comparison considered to be typical in TVA practice. The results of this comparison show that the use of the 318-63 Code does not significantly affect the safety of the plant.

<u>Contact Splice Lengths</u>

$f''_C = 3{,}000$ psi

$f_y = 60{,}000$

No. 11 bar in tension, top bar

Bar spacing, 6 inches to 17 inches

	318-63 W. S. Design	318-71 Alt. Sect.8.10
A_s Provided Equals That Required by Computation		
Less than 1/2 bars spliced at a given section	102 inches	99 inches
More than 1/2 bars spliced at a given section	Not specified TVA practice 1.2x102=122 inches	129 inches
A_s Provided Equals Twice That Required by Computation		
Less than 3/4 bars spliced at a given section	102 inches	76 inches
More than 3/4 bars spliced at a given section	102 inches	99 inches

3.8.4.3 Loads and Loading Combinations

3.8.4.3.1 Description of Loads

See Tables 3.8.4-1 through 3.8.4-13, Tables 3.8.4-15 through 3.8.4-23, and Appendix 3.8E for the loads for other Category I structures. Other Category I structures are subject to the same natural phenomena and basic dead, live, and earth pressure loading as described for the Shield Building in Section 3.8.1.3. In addition to active earth pressure loading described in Section 3.8.1.3, the other Category I structures are designed for at rest and passive earth pressures where applicable.

Construction loads differ for the Auxiliary Building because of the multistory effect of shoring from one floor to the next and the construction crane loading on the Control Building portion.

The maximum temperature gradient for walls above grade with exterior exposure is the same as the normal operating temperature gradient of the Shield Building. The spent fuel pit and fuel transfer canal require additional temperature considerations. Under accident conditions the water was assumed to reach 212°F in 8 hours with the inside building temperature initially at 60°F. The normal temperature of the water in the fuel pit and canal is 120°F.

Hydrostatic pressure loads in the fuel pit and canal vary with water levels in the pit, cask loading area, and canal. The water in the cask loading area is normally maintained at approximately the same level as the water in the spent fuel pool. The canal may be emptied.

The wind and tornado loading are described in Section 3.3. Blowout panels are necessary to restrict the maximum internal pressure to 1.25 psi above the Elevation 757.0 floor in the Auxiliary Building as shown between column lines t and y, and A-5 and A-11 in Figure 3.8.4-4 and 3.8.4-5.

The load associated with supports for cable trays, piping systems, and other fastenings to interior reinforced masonry walls was restricted to a maximum of 20 psf over one face of the wall (i.e., 10 psf on each face).

A 1730 psf surcharge loading was applied to the A-1 and A-15 line walls as a construction loading in the Auxiliary Building.

3.8.4.3.2 Load Combinations and Allowable Stresses

See Tables 3.8.4-1 through 3.8.4-13, Tables 3.8.4-15 through 3.8.4-23 and Appendix 3.8E for the loading combinations and allowable stresses for concrete, miscellaneous steel, and structural steel.

The normal allowable stresses of ACI 318-63 and ACI 318-71 were used for the basic loading combinations of dead, live, earth pressure, hydrostatic ground water to Elevation 710.0 (or full pool water levels in the spent fuel pit) and effects of normal temperature gradients.

For additional loads such as induced moments or shears resulting from operating basis earthquake (OBE), accident pressure loading caused by a LOCA or steam pipe rupture and thermal effects corresponding to the accident condition, a 25% increase in steel stress was allowed with concrete stresses restricted to normal allowables.

For construction loading instead of normal live loading or for hydrostatic pressure to Elevation 724.4, a 35% increase in both steel and concrete stresses was allowed.

For the combination of the basic loads with safe shutdown earthquake (SSE) effects, or tornado wind loads and associated missiles, or maximum possible flood loads, or impact loadings from jet impingement or jet loading on pipe restraints in conjunction with accident pressures a 67% percent increase in concrete stresses was allowed with steel stresses allowed to reach 0.9 of yield.

The maximum lateral forces generated by the SSE are transmitted to the base through shear walls which are designed in accordance with Section 2631 (c) of the "Recommended Lateral Force Requirements and Commentary," of the Seismology Committed, Structural Engineers Association of California, 1968.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Auxiliary-Control Building

Control Bay Portion

This concrete structure was designed in accordance with the ACI Building Code 318-63 using the elastic working stress theory. The loads, loading combinations, and allowable stresses used are as given in Section 3.8.4.3.2.

The control bay was designed as an independent structure. A standard frame analysis was performed on the building in the design of the main structural walls and a separate analysis was performed for each loading combination. The stage that construction of the building's component walls, slabs, and columns would have progressed by the time of the application of a particular loading, was taken into account and reflected accordingly in the model frame.

The floor slabs at Elevations 708.0 and 729.0 were designed by ICES STRUDL-II, Volume I program as flat slabs restrained at the exterior structural walls and supported on concrete columns. At Elevation 755.0 the two exterior bays on both ends of the building were designed by ICES STRUDL-II, Volume II program to resist a break in the main steam lines below. The roof slab was designed as a one-way slab spanning between the walls at column lines n and q, as shown in Figure 3.8.4-8. These walls act as shear walls in the event of east-west seismic motion or any other east-west lateral force, with the walls along column lines C1, C3, C11, and C13, as shown in Figure 3.8.4-3 acting as shear walls for north-south lateral forces. The roof slab and floor slabs act as diaphragms. The columns and main structural walls transmit vertical load to the base slab.

The floors and walls of the Auxiliary Building are continuous with the control bay north wall. Dowels and shear keys were provided in the wall in order to provide for this structural continuity.

Procedures used to design the structural steel framing were based on simple beam and column construction as covered in AISC 'Specification for the Design, Fabrication and Erection of Structural Steel for Buildings,' Part 1, with type 2 framing connections. The beams at Elevation 755.0, between column lines C3 and C5, and between column lines C9 and C11, were designed to function compositely with the concrete slab through the use of headed concrete anchor studs welded through steel decking to the top beam flanges. The beam-to-beam and beam-to-column connections were typical AISC double angle connections as required by the beam reactions, using either rivets or high strength bolts. Between column lines C5 and C9 the beams were not designed to function compositely. For column line references, see Figures 3.8.4-3 and 3.8.4-4. In these areas horizontal bracing is used to resist the horizontal forces for the support of components such as cable trays, conduit, and pipe supports. At Elevation 741.0, there were special connections required that were either bolted or welded in accordance with the codes, standards, and specifications identified in Section 3.8.4.2.1. Transfer of loads into the concrete structure was through bearing plates.

Reinforced concrete partition walls are shown in key plan on Figures 3.8.4-60 through 3.8.4-65. These walls were analyzed as free at the top, fixed at the base, and were designed to resist seismic stresses. A minimum steel percentage of 0.1 was provided horizontally for each face. A 2-inch space was left between the top of the walls and the bottom of the slab or beam above, in order to ensure that the walls do not act as structural components of the building frame. Each side of this space was filled with a minimum of 2-inch-wide grout.

All reinforced masonry walls are designed in accordance with ACI 531-79 and NUREG-0800, Section 3.8.4, Appendix A.

<u>Control Room Shield Doors</u>

The doors were designed assuming that the entire dead load is carried by the two vertical members in the door directly under the trolleys with the load from the lead shot acting as a fluid pressure load.

The end panels were designed as a fixed beam with uniform load, while the skin plate was designed as a square flat plate stayed at the four corners. The top and bottom members of the door were considered as simple beams.

Earthquakes are the only natural environmental condition which applies to the doors. Being inside the control room, the doors are protected from outside elements.

For design, the earthquake loads for the various parts consisted of the loads produced by an OBE or an SSE. Accelerations due to a SSE are greater than those due to an OBE by a factor of 2. The doors were designed to maintain their structural integrity based on loading conditions resulting from an OBE or an SSE.

Earthquake loads used in design of door and dogs were the greater loads produced by OBE or SSE having peak accelerations at floor Elevation 755.0 in the Control Building.

These accelerations were used as static loads for determining component and member sizes. After establishing the component and member sizes, a dynamic analysis, using appropriate response spectra, was made of the door and dogs. This analysis indicated that additional stiffness was required in order to limit the loads on the dogging linkages such that allowable stress would not be exceeded. After adding diagonal stiffeners, another dynamic analysis was made and it was determined that the allowable stresses had not been exceeded. The door assembly was qualified to the Set "B" response spectrum.

Equipment Hatch Covers

The covers were designed to resist a downward uniform pressure created by a 3-foot head of water caused by a Condenser Circulating Water System (CCWS) rupture in the Turbine Building in addition to the dead load of the structural steel components. The covers were also investigated for a uniform pressure differential of 3.0 psi upward caused by the rapid depressurization during the occurrence of a tornado.

These hatches were originally designed to be watertight which no longer is a requirement. Since the original hatches remain in place, the loads to which the hatches were designed remain the same.

The covers were designed as a structure supported around its periphery. The structural steel members were designed as simply supported beams with uniformly distributed loads. Loads from the covers are transmitted to embedded frames which are continuously supported by concrete. The embedded structural steel frames are solidly anchored in the concrete by headed steel studs welded to the frames.

Auxiliary Building Portion

This concrete structure is designed according to the ACI Building Code 318-63 and the stresses are determined by the working stress method for the principal design cases as shown in Section 3.8.4.3.2. Stresses resulting from the static analysis are combined by the method of superposition with stresses resulting from moments, shears, deflections, and accelerations determined by the dynamic earthquake analysis described in Section 3.7.2. The exterior concrete walls above grade Elevation 728± are designed to resist the tornado-generated missiles as described in Section 3.3.

The condition of rapid depressurization during a tornado is provided for in the following manner. The exterior part of the building is designed for an internal positive pressure of 3 psi occurring in 3 seconds with the following exceptions:

1. The area above the refueling floor at Elevation 757.0 as illustrated by Figure 3.8.4-3, is designed with blowout panels which open at 0.25 psi. The roof and exterior walls of the spent fuel pool room and cask loading area were evaluated for the effective tornado-generated pressure differential and were found to be within allowable stress limits.

2. The area below the Elevation 786.0 roof is vented from openings in the roof as illustrated in Figure 3.8.4-8. The roof and walls housing this area are nevertheless designed for 3 psi. The floor at Elevation 772.0 below this roof is also designed for an uplift of 3 psi recognizing the venting of the area above this floor.

3. The heating and ventilating rooms at Elevation 737.0 (see Figure 3.8.4-3) are vented by the air intakes on the exterior walls. This results in the floor, roof, and interior walls of these rooms being designed as exterior member for 3 psi pressure.

The exterior walls below grade Elevation 728± are designed for earth pressures. The exterior walls on the east and west ends of the Auxiliary Building are designed as cantilevered retaining walls from Elevation 690 to Elevation 711±. These walls are built early before any adjacent walls and slabs to allow the construction field force to backfill and have early access to the area at Elevation 711±. The lateral earth pressure is calculated using Coulomb theory and values are given in Section 3.8.1.3.

The exterior walls north of the Shield Buildings with the buttress walls framing into them, as shown In Figure 3.8.4-2, are designed as cantilevered retaining walls from Elevation 690 to Elevation 727± to allow for earth backfill and placement of the Elevation 729 slab on grade.

Horizontal seismic forces are resisted by shear walls with the floor slabs and roof acting as diaphragms. Only those walls parallel to the seismic motion are assumed to resist that motion. The total shear at any level is proportioned among the shear walls in accordance with the method in Reference [3].

For the Safe Shutdown Earthquake, an allowable ultimate shear stress of $5.4\Phi\sqrt{f'_c}$ is used. This is the value specified in the SEAOC Code in Section 2631 (c) for walls with a height to width ratio less than one, as is the case for this structure. For the operating Basis Earthquake, an allowable value of one-half of the above is used.

Main steam and feedwater water pipes penetrate the exterior walls of the valve rooms on the south sides of the Shield Building. These penetrations furnish pipe restraints through flued heads embedded in the walls. The flued heads restrict concrete surface temperatures to a maximum of 150° F.

The primary structural support system is designed as a flat slab floor system with concrete columns. Large openings that required separate design are framed with beams. The thickness for many slab sections throughout the building is determined by shielding requirements. The general thickness and live load requirements for different slab areas are shown on Figure 3.8.4-9.

The major portions of the building slabs are designed by the ICES STRUDL-II, Volumes I and II. Moments and shears from small frames, beams, and one-way slabs are designed by the moment distribution method. Where slabs act as two-way slabs due to walls or beams below, moments and shears are determined by use of method 2 of Appendix A in ACI Code 318-63. The effects of the relative deflections of the supports and the effects of column shortening were taken into account in the design of all slabs.

The minimum percentage of reinforcing in the slabs is 0.15% in the top face and 0.18% in bottom face.

The roof slab at Elevation 786.0 is designed for 3 psi uplift pressure as a flat slab using the ICES STRUDL-II, Volume I computer program.

The roof at Elevation 801.0 is also designed for 3 psi uplift pressure using the ICES STRUDL-II, Volume II Finite Element Method.

In the interior of the building there are many areas around equipment that require shielding which is provided by poured-in-place concrete walls. To permit equipment installation the construction of shielding walls is delayed until the building frame and floor construction is completed and equipment is installed. These shield walls contain minimum steel percentages in the horizontal and vertical directions as specified by the TVA Temperature and Shrinkage Standards and the ACI Code 318-63, Section 2202 (f). These walls were checked for stresses resulting from seismic loading; however, seismic stresses did not control.

Reinforced concrete partition walls are shown in plan on Figures 3.8.4-60 through 3.8.4-65. These walls were analyzed as free at the top, fixed at the base, and were designed to resist seismic stresses. Minimum steel percentages provided were the same as those described for the shield walls. A 2-inch space was left between the top of the walls and the bottom of the slab or beam above in order to ensure that the walls do not act as structural components of the building frame. Each side of this space was filled with a minimum of 2-inch-wide grout.

The thick concrete walls of the spent fuel pit and transfer canal are required for shielding. They are shown in Figure 3.8.4-3. The walls are supported by a concrete base slab, which is approximately 27 feet thick. The walls and base slab are built integrally with the slabs and walls of the Auxiliary Building. A structural wall separates the cask loading area from the spent fuel storage area. The design of the pool walls take into account hydrodynamic effects of the water caused by earthquake and temperature effects caused by failure of the Spent Fuel Cooling System. This structure was designed by moment distribution methods. The stresses in the walls between the spent fuel pit and fuel transfer canal and those between the spent fuel pit and cask loading area were checked by the ICES STRUDL-II, Volume II computer program to determine the effect of the slot in the walls.

Procedures used to design the structural steel framing were based on simple beam and column construction as covered in AISC 'Specification for the Design, Fabrication and Erection of Structural Steel for Buildings,' Part 1, with Type 2 framing connections.

Railroad Access Hatch Covers

Structural members for the covers were designed as simple beams. Members of the embedded frame were considered as being rigidly supported by concrete. Loads from the embedded frame are transferred to the concrete by embedded anchors.

The earthquake forces, specified as follows for design, were determined by dynamic analysis including amplification through the supporting structure.

Accelerations for the SSE were used as static loads for determining component and member sizes. After establishing the component and member sizes, a dynamic analysis, using appropriate response spectrum was made of the covers to determine that allowable stresses had not been exceeded.

Railroad Access Door

The horizontal structural members of the door were designed as simple beams with uniformly distributed loads. The end reactions from these members were then transferred to the door end posts as concentrated loads located between rollers. As a conservative design, it was assumed that one roller was not in contact with the track and that the loading from the two horizontal members with the highest reactions was carried by the two adjacent rollers.

The skin plate for the door was designed, without regard to support of the plate from diagonal stiffeners, for the largest open rectangle within the structure. The plate was assumed to be a rectangular diaphragm with fixed edges.

The embedded door frame is rigidly supported by concrete. The portions of the frame which form the door track were designed as cantilever members with loading as applied by the door rollers.

The structural members of the steel enclosure above the door were designed as simple beams and the hoist supports as cantilevers from the Auxiliary Building wall.

Earthquake loads used in design of the door, frame, and track were the loads produced by a SSE having peak accelerations at ground level Elevation 729.0, which is the bottom of the door.

Earthquake loads used in design of the hoist supports and enclosure were the loads due to accelerations at the hoist platform, Elevation 773.0, produced by a SSE. These accelerations were determined by dynamic analysis of the Auxiliary Building structure. These accelerations were used as static loads for determining component and member sizes. After establishing the component and member sizes, a dynamic analysis, using appropriate response spectra, was made of the door, embedded frame, door track, and hoisting unit enclosure to determine that allowable stresses had not been exceeded.

Manways in the (RHR) Sump Valve Room

In the closed position, each door was considered as a structure supported around the periphery. In the open position, each door was considered as a cantilevered structure with the hinges and hinge anchorages being designed for their loading from the door in the open position. Each embedded frame was considered as being rigidly supported by concrete. Loads from the embedded frame are transferred to the concrete by embedded anchors.

Earthquake loads used in designing the manways were the forces due to accelerations determined for the sump valve room walls at the center of the manways by dynamic analysis of the Auxiliary Buildings for an OBE or SSE. These forces were used as static loads since the manways are rigid and firmly secured to the walls when closed.

Pressure Confining Personnel Doors

Structural members for the doors, in the closed position, were designed as simple beams with end reactions carried by the outside members to the frames which were considered as being rigidly supported by concrete. Loads are transferred to the concrete through embedded anchors or bolt anchors.

In the open position, the doors were designed as cantilever structures with resultant concentrated loads being used for design of the hinge members. For design, the earthquake loads for the various doors consisted of the loads produced by an OBE or SSE.

Earthquake forces due to building accelerations at the elevation of the center of gravity of the various doors were used as static loads for determining door component and member sizes. The building accelerations were determined by dynamic analysis including amplification through the supporting structures. After establishing the component and member sizes, a dynamic analysis, using appropriate response spectra, was made of the doors to determine that allowable stresses had not been exceeded.

Fuel Pool Gates

The gates are designed for a waterhead load of 25.0 feet imposed from the fuel pool side as measured from the centerline of the horizontal bottom seal to the normal pool level at Elevation 749.13. The gates are constructed of welded corrosion resistant steel. When dewatering the fuel transfer canal inflatable elastomer seals provide a near watertight seal between the skin plate and the pool wall liner face.

The gates have been analyzed for the effects of the OBE and SSE for both the operating and stored position. The gates are designed to maintain their sealing and structural integrity during and after an OBE or SSE. Earthquake loading considers simultaneous vertical and horizontal dynamic forces that act on the gates when there is water either on both sides or on the fuel pool side only. The gates are restrained by guides at the top, mid-height, and bottom. When in the storage position, the gates are horizontally restrained by top and bottom guides and vertically supported by hanger brackets.

Waste Packaging Structure

The Waste Packaging Structure Building is a shear wall structure. The structure is basically a box-like structure. The sloping roof of the structure is a series of 19 pre-cast 20-inch thick panels which span one-way from the south to north walls of the structure. The entire roof section is covered by a 4-inch thick topping slab which is bonded to the pre cast panels. The response of the structure varies with the direction of loading. Load cases which have lateral motion to the north result in shear wall and roof diaphragm action with a triangular soil distribution developing beneath the structure. Loading which produces motion southward brings the Waste Packing Structure Building against the fibrous glass filler material and the Auxiliary Building.

Condensate Demineralizer Waste Evaporator Structure

This two story structure is designed using the loads, loading combinations, and allowable stresses as given in Tables 3.8.4-1 and 3.8.4-2. The concrete portion is designed in accordance with the ACI 318-71 Building Code and the structural steel portion in accordance with AISC 'Manual of Steel Construction,' Seventh Edition. The building is designed to be supported by a bearing pile foundation, with the piles founded on sound rock. The intermediate floor and roof are supported by interior bearing walls and metal decking spanning between steel beams.

3.8.4.4.2 Diesel Generator Building

The structure is designed in accordance with the ACI 318-63 Building Code and is analyzed as a box-type structure assuming all walls fixed at the base slab, Elevation 742.0. The frame is analyzed by the moment distribution methods. Floor Elevation 760.5 and the roof Elevation 773.5 are one-way slabs continuous across interior walls and restrained at exterior walls. All horizontal forces are transmitted through the floor and roof slabs as diaphragms to parallel shear walls and then to the foundation base slab as discussed in Section 3.8.4.4.1 for the Auxiliary Building.

The 9-foot 9-inch base slab distributes superstructure loads uniformly to the supporting crushed stone fill and was analyzed as a flat slab.

The exterior walls and roof of the building are designed to resist the tornado-generated missiles of Spectrum A in Table 3.5-7. Due to the openings in the exterior walls and floor slab at Elevation 760.5, the building is assumed to depressurize. In the hallway and stairway the glass is assumed to break in the event of a tornado, thereby preventing pressure buildup.

A reinforced concrete curb is provided to protect the diesel exhaust stacks from closure due to the impact of tornado-generated missiles.

The exhaust stacks extend 24 inches above roof level. The concrete curb is 18 inches thick and extends 12 inches above the exhaust stacks. The fuel oil storage tank vent lines on the roof are encased in concrete to prevent closure due to missile impact. Details of the curbs, exhaust stacks, and fuel oil storage tank vent line encasements are shown on Figures 3.8.4-26 and 3.8.4-33. Missile entry through the air-intake opening in the ceiling over each electrical board room is prevented by the use of steel canopy with barrier protection to intercept missiles. The items discussed above are also listed in Table 3.5-14.

Concrete block walls are shown on Figures 3.8.4-24 and 3.8.4-25. All reinforced masonry walls are designed in accordance with ACI 531-79 and NUREG-0800, Section 3.8.4, Appendix A.

Diesel Generator Building Doors and Bulkheads

Structural members for the doors and bulkheads were designed as simple beams. The skin plates were designed as square or rectangular diaphragms with all edges fixed.

Earthquake loads used in designing the doors and bulkheads were the accelerations determined for ground level Elevation 742.0, which is the bottom of the doors, for an SSE. These accelerations were used as static loads for determining component and member sizes. After establishing the component and member sizes, a dynamic analysis was made of the doors and bulkheads.

The precast concrete bulkheads (see Figure 3.8.4-33) covering the doors were analyzed for the missile impact loads discussed in Section 3.8.4.1.2.

3.8.4.4.3 Category I Water Tanks and Pipe Tunnels

See Section 9.2.7 for a description of the refueling water storage tanks.

Pipe Tunnels

The pipe tunnels were analyzed using the ICES STRUDL-II Volume I computer program frame analysis and designed in accordance with the provision of the ACI 318-71 Building Code.

3.8.4.4.4 Class 1E Electrical System Manholes

The manholes were analyzed using a continuous frame or a series of flat plates, depending upon the boundary conditions created by the duct run openings. The frames were modeled using the computer program STRESS. The concept of joint continuity was utilized with the plate analysis, i.e., joints were designed for the larger moment from adjacent plates.

The design of the duct runs assumes bending moments due to earthquake loading are caused by direct imposition of the soil curvature on the duct bank.

3.8.4.4.5 North Steam Valve Room

The concrete structure is analyzed as a three-sided open box structure. The 7-foot-thick base slab is designed to span between the foundation walls. The slab is subjected to a pressure loading due to a main steam pipe rupture as well as anchorage loads from restraints located in the slab. The slab was designed using the SAP IV Finite Element computer program. No support from the soil and crushed stone beneath was assumed in the design of the slab. The main steam and feedwater lines exit from the 4-foot-thick west wall where restraints for these lines are anchored. Pipe restraints are also located in the 5-foot thick interior wall in the east end as well as in the 7-foot by 10-foot-deep beam portion of the north wall. The 5-foot interior wall at the east end stiffens the 3-foot-thick east exterior wall. The 2-foot-thick north wall spans horizontally between the stiff complex of end walls and vertically from the base slab to the 7-foot-thick beam portion. The walls are investigated using the SAP IV Finite Element computer programs.

Design procedures for the roof steel were based on simple beam construction as covered in AISC "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings," Part 1 with Type 2 framing connections. The metal decking was attached to a cold formed steel frame which in turn is attached to structural steel with the appropriate number of pressure relief fasteners designed to fail allowing the deck/frame to blow off when the internal pressure at the roof reaches 72 pounds per square foot. The deck/frame is restrained from becoming a missile by using wire rope and clamps which are attached to the main concrete structure.

3.8.4.4.6 Intake Pumping Station and Retaining Walls

The box-type structure is analyzed using conventional structural analysis methods. In accordance with ACI 318-71 Code and subsequent addenda, the alternate design method is used in the design of the structure.

The base slab was analyzed as a flat plate fixed on four sides for areas within the walls. The overhanging areas of the base slab were analyzed as a cantilever or flat plate fixed on three sides. The other floors were analyzed as flat plates with either three or four sides fixed. The walls were analyzed as one-way span vertically for the first 20 feet, above that the walls were analyzed as flat plates fixed on four sides. The missile barrier walls around the top deck were analyzed as cantilevers. The structure is investigated as a whole to ensure continuity of design. The structure has also been investigated for stability against overturning, floating, and sliding. In addition, the structure is designed to resist the pressure differential during a tornado and to maintain its stability under all credible environmental conditions. The structural adequacy is also checked for missile penetration.

Concrete Retaining Walls

The concrete retaining walls were analyzed as cantilevers. The walls have also been investigated for stability against overturning and sliding.

Sheet Pile Retaining Walls

The sheet pile walls are connected by ties to a common concrete "dead man" placed midway between the walls in the earthfill. The ties are steel cables and are anchored in the "dead man" at one end and the sheet piling at the other end. The steel walls are on the inside of the sheet piling and bolted to each pile to transfer the reaction of the pile to the wall. The wall was divided into sections and analyzed as a multibraced wall or cantilever wall depending upon the depth of backfill on the wall.

3.8.4.4.7 Miscellaneous ERCW Structures

Slabs and Beams Supporting ERCW Pipes

The slab supporting the ERCW pipes was analyzed by the use of McDonnell-Douglas' ICES STRUDL computer program. Support was assumed to be furnished entirely by the bearing piles and the piles were designed for the reaction from the computer analysis. Missile protection is provided by roller compacted concrete above the pipes.

The beam encasing the ERCW pipes are analyzed as simple beams with no support from the soil. The encased pipes are in the tension zone of the beam; therefore, the design is for a rectangular beam with no special consideration given to the embedded pipe for flexure or shear. The concrete encasement is designed for missile penetration.

Discharge Overflow Structure

The discharge overflow structure was analyzed assuming it as a series of flat plates. The concept of joint continuity was utilized with the plate analysis by designing the joints for the larger moment from adjacent plates.

Standpipe Structures

The standpipe structures consist of a free standing cantilever supported on a flat slab base on in-situ soil. Generally, the structures were considered solid mass concrete and the design was controlled by structural response for missile impact utilizing an elastic analysis.

Valve Covers

The function of these structures is solely to protect the ERCW valves from tornado missiles; therefore, the design was for missile penetration only.

Missile Protection Slabs and Backfill

See Section 3.8.4.1.7

3.8.4.4.8 Additional Diesel Generator Building

The building is a 96 feet long by 53 feet wide by 32 feet tall (measured from top of base slab) reinforced concrete structure, consisting of a base slab supported by end bearing H-piles, interior floor, roof, and interior and exterior walls. The structure was analyzed as a box-type structure assuming all walls are fixed at the base slab. The building span in the short direction is analyzed using a STRUDL frame program and is designed to withstand all loading conditions assuming a one-way span. In the short direction the interior walls are not considered effective shear walls, but the exterior walls are. Therefore, shear wall and diaphragm deflections are considered in the short direction frame analysis. The building span in the long direction is designed using standard plate theory assuming the interior and exterior walls effectively prevent side sway. The building base slab is a 96 foot long by 53 foot wide by 12 foot thick reinforced concrete slab supported by 154 end-bearing steel H-piles. See Section 3.8.5.5 for additional information on the piles and base slab.

The load definitions, load combinations and allowable stresses are as specified in Section 3.8.4.3.2.

Base Slab Design

The base slab is pile supported. The slab was designed for a uniform live load except where equipment weights dictated a higher value. Equipment loads due to vibration or earthquake acceleration that were transmitted to the slab from anchor bolts were also taken into consideration. In addition, the slab was designed for hydrostatic pressures.

The base slab is a rectangular, cast-in-place, reinforced concrete structure with embedded diesel fuel storage tanks and is supported by piles bearing on rock.

Roof Slab Design

The roof slab was designed for live, seismic, and tornado loads.

Floor Slab

The floor slab is a poured-in-place reinforced concrete slab designed to carry and transmit the floor loads to the building walls. The slab was designed for a uniform live load.

Exterior Walls

The building was designed for tornado venting. However, the exterior walls were designed for tornado, wind and seismic loads.

Fuel Oil Storage Tanks

The steel liner serves no other function except to maintain leak tightness and, therefore, was designed in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Division I. In addition, the liner was designed to prevent buckling of the steel shell due to the following external loads:

a. Hydrostatic pressure from underground water.
b. Shrinkage of the concrete encasement during construction.
c. Expansion or contraction due to temperature differentials.

For flammable liquids storage requirements, the fuel oil storage tanks meet the requirements of the National Fire Protection Association (NFPA) Code 30, which applies to fuel oil storage tanks supplying underground storage of a Class II liquid (diesel fuel).

Equipment Door

The equipment door is composed of a structural steel frame and covered on both sides with a steel-skin plate.

The removable precast concrete missile barrier bulkheads are placed in front of the equipment doors to provide protection from tornado-generated missiles, which are discussed in Section 3.8.4.1.8. In establishing the required concrete thickness for these missile barriers, no consideration was given to the equipment door. Therefore, these barriers are designed to absorb the full missile impact.

Allowable Settlement

The building was designed to accommodate a settlement of 2 inches, with a differential settlement of 1-inch over a 96-foot structure length.

End-Bearing Steel H-Piles

The piles were designed to withstand and transmit to rock the effects of the design loads and conditions.

Seismic Analysis

The structure was analyzed for the effects of the OBE and the SSE as described in Section 3.7.2.1.1.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Concrete

The Category I structures were proportioned to maintain elastic behavior and stresses within stress allowables when subject to the loading combinations of Section 3.8.4.3.

Most Category I structures are essentially low profile box structures with height to base ratios less than 1 and a high factor of safety against sliding or overturning under the most severe loading conditions. Those structures with height to base ratios greater than 1 are designed with adequate factors of safety applied to stability. In addition, all structures are designed to flood or have sufficient weight to prevent flotation under maximum flood conditions. For consideration of sliding, overturning, and floatation of the Additional Diesel Generator Building, see the loading combinations and minimum factor of safety in Table 3.8.4-22.

3.8.4.5.2 Structural and Miscellaneous Steel

Structural and miscellaneous steel (including inside containment) and welds are designed in accordance with AISC "Manual of Steel Construction," Seventh Edition, for Case I loading condition so that the stress in the members and connections do not exceed the allowable stress criteria as set forth in the February 1969 AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," as amended through June 12, 1974. For the factor of safety of these allowable stresses with respect to specified minimum yield point of the material used, see Section 1.5 of "Commentary on the Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings." Both specifications and commentary are included in the AISC "Manual of Steel Construction."

For Case II loading condition the actual stresses do not exceed the allowable stresses as set forth in Table 3.8.4-2. The allowable stresses for Case II loading have a minimum factor of safety of 1.11 based on the specified minimum yield point of the material used.

TVA has generally installed, and will continue to install fillet welds to meet the minimum weld size specifications of Table 1.17.5 of AISC Manual of Steel Construction. Where TVA drawings have specified fillet welds below the minimum sizes specified by AISC, these welds do meet the allowable stress requirements identified above. Weld qualification testing has demonstrated the adequacy of all fillet welds that were installed below minimum AISC specifications.

The Additional Diesel Generator Building structural steel was proportioned to meet the applicable codes discussed in Appendix 3.8E and load combinations in Section 3.8.4.3.

Structural steel and miscellaneous steel, which is highly restrained and is located in a high temperature environment, is evaluated for effects of thermal loads.

3.8.4.5.3 Miscellaneous Components of the Auxiliary Building

Control Room Shield Doors

Allowable stresses for all load combinations used for the various parts of the door and dogs are given in Table 3.8.4-3. For normal load conditions the allowable stresses provide a safety factor of 2 to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts. For the limiting condition of SSE, stresses do not exceed 0.9 yield.

Watertight Equipment Hatch Covers

Allowable stresses for all load combinations used for the various parts are given in Table 3.8.4-23. Allowable stresses for normal loading combinations are based on the AISC specification (see Section 3.8.4.2 and Table 3.8.4-23). For limiting conditions, such as SSE, tornado, and flood, stresses do not exceed 0.9 yield.

Railway Access Hatch Covers

Allowable stresses for all load combinations used for the various parts are given in Table 3.8.4-4. For normal load conditions, the allowable stresses provide safety factors of 2 to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts. For limiting conditions, such as an OBE or SSE, stresses do not exceed 0.9 yield.

Railroad Access Door

Allowable stresses for all load combinations used for the various parts of the door, embedded frame, and hoist enclosure are given in Table 3.8.4-5. For normal load conditions the allowable stresses provide a safety factor of 2 to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts. For limiting conditions such as an OBE or SSE and hoist stall, stresses do not exceed 0.9 yield.

Manways in RHR Sump Valve Room

Allowable stresses for all load combinations used for the various parts are given in Table 3.8.4-6. For limiting conditions, such as a SSE, stresses do not exceed 0.9 yield.

Pressure Confining Personnel Doors

Allowable stresses for all load combinations used for the various parts are given in Table 3.8.4-7. For normal load conditions, the allowable stresses provide safety factors of 2 to 1 on yield on structural parts and 5 to 1 on ultimate for mechanical parts. For limiting conditions, such as an SSE, flood, and tornado loadings, stresses do not exceed 0.9 yield.

Fuel Pool Gates

Allowable stresses for all load combinations used for the gates are given in Table 3.8.4-21. For normal load conditions the allowable stresses do not exceed 0.6 of yield. For limiting conditions, such as the SSE, the stresses do not exceed 0.90 of yield.

3.8.4.5.4 Intake Pumping Station Traveling Water Screens

Allowable stresses for all load combinations used for the various parts are given in Table 3.8.4-11. For normal load conditions, the allowable stresses provide safety factors of 1.79 (F_y/0.56 F_y) to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts. For limiting conditions, such as a safe shutdown earthquake, stresses do not exceed 0.9 yield.

3.8.4.5.5 Diesel Generator Building Doors and Bulkheads

Load combinations and allowable stresses for all combinations are given in Table 3.8.4-13. For missile impact, yield point of material will be exceeded and the member practically deform. For normal load condition, the allowable stresses provide safety factors of 1.67 (F_y/0.6 F_y) to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts. For limiting conditions, except for missile impact, stresses do not exceed 0.9 yield.

3.8.4.5.6 Additional Diesel Generator Building Missile Barriers

Design of missile barriers for the Additional Diesel Generator Building is discussed in Section 3.5.3.1.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

General

See Section 3.8.1.6.

3.8.4.6.1 Materials

See Section 3.8.1.6.1.

For the Additional Diesel Generator Building the following materials were used:

Structural Steel

Rolled shapes, plates, and bars meet Specification ASTM A 36. Fabricated high-strength steel meets Specification ASTM A 572 and bolting meets Specification ASTM A 325 or A 490. Anchor bolts meet ASTM A 307 or A 36 steel.

Reinforcing Steel

The yield strength of reinforcing steel used in the building is 60,000 psi (ASTM A 615, grade 60) or greater.

Concrete

The compressive strength of concrete is 3000 psi or greater.

3.8.4.6.2 Quality Control

Concrete production and testing are discussed in Section 3.8.1.6.2.

In addition to the 4,000-psi-at-28-days mix discussed in Section 3.8.1.6.2, a 3,000-psi-at-28-days mix, a 3,000-psi-at-90-days mix, a 5,000-psi-at-28-days mix, and a 4,000-psi-at-90-days mix were used. Some concrete did not meet specification requirements. This was evaluated and documented in the report CEB-86-19-C "Concrete Quality Evaluation."[2] Results have been documented in affected calculation packages and drawings.

Testing of reinforcing steel and cadweld splices is discussed in Section 3.8.1.6.2.

The control room shield doors, watertight equipment hatch covers, railway access hatch covers, railroad access doors, equipment hatch doors and sleeves, manways in the RHR sump valve room, and the pressure confining personnel doors were designed and erected by TVA in accordance with TVA's quality assurance program. Design and fabrication by the contractor were in accordance with the contractor's quality assurance program which was reviewed and approved by TVA's design engineers. The contractor's quality assurance program covers the criteria in Appendix B of 10 CFR 50.

Fabrication procedures such as welding and nondestructive testing were included in appendices to the contractor's quality assurance program.

ASTM standards were used for all material specifications and certified mill test reports were provided by the contractor for materials used for load-carrying members.

The fuel pool gates were designed and procured before quality assurance requirements were imposed. An evaluation was conducted to verify that the gates were equivalent to gates that would have been designed and fabricated under a quality assurance program.

3.8.4.6.3 Special Construction Techniques

No special construction techniques were used, except for the fuel oil storage tanks in the Additional Diesel Generator Building. For these tanks joint welding procedures used in fabrication of the steel liner were qualified in accordance with ASME Boiler and Pressure Vessel Code, Section IX, prior to use by TVA or the fabricator.

3.8.4.7 Testing and Inservice Surveillance Requirements

Testing for the steel liners in the fuel oil storage tanks for the Additional Diesel Generator Building was accomplished by subjecting them to a standard hydrostatic test in accordance with ASME, Section VIII.

3.8.4.7.1 Concrete and Structural Steel Portions of Structures

A program to monitor the settlement of other Category I structures is as shown in Figures 3.8.4-66 and 3.8.4-67.

3.8.4.7.2 Miscellaneous Components of Auxiliary-Control Building

Control Room Shield Doors

After erection and adjustment the doors were inspected for proper operation of the dogs and free movement on the trolleys.

After the initial inspection, periodic visual inspections of the doors are to be made. Parts inspected during these visual inspections are to include connections to trolleys, structural members for paint deterioration, and dogs.

Watertight Equipment Hatch Covers

After initial inspection, periodic visual inspections of the covers are to be made. A visual inspection is made of all screws to see that they are securely tightened and that none are missing. The painted inscriptions on the covers are inspected for any deterioration. In the event that the hatch covers are removed, an inspection is made of the gaskets to ensure that they are clean and free of any damage or deterioration which would prevent their forming a proper seal. The embedded frames are inspected to ensure that the mating surfaces are clean and free of foreign material before the covers are reinstalled.

Railway Access Hatch Covers

After the initial inspection, periodic visual inspections of the covers are made. Parts inspected during the visual inspection are to include all bolted connections, structural members for paint deterioration, limit switches, and rubber seals. The seals are carefully, inspected for cracks, blemishes, or any other indications of deterioration of the rubber and for properly seating at the sealing surfaces.

Railway Access Door

Prior to shipment of the door from the contractor's plant, the splice welds in the skin plate of the door and welds among the periphery of the skin plate and structural members were magnetic particle tested.

After completion of the initial tests and inspection, periodic visual inspections of the door and its parts are made. Parts inspected are to include all bolted connections, limit switches, door tracks, and rollers. Painting is to be inspected for evidence of deterioration, and the seals are carefully inspected for cracks, blemishes, or any other indications of deterioration of the rubber.

Pressure Confining Personnel Doors

After the initial inspection, periodic visual inspections of the doors are made. Parts inspected during these visual inspections include all bolted connections, structural members for paint deterioration, latching or dogging mechanisms and limit switches for physical condition, and the seals. The seals are carefully inspected for cracks, blemishes, or any other indications or deterioration and for proper seating at the sealing surfaces.

Fuel Pool Gates

After initial inspection, periodic visual inspections of the gates are made. The seals are carefully inspected for cracks, blemishes, or any other indications of deterioration.

3.8.4.7.3 Miscellaneous Components of the Intake Pumping Station

Traveling Water Screens

After the initial inspection and testing, the screens are inspected at periodic intervals. Parts inspected include drive components, carrier chains, baskets including the wire panels, spray pipes, spray nozzles, main frames, lights, and lubricating devices.

Watertight Doors

After the initial inspection, periodic visual inspections of the doors are made. Parts inspected during these visual inspections include the bolted connections, structural members for paint deterioration, latching or dogging mechanisms and limit switches for physical condition, and the seals. The seals are carefully inspected for cracks, blemishes, or any other indications or deterioration and for proper seating at the sealing surfaces.

REFERENCES

1. TVA drawing series 46W454 "Architectural Door and Hardware Schedule."

2. TVA Civil Engineering Branch Report Number CEB-86-19-C, "Concrete Quality Evaluation."

3. Portland Cement Association publication, T18-4, "Analysis of Small Reinforced Concrete Buildings for Earthquake Forces," pp. 30-32.

TABLE 3.8.4-1 (SHEET 1 of 4)

AUXILIARY-CONTROL BUILDING CONCRETE STRUCTURE LOADS, LOADING COMBINATIONS AND ALLOWABLE STRESSES

I. Loads

The following terms are used in the load combination equations.

C = Construction condition.

C' = Crane load, including wind on crane.

D = Dead load of structure and equipment plus any other permanent load contributing stress, such as soil pressure. Hydrostatic pressure from ground water Elevation 710, exterior walls; Elevation 726, uplift.

D' = D + hydrostatic pressure from groundwater Elevation 724.4.

E = Operating Basis Earthquake

E' = Safe Shutdown Earthquake

H = Spent fuel pit hydrostatic pressure. Worst condition of the following except as noted:

1. Normal water level in pit, cask loading area and canal.
2. Canal empty of water. Normal water level in other areas.
3. Cask loading area empty of water. Normal water level in other areas. Considered for Case I load combinations only.

L = Live load. For live load on slabs, see Figure 3.8.4-9.

P = Accidental drop of fuel cask on walls of cask loading area.

T_a = Accidental increase in temperature of water in pit to 212°F in 8 hours. Temperature inside building 60°F.

T_N = Normal temperature of water in fuel pit and canal 120°F. Temperature inside building 60°F.

TABLE 3.8.4-1 (SHEET 2 of 4)

AUXILIARY-CONTROL BUILDING CONCRETE STRUCTURE LOADS, LOADING COMBINATIONS AND ALLOWABLE STRESSES (Cont'd)

W	=	Wind load, see Section 3.3.
W_t	=	Tornado, see Section 3.3.
P_a	=	Pressure from main steam break.
R_a	=	Pipe reaction from thermal effects of main steam break.
T_a	=	Thermal effects from main steam break.
Y_r	=	Pipe anchor force due to jet from pipe break.
Y_j	=	Jet force from pipe break.
Y_m	=	Missile impact froce from pipe whip.

II. Load Combinations and Allowable Stresses

Auxiliary-Control Building

Load combinations	Allowable WSD Stresses
Case I = D+L	Normal (ACI 318-63 or 318-71)
Case Ia = D'+L	1.35 x normal
Case Ib = D+L+W+C	1.33 x normal
Case II = D+L+E	f_c = normal (ACI 318-63 or 318-71) $f_s = 0.50 f_y$
Case III = D+L+E'	*$f_c = 0.75 f''_c$ $f_s = 0.90 f_y$
Case IV = D+L+W_t	*$f_c = 0.75 f''_c$ $f_s = 0.90 f_y$

TABLE 3.8.4-1 (SHEET 3 of 4)

AUXILIARY-CONTROL BUILDING CONCRETE STRUCTURE LOADS, LOADING COMBINATIONS AND ALLOWABLE STRESSES (Cont'd)

Where main steam lines pass through the Auxiliary-Control Building at the south main steam valve room, the following factored load combinations were considered in addition to those listed above:

Load Combinations		Allowable WSD Stresses	USD Load Factors
Case VI =	$D+L+P_a$	$*f_c = .75f''_c$ $fs = .9f_y$	$1.0D+1.0L+1.5P_a$
Case VII =	$D+L+P_a+1.0$ $Y_r+Y_j+Y_m)+1.0E$	$*f_c = .75f''_c$ $f_s = .9f_y$	$1.0D+1.0L+1.25P_a+$ $1.0(Y_r+Y_j+Y_m)+1.25E$
Case VIII =	$D+L+1.0P_a+1.0$ $Y_r+Y_j+Y_m)+1.0E'$	$*f_c = .75f''_c$ $f_s = .9f_y$	$1.0(D+L+P_a+Y_r+Y_j+Y_m +E')$

*Concrete stresses other than flexure = 1.67 x normal

Where the main steam lines pass under the elevation 755.0 floor slab of the Control Building, vital structural elements in that area were designed for cases I through IV and the case IX listed below.

		Allowable WSD Stresses
Case IX =	D+E'+J*	$f_c = 1.67$ (Normal Concrete) $f_s = 0.9\ f_y$

*J is a jet load of 360 kips spread over 50 ft^2

Material Properties

Concrete	
Slabs and walls	$f''_c = 3000$ or 4000 psi
Columns	$f''_c = 4000$ psi
Concrete weight	w = 145 pcf
Reinforcing steel	$f_y = 60{,}000$ psi (ASTM A615. Grade 60)

TABLE 3.8.4-1 (SHEET 4 of 4)

AUXILIARY-CONTROL BUILDING CONCRETE STRUCTURE LOADS, LOADING COMBINATIONS AND ALLOWABLE STRESSES (Cont'd)

Auxiliary Building Spent Fuel Pit

		Load Combinations	Allowable WSD Stresses
Case I	=	D+H	Normal (ACI 318-63)
	=	D+H+T_N	Normal (ACI 318-63)
Case II	=	D+H+E	f_c = normal (ACI 318-63) f_s = 0.50 fy
	=	D+H+E+T_N	fc = normal (ACI 318-63) fs = 0.50 f_y
Case III	=	D+H+E'	*fc = 0.75 f''_C fs = 0.90 f_y
	=	D+H+E'+T_N	*fc = 0.75 f''_C fs = 0.90 f_y
Case IV	=	D+H+T_a	*fc = 0.75 f''_C fs = 0.90 f_y
Case IVa	=	D+H+P	*fc = 0.75 f''_C fs = 0.90 fy

*Concrete stresses other than flexure = 1.67 x normal.

Material Properties (see above)

Auxiliary Building Concrete Structure Earth Values

Angle of internal friction	ϕ = 32°
Angle of friction between fill and structure	ϕF = 16°
Unit weight of fill	
Surcharge	w = 120 psf
Dry	
Saturated	w = 65 psf
Surcharge	
A1 and A15 line walls	1730 psf
Others	200 psf

TABLE 3.8.4-2 (Sheet 1 of 4)

AUXILIARY-CONTROL BUILDING STRUCTURAL STEEL LOADS, LOADING CONDITIONS AND ALLOWABLE STRESSES

Control Building Portion

1. Live Loads (LL)

   a. Elevation 755.0 - 400 psf (to include cable trays, ducts, walls, and electrical boards)
   b. Elevation 741.0 - 100 psf when seismic loads (E and E') are not present, 10 psf when seismic loads are present
   c. Elevation 729.0 - 100 psf

2. Dead Loads (DL)

   a. 8-inch concrete brick wall - 100 psf
   b. 1-1/2-inch steel grating - 12 psf
   c. Concrete - 12.5 psf per inch thickness
   d. Steel framing - 15 psf
   e. Piping - varies

3. Tornado (TOR)

   a. Elevation 729.0 - 3.0 psi (between column lines C-1 to C-3 and C-11 to C-13).

Auxiliary Building Portion

1. Live Loads (LL) - The following loads shall be used unless shown otherwise on Figure 3.8.4-9, "Concrete Floor Design Data."

   a. Roof (Refueling Area) - 50 psf or equipment
   b. Roof (Main steam valve room) - 50 psf when seismic loads are not present and 5 psf when seismic loads are present, or equipment
   c. Floor (Main steam valve room) - 100 psf or equipment
   d. Roof (CDWE Building) - 300 psf or equipment
   e. Floor (CDWE Building) - 300 psf or equipment
   f. Construction Load - 20 psf
   g. Miscellaneous live load - 30 psf

2. Dead Loads (DL)

   a. Concrete - 12.5 psf per inch thickness
   b. Steel roof decking - 4 psf
   c. Steel roof framing - 30 psf
   d. Steel floor framing - 15 psf

3. Tornado (TOR)

   a. Velocity - 360 mph

TABLE 3.8.4-2 (Sheet 2 of 4)

AUXILIARY-CONTROL BUILDING STRUCTURAL STEEL LOADS, LOADING CONDITIONS AND ALLOWABLE STRESSES (Cont'd)

Auxiliary-Control Building

Seismic Loads

a. Operating Basis Earthquake (OBE) maximum ground acceleration

Horizontal 0.09g

Vertical 0.06g

b. Safe Shutdown Earthquake (SSE) maximum ground acceleration

Horizontal 0.18g

Vertical 0.12g

Loading Condition	Tension on Net Section	Shear on Gross Section	Compression on Gross Section	Bending	Concrete Bearing
Case I DL + LL ± OBE	0.60 F_Y	0.40 F_Y	See Note 1	0.66 F_Y to -0.60 F_Y	0.25 f''$_c$
Case II DL + LL + SSE	0.90 F_Y	$\frac{0.9F_Y}{\sqrt{3}}$	See Note 2	0.90 F_Y	.595 f''$_c$ See Note 3
Case III DL + LL + TOR	0.90 F_Y	$\frac{0.9F_Y}{\sqrt{3}}$	See Note 2	0.90 F_Y	.595 f''$_c$ See Note 3

Note 1 - Varies with slenderness ratio, see AISC "Manual of Steel Construction," 7th Edition, Table 1-36, Page 5-84.

Note 2 - Varies with slenderness ratio:

Main and secondary members, where $KL/r \le C_c$:

$$F_a = 0.9F_Y\left[1-\frac{(KL/r)^2}{2C_c^2}\right] \qquad \text{(A)}$$

TABLE 3.8.4-2 (Sheet 3 of 4)

AUXILIARY-CONTROL BUILDING STRUCTURAL STEEL LOADS, LOADING CONDITIONS AND ALLOWABLE STRESSES (Cont'd)

Main members, where $C_c < KL/r < 200$:

$$F_a = \frac{0.9\pi^2 E}{(KL/r)^2} \qquad \text{(B)}$$

Secondary members, where $120 < L/r \leq 200$:

$$F_{as} = \frac{F_a \text{ [by Formula (A) or (B)]}}{1.6 - L/200r}$$

Where:

$$C_c = \sqrt{\frac{2\pi^2 E}{F_Y}}$$

E = Modulus of elasticity of steel

F_a = Axial compressive stress permitted in the absence of bending moment (kips per square inch)

F_{as} = Axial compressive stress, permitted in the absence of bending moment, for bracing and other secondary members (kips per square inch)

F_Y = Specified minimum yield stress of material (kips per square inch)

$f''c$ = Compressive strength of concrete

K = Effective length factor

L = Actual unbraced length (inches)

r = Governing radius of gyration (inches)

Material Properties

Steel Properties

Cc = 126.1

E = 29,000,000 psi

Fy = 36,000 psi

TABLE 3.8.4-2 (Sheet 4 of 4)

AUXILIARY-CONTROL BUILDING STRUCTURAL STEEL LOADS, LOADING CONDITIONS AND ALLOWABLE STRESSES (Cont'd)

Note 3- When the supporting surface is wider on all sides than the loaded area, the permissible bearing stress on the loaded area may be multiplied by $\sqrt{A_2 / A_1}$, 1 but not more than 2.

Where: A_1 = Loaded area

A_2 = Maximum area of the portion of the supporting surface that is geometrically similar to and concentric with the loaded area.

TABLE 3.8.4-3

CONTROL ROOM SHIELD DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Door and Jamb Shield Assemblies
Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
	Doors Open or Closed			
I	Dead	$0.50F_y$	$0.47F_y$	$0.33F_y$
II	Dead + OBE[1]	$0.60F_y$	$0.60F_y$	$0.40F_y$
III	Dead + SSE[1]	$0.90F_y$	$0.90F_y$	$0.60F_y$

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
	Doors Open or Closed			
I	Dead	Ultimate 5	Ultimate 5	Ultimate 7.5
II	Dead + OBE[1]	$0.6F_Y$	$0.6F_y$	$0.4F_y$
III	Dead + SSE[1]	$0.9F_y$	$0.9F_y$	$0.6F_y$

Notes:

(1) Acts in any one horizontal direction only at any given time and acts in vertical and horizontal directions simultaneously.

(2) The value given for allowable compression stress is the maximum value permitted when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y\left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2}\right] \text{ when } \frac{Kl}{r} \le C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2} \text{ when } \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8.4-4 (SHEET 1 of 2)

AUXILIARY BUILDING RAILROAD ACCESS HATCH COVERS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Cover Structure and Embedded Frame

No.	Load Combinations	Allowable Stresses (psi) Tension	Shear Compression(1)	
	Covers Closed			
I	Dead load plus live load at 100 lb/ft^2	$0.50F_y$	$0.47F_y$	$0.33F_y$
II	Dead load plus live load at 100 lb/ft^2 plus OBE	$0.60F_y$	$0.60F_y$	$0.40F_y$
III	Dead load plus live load at 100 lb/ft_2 plus SSE	$0.90F_y$	$0.90F_y$	$0.60F_y$
	Covers Open			
IV	Dead load plus hoist pull	$0.50F_y$	$0.47F_y$	$0.33F_y$
V	Dead load plus hoist pull plus OBE	$0.60F_y$	$0.60F_y$	$0.40F_y$
VI	Dead load plus hoist pull plus SSE	$0.90F_y$	$0.90F_y$	$0.60F_y$

Mechanical Parts on Covers and Frame

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression(1)	Shear
	Covers Closed		
I	Dead load plus live load at 100 lb/ft^2	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
II	Dead load plus live load of 100 lb/ft^2 plus OBE	$0.6F_y$	$0.4F_y$
III	Dead load plus live load at 100 lb/ft^2 plus SSE	$0.9F_y$	$0.6F_y$
	Covers Open		
IV	Dead load plus hoist pull	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
V	Dead plus live load of 100 lb/ft^2 plus OBE	$0.6F_y$	$0.4F_y$
VI	Dead load plus hoist pull plus SSE	$0.9F_y$	$0.6F_y$

TABLE 3.8.4-4 (Sheet 2 of 2)

AUXILIARY BUILDING RAILROAD ACCESS HATCH COVERS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

	Hoist Unit Supports	Allowable Stresses(psi)		
No.	Load Combinations	Tension	Compression(1)	Shear
	Hatch Opening			
I	Dead load Hoist pull	18,000	17,000	12,000
II	Dead load Stall	32,400	32,400	21,600

Other Mechanical Parts

		Allowable Stresses (psi)	
No.	Load Combinations	Tension and Compression(1)	Shear
	Covers Open		
I	Dead load Hoist pull	$\frac{\text{Ult}}{5}$	$\frac{2 \times \text{Ult}}{15}$
II	Dead load Stall	$0.9F_y$	$2/3 \times 0.9F_y$

Notes:

(1) The value given for allowable compression stress is the maximum value, F_{cr}, permitted when buckling does not control. The critical buckling stress shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8.4-5 (SHEET 1 of 2)

RAILROAD ACCESS DOOR
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Door, Embedded Frame and Door Track

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
	Door Closed			
I	Dead load plus windload at 10 lb/ft^2	$0.50F_y$	$0.47F_y$	$0.33F_y$
II	Dead load plus windload at 30 lb/ft^2	$0.90F_y$	$0.90F_y$	$0.60F_y$
III	Dead load plus windload at 10 lb/ft^2 plus OBE	$0.60F_y$	$0.60F_y$	$0.40F_y$
IV	Dead load plus windload at 10 lb/ft^2 plus SSE	$0.90F_y$	$0.90F_y$	$0.60F_y$
	Door Open			
V	Dead load plus hoist pull	$0.50F_y$	$0.47F_y$	$0.33F_y$
VI	Dead load plus hoist pull plus OBE	$0.60F_y$	$0.60F_y$	$0.40F_y$
VII	Dead load plus hoist pull plus SSE	$0.90F_y$	$0.90F_y$	$0.60F_y$

Hoist Unit & Enclosure

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
I	Dead load plus hoist pull	$0.50F_y$	$0.47F_y$	$0.33F_y$
II	Dead load plus stall	$0.90F_y$	$0.90F_y$	$0.60F_y$
III	Dead load plus hoist stall plus OBE	$0.60F_y$	$0.60F_y$	$0.40F_y$
IV	Dead load plus hoist pull plus SSE	$0.90F_y$	$0.90F_y$	$0.60F_y$

TABLE 3.8.4-5 (SHEET 2 of 2)

RAILROAD ACCESS DOOR
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

Mechanical Parts on Door

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[2]	Allowable Stresses (psi) Shear
	Door Open		
I	Dead load plus windload at 10 lb/ft^2	$\frac{Ult}{5}$	$\frac{2x\ Ult}{15}$
II	Dead load plus windload a 10 lb/ft^2 plus OBE	$0.6F_y$	$0.4F_y$
III	Dead load plus windload at 10 lb/ft^2 plus SSE[1]	$0.9F_y$	0.6Fy

Other Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[2]	Allowable Stresses (psi) Shear
	Door Open		
I	Dead load Hoist pull	$\frac{Ult}{5}$	$\frac{2\ x\ Ult}{15}$
II	Dead load Stall	$0.9F_y$	$0.6F_y$

NOTE:

(1) Acts in one horizontal direction only at any given time and acts in the horizontal and vertical directions simultaneously.

(2) The value indicated for the allowable compression stresses is the maximum value permitted when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8.4-6

MANWAYS IN RHR SUMP VALVE ROOM
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression(2)	Shear
	Manway Closed		
I	Dead load plus OBE(1)	$0.6F_Y$	$0.4F_Y$
II	Dead load plus SSE(1)	$0.9F_Y$	$0.6F_Y$
III	Dead load plus 19 psi from outside	$0.9F_Y$	$0.6F_Y$
	Manway Open		
IV	Dead load plus OBE(1)	$0.6F_Y$	$0.4F_Y$
V	Dead load plus SSE(1)	$0.9F_Y$	$0.6F_Y$

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression(2)	Shear
	Manway Closed		
I	Dead load plus OBE(1)	$0.6F_Y$	$0.4F_Y$
II	Dead load plus SSE(1)	$0.9F_Y$	$0.6F_Y$
III	Dead load plus 19 psi from outside	$0.9F_Y$	$0.6F_Y$
	Manway Open		
IV	Dead load plus OBE(1)	$0.6F_Y$	$0.4F_Y$
V	Dead load plus SSE(1)	$0.9F_Y$	$0.6F_Y$

NOTES:

(1) Acts in one horizontal direction only at any given time and acts in vertical and horizontal directions simultaneously.

(2) The values given for allowable compression stress is the maximum value permitted when buckling does not control. The critical buckling stress, Fcr, shall be used in place of FY when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

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TABLE 3.8.4-7 (Sheet 1 of 10)
UNIT 1

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1]

(All Doors as shown in Table 3.8.4-7a except A55, A57, C20, C26, A101, A105, and A216)

Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[(2)]	Shear
	Doors Open or Closed			
I	DL + Load from Door Closers	0.50 F_y	0.47 F_y	0.33 F_y
II	DL + OBE + Load from Door Closers	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + Load from Door Closers	0.90 F_y	0.90 F_y	0.60 F_y
	Doors Closed			
III[3]	DL + 3-psi pressure (bidirectional where applicable)	0.90 F_y	0.90 F_y	0.60 F_y
IV[4]	DL + OBE + 2-psi toward annulus	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + 2-psi toward annulus	0.90 F_y	0.90 F_y	0.60 F_y
V[5]	DL + 3 inches of water pressure on either side of door	0.50 F_y	0.47 F_y	0.33 F_y
VI[6]	DL + Flood to elevation 739.7	0.90 F_y	0.90 F_y	0.60 F_y

1. Thermal load effects are insignificant and hence need not be considered in the design of doors.
2. The values indicated for the allowable compression stresses are the maximum values permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

3. Applies to all doors except A64, A65, A56, A60, A77, A111, A117, A122, A125, A130, A133, A151, A160, A162, A183, A192, A204, A206, A207, A208, A209, A212, A213, C37, C49, C50, C53, C60, DE1, DE4 and DE5.
4. Applies to doors A64, A65, and A77 only.
5. For doors A56, A60, A65, A94, A111, A113, A114, A122, A123, A125, A130, A132, A133, A151, A152, A159, A160, A161, A162, A183, A192, A204, A206, A207, A208, A209, A212, A213, A214, A215, DE1, DE4, and DE5, the load combination is: DL + 1/2" water pressure on either side of door.
   For doors C36, C37, C49, C50, C53, C54 and C60, the load combination is: DL + 1/8" water pressure on either side of door.
6. Applies to door A65 only.

TABLE 3.8.4-7 (Sheet 2 of 10)
UNIT 1

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(All Doors as shown in Table 3.8.4-7a except A55, A57, C20, C26, A101, A105, and A216)

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi): Tension and Compression[2]	Allowable Stresses (psi): Shear
	Doors Open or Closed		
I	DL + Load from door closers	$F_u/5$	$2\ F_u/15$
II	DL + OBE + Load from door closers	$0.60\ F_y$	$0.40\ F_y$
	DL + SSE + Load from door closers	$0.90\ F_y$	$0.60\ F_y$
	Doors Closed		
III[3]	DL + 3-psi pressure (bidirectional where applicable)	$0.90\ F_y$	$0.60\ F_y$
IV[4]	DL + OBE + 2-psi toward annulus	$0.60\ F_y$	$0.40\ F_y$
	DL + SSE + 2-psi toward annulus	$0.90\ F_y$	$0.60\ F_y$
V[5]	DL + 3 inches of water pressure on either side of door	$F_u/5$	$2\ F_u/15$
VI[6]	DL + Flood to elevation 739.7	$0.90\ F_y$	$0.60\ F_y$

1. Thermal load effects are insignificant and hence need not be considered in the design of doors.
2. The values indicated for the allowable compression stresses are the maximum values permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \le C_c \qquad 3$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 4$$

3. Applies to all doors except A64, A65, A56, A60, A77, A111, A117, A122, A125, A130, A133, A151, A160, A162, A183, A192, A204, A206, A207, A208, A209, A212, A213, C37, C49, C50, C53, C60, DE1, DE4 and DE5.
4. Applies to doors A64, A65 and A77 only.
5. For doors A56, A60, A65, A94, A111, A113, A114, A122, A123, A125, A130, A132, A133, A151, A152, A159, A160, A161, A162, A183, A192, A204, A206, A207, A208, A209, A212, A213, A214, A215, DE1, DE4, and DE5, the load combination is:
   DL + 1/2" water pressure on either side of door.
   For doors C36, C37, C49, C50, C53, C54, and C60, the load combination is:
   DL + 1/8" water pressure on either side of door.
6. Applies to door A65 only.

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TABLE 3.8.4-7 (Sheet 3 of 10)
UNIT 1

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(Doors A55, A57, C20, C26, A101, and A105)

Structural Parts

		Allowable Stresses (psi)		
No.	Load Combinations	Tension	Compression[2]	Shear
	Doors Open			
I	DL + Load from Door Closers	0.50 F_y	0.47 F_y	0.33 F_y
II	DL + OBE + Load from Door Closers	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + Load from Door Closers	0.90 F_y	0.90 F_y	0.60 F_y
	Doors Closed			
III[3]	DL + CCWS Flood + OBE + 3-psi pressure (bidirectional where applicable)	0.60 F_y	0.60 F_y	0.40 F_y
IV[3]	DL + CCWS flood + SSE + 3-psi pressure (bidirectional where applicable)	0.90 F_y	0.90 F_y	0.60 F_y
V[4]	DL + OBE + Pressure from valve rooms	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + Pressure from valve rooms	0.90 F_y	0.90 F_y	0.60 F_y

1. Thermal load effects are insignificant and hence need not be considered in the design of doors.

2. The values indicated for the allowable compression stresses are the maximum values permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 5$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 6$$

3. The CCWS flood condition does not apply to doors A101 and A105, and differential pressure load due to tornado need not be considered simultaneously with seismic load.

4. Applies to doors A101 and A105 only.

WBN

TABLE 3.8.4-7 (Sheet 4 of 10)
UNIT 1

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(Doors A55, A57, C20, C26, A101, and A105)

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi): Tension and Compression[2]	Allowable Stresses (psi): Shear
	Doors Open		
I	DL + Load from door closers	$F_u/5$	$2\ F_u/15$
II	DL + OBE + Load from door closers	$0.60\ F_y$	$0.40\ F_y$
	DL + SSE + Load from door closers	$0.90\ F_y$	$0.60\ F_y$
	Doors Closed		
III[3]	DL + CCWS flood + 3-psi pressure (bidirectional where applicable)	$0.90\ F_y$	$0.60\ F_y$
IV[4]	DL + OBE + Pressure from valve room	$0.60\ F_y$	$0.40\ F_y$
	DL + SSE + Pressure from valve room	$0.90\ F_y$	$0.60\ F_y$

1. Thermal Load effects are insignificant and hence need not be considered in the design of doors.

2. The values indicated for the allowable compression stresses is the maximum value permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] \text{ when } \frac{Kl}{r} \le C_c \qquad 7$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2} \text{ when } \frac{Kl}{r} > C_c \qquad 8$$

3. The CCWS flood condition does not apply to doors A101 and A105, and differential pressure load due to tornado need not be considered simultaneously with seismic load.

4. Applies to doors A101 and A105 only.

TABLE 3.8.4-7 (Sheet 5 of 10)
UNIT 1

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(Door A216)

Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
I	DL + P	0.50 F_y	0.47 F_y	0.33 F_y
II	DL + P + OBE	0.60 F_y	0.60 F_y	0.40 F_y
III	DL + P + SSE	0.90 F_y	0.90 F_y	0.60 F_y

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[2]	Shear
I	DL + P	$F_u/5$	2 $F_u/15$
II	DL + P + OBE	0.60 F_y	0.40 F_y
III	DL + P + SSE	0.90 F_y	0.60 F_y

DL - Stresses generated by dead loads and door closer loads.

P - Stresses generated by a pressure differential of 1/2 inch of water acting to open doors.

1. Thermal Load effects are insignificant and hence need not be considered in the design of doors.

2. The values indicated for the allowable compression stresses is the maximum value permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 9$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 10$$

TABLE 3.8.4-7 (Sheet 6 of 10)
UNIT 2

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1]

(All Doors as shown in Table 3.8.4-7a except A55, A57, C20, C26, A101, A105, A216, and A217)

Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[(2)]	Shear
	Doors Open or Closed			
I	DL + Load from Door Closers	0.50 F_y	0.47 F_y	0.33 F_y
II	DL + OBE + Load from Door Closers	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + Load from Door Closers	0.90 F_y	0.90 F_y	0.60 F_y
	Doors Closed			
III[3]	DL + 3-psi pressure (bidirectional where applicable)	0.90 F_y	0.90 F_y	0.60 F_y
IV[4]	DL + OBE + 2-psi toward annulus	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + 2-psi toward annulus	0.90 F_y	0.90 F_y	0.60 F_y
V[5]	DL + 3 inches of water pressure on either side of door	0.50 F_y	0.47 F_y	0.33 F_y
VI[6]	DL + Flood to elevation 739.7	0.90 F_y	0.90 F_y	0.60 F_y

1. Thermal load effects are insignificant and hence need not be considered in the design of doors.
2. The values indicated for the allowable compression stresses are the maximum values permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 11$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 12$$

3. Applies to all doors except A64, A65, A77, A78, A56, A60, A111, A117, A118, A122, A125, A130, A133, A151, A160, A162, A183, A192, A206, A207, A208, A209, A212, A213, C37, C49, C50, C53, C60, DE1, DE4 and DE5.
4. Applies to doors A64 and A65 only.
5. For doors A56, A60, A65, A78, A94, A99, A111, A122, A123, A125, A130, A132, A133, A151, A152, A159, A160, A161, A162, A183, A192, A206, A207, A208, A209, A212, A213, A214, A215, DE1, DE4, and DE5, the load combination is:
   DL + 1/2" water pressure on either side of door.
   For doors C36, C37, C49, C50, C53, C54 and C60, the load combination is:
   DL + 1/8" water pressure on either side of door.
6. Applies to door A65 and A78 only.

TABLE 3.8.4-7 (Sheet 7 of 10)
UNIT 2

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(All Doors as shown in Table 3.8.4-7a except A55, A57, C20, C26, A101, A105, and A216)

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[2]	Allowable Stresses (psi) Shear
	Doors Open or Closed		
I	DL + Load from door closers	$F_u/5$	$2\ F_u/15$
II	DL + OBE + Load from door closers DL + SSE + Load from door closers	$0.60\ F_y$ $0.90\ F_y$	$0.40\ F_y$ $0.60\ F_y$
	Doors Closed		
III[3]	DL + 3-psi pressure (bidirectional where applicable)	$0.90\ F_y$	$0.60\ F_y$
IV[4]	DL + OBE + 2-psi toward annulus DL + SSE + 2-psi toward annulus	$0.60\ F_y$ $0.90\ F_y$	$0.40\ F_y$ $0.60\ F_y$
V[5]	DL + 3 inches of water pressure on either side of door	$F_u/5$	$2\ F_u/15$
VI[6]	DL + Flood to elevation 739.7	$0.90\ F_y$	$0.60\ F_y$

1. Thermal load effects are insignificant and hence need not be considered in the design of doors.
2. The values indicated for the allowable compression stresses are the maximum values permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \le C_c \qquad 13$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 14$$

3. Applies to all doors except A64, A65, A77, A78, A56, A60, A111, A117, A118, A122, A125, A130, A133, A151, A160, A162, A183, A192, A206, A207, A208, A209, A212, A213, C37, C49, C50, C53, C60, DE1, DE4 and DE5.
4. Applies to doors A64 and A65 only.
5. For doors A56, A60, A65, A94, A111, A113, A114, A122, A123, A125, A130, A132, A133, A151, A152, A159, A160, A161, A162, A183, A192, A206, A207, A208, A209, A212, A213, A214, A215, DE1, DE4, and DE5, the load combination is:
   DL + 1/2" water pressure on either side of door.
   For doors C36, C37, C49, C50, C53, C54, and C60, the load combination is:
   DL + 1/8" water pressure on either side of door.
6. Applies to door A65 and A78 only.

TABLE 3.8.4-7 (Sheet 8 of 10)
UNIT 2

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(Doors A55, A57, C20, C26, A101, and A105)

Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
	Doors Open			
I	DL + Load from Door Closers	0.50 F_y	0.47 F_y	0.33 F_y
II	DL + OBE + Load from Door Closers	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + Load from Door Closers	0.90 F_y	0.90 F_y	0.60 F_y
	Doors Closed			
III[3]	DL + CCWS Flood + OBE + 3-psi pressure (bidirectional where applicable)	0.60 F_y	0.60 F_y	0.40 F_y
IV[3]	DL + CCWS flood + SSE + 3-psi pressure (bidirectional where applicable)	0.90 F_y	0.90 F_y	0.60 F_y
V[4]	DL + OBE + Pressure from valve rooms	0.60 F_y	0.60 F_y	0.40 F_y
	DL + SSE + Pressure from valve rooms	0.90 F_y	0.90 F_y	0.60 F_y

1. Thermal load effects are insignificant and hence need not be considered in the design of doors.

2. The values indicated for the allowable compression stresses are the maximum values permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 15$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 16$$

3. The CCWS flood condition does not apply to doors A101 and A105, and differential pressure load due to tornado need not be considered simultaneously with seismic load.

4. Applies to doors A101 and A105 only.

TABLE 3.8.4-7 (Sheet 9 of 10)
UNIT 2

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(Doors A55, A57, C20, C26, A101, and A105)

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi): Tension and Compression[2]	Allowable Stresses (psi): Shear
	Doors Open		
I	DL + Load from door closers	$F_u/5$	$2\ F_u/15$
II	DL + OBE + Load from door closers	$0.60\ F_y$	$0.40\ F_y$
	DL + SSE + Load from door closers	$0.90\ F_y$	$0.60\ F_y$
	Doors Closed		
III[3]	DL + CCWS flood + 3-psi pressure (bidirectional where applicable)	$0.90\ F_y$	$0.60\ F_y$
IV[4]	DL + OBE + Pressure from valve room	$0.60\ F_y$	$0.40\ F_y$
	DL + SSE + Pressure from valve room	$0.90\ F_y$	$0.60\ F_y$

1. Thermal Load effects are insignificant and hence need not be considered in the design of doors.

2. The values indicated for the allowable compression stresses is the maximum value permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 17$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2} \text{ when } \frac{Kl}{r} > C_c \qquad 18$$

3. The CCWS flood condition does not apply to doors A101 and A105, and differential pressure load due to tornado need not be considered simultaneously with seismic load.

4. Applies to doors A101 and A105 only.

TABLE 3.8.4-7 (Sheet 10 of 10)
UNIT 2

PRESSURE CONFINING PERSONNEL DOORS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES[1] (Cont'd)

(Door A216 and A217)

Structural Parts

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[2]	Shear
I	DL + P	0.50 F_y	0.47 F_y	0.33 F_y
II	DL + P + OBE	0.60 F_y	0.60 F_y	0.40 F_y
III	DL + P + SSE	0.90 F_y	0.90 F_y	0.60 F_y

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[2]	Shear
I	DL + P	$F_u/5$	2 $F_u/15$
II	DL + P + OBE	0.60 F_y	0.40 F_y
III	DL + P + SSE	0.90 F_y	0.60 F_y

DL - Stresses generated by dead loads and door closer loads.

P - Stresses generated by a pressure differential of 1/2 inch of water acting to open doors.

1. Thermal Load effects are insignificant and hence need not be considered in the design of doors.

2. The values indicated for the allowable compression stresses is the maximum value permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 19$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 20$$

WBN

Table 3.8.4-7a

LIST OF PERSONNEL ACCESS DOORS IN AUXILIARY/CONTROL BUILDING

A55, A57, A64, A65, A77, A78,A94, A95, A96, A101, A105, A113, A114, A115, A123, A132, A152, A153, A154, A155, A156, A157, A158, A159, A161, A164, A165, A173, A184, A191, A204, A214, A215, A216, A217

C20, C26, C29, C34, C36, C54

A111, A117, A125, A130, A151, A160, A162, A183, A192, A206, A207, A208, A209, A212, A213

C37, C49, C50, C53, C60

DE1, DE4, DE5

A56, A60, A122, A133

WBN

TABLE 3.8.4-7b

ABSCE AIR LOCK BOUNDARY DOORS

The following pairs of doors function as ABSCE air lock boundary doors. These doors become part of the ABSCE boundary when the associated airlock boundary door (or damper) is open.

Primary ABSCE Boundary Door	Associated (Extended Air lock Boundary Door
A55	A60
A57	A56
A65	A64[1]
A112 (Railroad access door)	A111, A113, A114
A123	A122
A125	A101
DE1	DE4, DE5[2]
A130	A105
A132	A133
A206	A207
A208	A209
A214	A192
A215	A183

NOTES:

1. Doors A64 and A65 are interlocked to protect the EGTS (Annulus) pressure boundary. A64 is not an ABSCE boundary component.

2. These doors are not required to be interlocked because of the infrequent egress/ingress through them during an ABI.

TABLE 3.8.4-8 (SHEET 1 of 2)

INTAKE PUMPING STATION

LOADING CASES, ALLOWABLE STRESSES, FACTORS OF SAFETY, AND MATERIAL PROPERTIES

Case	Description	WSD Allowable Stresses	Calculated Factors of Safety Overturning	Sliding
I	Reservoir level at El. 690.0, operating loads, earthfill, one pump well unwatered (1/2 of structure)	Normal per ACI 318-71	2.27	3.52
II	Reservoir level at El. 713.0, operating loads, earthfill, Operating Bases Earthquake (OBE)	Normal Concrete per ACI fs = 0.5 fy	1.33	1.39
IIa	Reservoir level at El. 732.1 (assuming upstream dam failure), operating loads, earthfill, OBE both pump wells full	1.35 x Normal per ACI	1.36	2.18
III	Reservoir level at El. 724.4, operating loads, earthfill both pump wells full	1.35 x Normal per ACI	N.A.	N.A.
IV	Reservoir level at El. 675.0, operating loads, earthfill,both pump wells full, tornado	1.67 x Normal per ACI fs = 0.9 fy	2.84	2.96
V	Reservoir level at El. 695.0 (25-year flood), operating loads, earthfill, both pump wells full, safe shutdown earthquake (SSE)	1.67 x Normal per ACI fs = 0.9 fy	1.23	1.10

TABLE 3.8.4-8 (SHEET 2 of 2)

INTAKE PUMPING STATION

LOADING CASES, ALLOWABLE STRESSES, FACTORS OF SAFETY, AND MATERIAL PROPERTIES (Cont'd)

Case	Description	WSD Allowable Stresses	Factors of Safety Overturning	Factors of Safety Sliding
Va	Reservoir level at El. 732.8 operating loads earthfill, both pump wells full, SSE	x Normal per ACI $f_s = 0.9$ Fy	1.11	1.10
VI	Construction condition-dead load of structure, no equipment, earthfill, no ground water	1.33 x Normal per ACI	14.5	4.0
VII	Reservoir level at El. 742.6 (739.2 PMF + 3.4 ft wave run up) operating loads, both pump wells full. (The electrical equipment room begins to flood when the reservoir level exceeds El. 728)	1.67 x Normal per ACI $f_s = 0.9\ f_y$	1.512	4.345

ACI = Allowable stresses per ACI 318-71 Edition (working stress design).
Material Properties

Concrete: $f''_c = 3000$ psi
$w = 145$ pcf
Reinforcing Steel: $f_y = 60{,}000$ psi (ASTM A615, grade 60)

TABLE 3.8.4-9

CONCRETE RETAINING WALLS

LOADING CASES ALLOWABLE STRESSES, FACTORS OF SAFETY, AND MATERIAL PROPERTIES

Case	Description-Reservoir	WSD Allowable Stresses
I	Normal Operating condition - level at El. 675.0	Normal per ACI 318-71
IA	Same as (I) + Operating Basis	$f_c = 0.45\ f'_c$ $f_s = .5\ f_y$
IB	Same as (I) + Safe Shutdown Earthquake	$f_c = .75\ f'_c$ $f_s = .67\ f_y$
II	Construction condition - earth pressure, 200 psf surcharge	$f_c = .5\ f'_c$ $f_s = .5\ f_y$

Material Properties

Concrete: $f''_c = 3000$ psi
$w = 145$ pcf

Reinforcing Steel: $f_y = 60{,}000$ psi (ASTM A615, grade 60)

TABLE 3.8.4-10

SHEET PILE RETAINING WALL

DESIGN LOADINGS, ALLOWABLE STRESSES, MATERIAL PROPERTIES

Case	Description	WSD Allowable Stresses	Allowable Stresses ASTM A36	Allowable Stresses ASTM A328
I	Earth pressure plus 200 psf surcharge	Normal per ACI 318-71	0.8* (AISC allowable)	18,000 psi
II	Same as I plus Operating Basis Earthquake	Normal per ACI 318-71 fs = 0.5 f_y	0.8* (AISC allowable)	18,000 psi
III	Same as I plus Safe Shutdown Earthquake	1.67 x Normal per ACI 318-71 $f_s = 0.9 f_y$	0.8* x F_y	28,000 psi

*Reduced allowable stresses are used to provide corrosion allowance.

Material Properties

ASTM A36 Steel: $F_y = 36{,}000$ psi

TABLE 3.8.4-11 (SHEET 1 of 2)

TRAVELING WATER SCREENS (INTAKE PUMPING STATION)

LOAD COMBINATIONS AND ALLOWABLE STRESSES

Structural Parts

		Allowable Stresses (psi)	
No.	Load Combinations	Tension and Compression(2)	Shear
I	Dead Live with water at El. 683.0 and 2' 6" head loss Impact from live load	0.56 F_y	0.38 F_y
II	For headframe only Dead Live with water at El. 683.0 and 2' 6" head loss Impact from live load Snow and ice	0.56 F_y	0.38 F_y
III	Dead Live with water at El. 713.0 and 2' 6" head loss OBE(1)	0.56 F_y	0.38 F_y
IV	Dead Live with water at El. 695.0 and 2' 6" head loss SSE(1)	0.9 F_y	0.6 F_y
V	Dead Live with water at El. 741.7 and 5' 0" head loss Impact	0.9 F_y	0.6 F_y
VI	Dead Stall at 300% capacity	0.9 F_y	0.6 F_y

TABLE 3.8.4-11 (SHEET 2 of 2)

TRAVELING WATER SCREENS (INTAKE PUMPING STATION)

LOAD COMBINATIONS AND ALLOWABLE STRESSES (Cont'd)

Other Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression(2)	Shear
I	Dead Live with water at El. 683.0 and 2' 6" head loss Impact from live load	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
II	For headframe only Dead Live with water at El. 683.0 and 2' 6" head loss Impact from live load Snow and ice	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
III	Dead Live with water at El. 713.0 and 2' 6" head loss OBE(1)	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
IV	Dead Live with water at El. 695.0 and 2' 6" head loss SSE(1)	$0.9\ F_y$	$2/3 \times 0.9\ F_y$
V	Dead Live with water at El. 741.7 and 5' 0" head loss Impact	$0.9\ F_y$	$2/3 \times 0.9\ F_y$
VI	Dead Stall at 300% capacity	$0.9\ F_y$	$2/3 \times 0.9\ F_y$

Notes:

(1) Acts in one horizontal direction only at any given time and acts in vertical and horizontal directions simultaneously.

(2) The value given for allowable compression stress is the maximum value permitted when buckling does not control. The critical buckling stress. Fcr, shall be used in place of Fy when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^{\ 2}} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8.4-12 (SHEET 1 of 2)

DIESEL GENERATOR BUILDING

LOADS, LOADING COMBINATIONS, ALLOWABLE STRESSES, AND MATERIAL PROPERTIES

Loads

D	=	Dead load of structure including the weight of the diesel generators
L	=	Live load - 200 psf or equipment load in mechanical areas - 300 psf in electrical areas - 20 psf on roof
L_c	=	Construction live load (50 psf on roof)
E	=	Operational basis earthquake (OBE)
E'	=	Safe shutdown earthquake (SSE)
W_T	=	Tornado-generated missiles(1)

TABLE 3.8.4-12 (SHEET 2 of 2)

DIESEL GENERATOR BUILDING

LOADS, LOADING COMBINATIONS, ALLOWABLE STRESSES, AND MATERIAL PROPERTIES (Cont'd)

Load Combinations Case	Description[3]	Allowable Stresses
I	D+L	Normal stresses[2]
II	$D+L_c$	Normal stresses[2] + 33%
III	D+L+E	$f_c = 0.45\ f''_c$
IV	D+L+E'	$f_s = 0.50\ f_y$ $f_c = 0.75\ f''_c$ $f_s = 0.90\ f_y$
V	D+L+Wt	$f_c = 0.75\ f''_c$ $f_s = 0.90\ f_y$

Material Properties

Concrete: f''_c = 3000 psi and (4000 psi for vent hoods)
w = 145 pcf

Reinforcing Steel: f_y = 60,000 psi (ASTM A615, grade 60)

Notes:

(1) The exterior walls and roof are designed to resist missile spectrum A of Table 3.5-7. The precast concrete bulkheads placed in front of the equipment doors are designed to withstand tornado-generated missiles of missile spectrum B in Table 3.5-8 (see discussion in Section 3.8.4.1.2).

(2) Normal stresses are given for working stress design in ACI Code 318-63 or ACI Code 318-71 (See Section 3.8.4.2.2).

(3) Both conditions of L, having its full value or being completely absent, are checked.

TABLE 3.8.4-13 (SHEET 1 of 2)

DIESEL GENERATOR BUILDING DOORS AND BULKHEADS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Structural Parts

No.	Load Combinations	Allowable Stresses(psi): Tension and Compression[4]	Shear
	Door Open or Closed		
I	Dead load	0.6 F_y	0.4 F_y
	Door Closed		
I	Dead Load plus OBE	0.6 F_y	0.4 F_y
III	Dead load plus SSE	0.9 F_y	0.6 F_y

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi): Tension and Compression[4]	Shear
	Door Open or Closed		
I	Dead load	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
	Door Closed		
II	Dead load plus OBE	0.6 F_y	0.4 F_y
III	Dead load plus SSE	0.9 F_y	0.6 F_y

Concrete Bulkheads

No.	Load Combinations	Allowable Stresses: Concrete	Reinforcing Steel
I	Dead Load	1.0 ACI 318	1.0 ACI 318
II	Dead Load plus Wind[2] or OBE	1.0 ACI 318	0.5 F_y
III	Dead Load plus SSE	1.67 ACI 318	0.9 F_y
IV	Dead Load plus Tornado[2]	[3]	[3]

TABLE 3.8.4-13 (SHEET 2 of 2)

DIESEL GENERATOR BUILDING DOORS AND BULKHEADS
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

Notes:

(1) Acts in one horizontal direction only at any given time and acts in vertical and horizontal directions simultaneously.

(2) The steel doors and steel bulkheads are protected from wind, snow, ice, rain, tornado, and wind and tornado missiles by precast concrete bulkheads as discussed in Section 3.8.4.1.2.

(3) The structure may be allowed to yield for load combination IV when considering impactive loads from missiles.

(4) The value indicated for the allowable compression stresses is the maximum value permitted when buckling does not control. The critical buckling stress, Fcr, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8-4-14

DELETED

TABLE 3.8.4-15

PRIMARY AND REFUELING WATER PIPE TUNNELS LOADS, LOAD COMBINATIONS, ALLOWABLE STRESSES, AND MATERIAL PROPERTIES

Loads

D	=	Dead load of structure plus any permanent load contributing stress, such as vertical soil pressure, hydrostatic pressure from ground water Elevation 710, walls; Elevation 726, uplift on slab.
L	=	Surcharge loading from trucks and other equipment operating above tunnel.
D'	=	Hydrostatic pressure from reservoir Elevation 729, maximum level buildings remain unflooded.
R_o	=	Temperature effects on pipe restraints inside tunnel.
E	=	Operating basis earthquake.
E'	=	Safe shutdown earthquake.
Y_r	=	Pipe restraint load due to main steam pipe rupture.
W_t	=	Tornado missile striking earth above top slab of tunnel.

Case	Load Combination	Allowable Stresses
I	D+L (Construction Condition)	$f_c = .45\, f''_c$ (ACI 318-71) $f_s = .5\ f_y$
II	$D+L+R_o+E$	$f_c = .45\, f''_c$ $f_s = .5\ f_y$
III	$D'+L+R_o$	$f_c = .75\, f''_c$ $f_s = .9\ f_y$
IV	$D+L+R_o+E'$	$f_c = .75\, f''_c$ $f_s = .9\ f_y$
V	$D+L+R_o+E'+Y_r$	$f_c = .75\, f''_c$ $f_s = .9\ f_y$
VI	$D+L+R_o+W_t$	$f_c = .75\, f''_c$ $f_s = .9\ f_y$

Material Properties

Concrete f''_c = 3000 and 4000 psi
w = 145 pcf

Reinforcing Steel: f_y = 60,000 psi (ASTM A615, grade 60)

TABLE 3.8.4-16 (SHEET 1 of 2)

CLASS 1E ELECTRIC SYSTEMS STRUCTURES
LOADS, LOAD COMBINATIONS, ALLOWABLE STRESSES,
AND MATERIAL PROPERTIES

STRUCTURES - Manholes

		Design Cases	Allowable Stresses		
I.	SEISMIC OPERATING				
	a.	Dry earthfill plus 1/2 safe shutdown earthquake (1/2 SSE)	f_c f_s	= =	$0.45f\_{c}$ $0.50f_y$
	b.	Earthfill with ground water at elevation 726.0 or finished grade, whichever is lower, plus 1/2 SSE	f_c f_s	= =	$0.45\ f\_{c}$ $0.50\ f_y$
II.	FULL SEISMIC				
	a.	Dry earthfill plus SSE	f_c f_s	= =	$0.75\ f\_{c}$ $0.90\ f_y$
	b.	Earthfill with ground water at elevation 726.0 or finished grade, whichever is lower, plus SSE	f_c fs	= =	$0.75\ f\_{c}$ $0.90\ f_y$
III.	NORMAL OPERATING				
	Earthfill with ground water at elevation 726.0 or finished grade, whichever is lower, plus 200 psf surcharge (or concentrated surcharge where applicable)		f_c f_s	= =	$0.45\ f\_{c}$ $0.40\ f_y$
IV.	TEST				
	Dry earthfill, one compartment of manhole filled with water, water surface elevation 743.5		f_c f_s	= =	$0.50\ f\_{c}$ $0.60\ f_y$
V.	FLOOD				
	Earthfill plus probable maximum flood. No water inside structure		f_c f_s	= =	$0.75\ f\_{c}$ $0.90\ f_y$
VI.	TORNADO LOADING				
	$D+L+W_T$(Vertical Missile)		See UFSAR Section 3.5		
	$D+L+W_T$(Differential Pressure)		f_c f_s	= =	$0.75f'_c$ $0.9f_y$

TABLE 3.8.4-16 (SHEET 2 of 2)

CLASS 1E ELECTRIC SYSTEMS STRUCTURES
LOADS, LOAD COMBINATIONS, ALLOWABLE STRESSES,
AND MATERIAL PROPERTIES (cont'd)

STRUCTURES - Duct Banks

	Design Cases	Required Strength		
I.	SEISMIC OPERATING			
	1/2 SSE (E)	U	=	1.4D + 1.7L + 1.9E
II.	FULL SEISMIC			
	SSE (E_)	U	=	D + L + E_
III.	TORNADO GENERATED MISSILES	U	=	D + L + W_T
IV.	SURCHARGE LOAD L (CRANE OR TRAIN) (D = Dead Load)	U	=	1.7L + 1.4D

Material Properties

Concrete: F_c = 3000 psi
w = 145 pcf

Reinforcing Steel: Fy = 60 ksi (ASTM, A615, Grade 60)

TABLE 3.8.4-17

NORTH STEAM VALVE ROOM
LOADING COMBINATIONS AND ALLOWABLE STRESSES

Load Combinations		WSD Allowable Stresses	USD Load Factors
Case I	= D+L	f_c =.45 f''_c f_s =.40 f_y	1.4D+1.7L
Case II	= D+L+E	f_c =.45 f'' f_s =.50 f_y	1.4D+1.7L+1.9E
Case III	= D+L+E'	*f_c =.75 f''_c f_s =.90 f_y	1.0(D+L+E')
Case IV	= D+L+W_t	*f_c =.75 f''_c f_s =.90 f_y	1.0(D+L+W_t)
Case V	= D+L+P_a	*f_c =.75 f''_c f_s =.90 f_y	1.0D+1.0L+1.5 P_a
Case VI	= D+L+P_a+Y_r+Y_j+Y_m+E	*f_c =.75 f''_c f_s =.90 f_y	1.0D+1.0L+1.25 P_a + 1.0(Y_r+Y_j+Y_m) +1.25E
Case VII	= D+L+P_a+Y_r+Y_j+Y_m+E'	*f_c =.75 f''_c f_s =.90 f_y	1.0(D+L+P_a+Y_r+ Y_j+Y_m+E')

*Concrete stresses other than flexure = 1.67 x normal

The following terms are used in the load combination equations:

D - Dead load of structure and any permanent equipment loads or hydrostatic loads

L - Live loads, including any moveable equipment loads such as soil pressure

E - Operational Basis Earthquake

E' - Safe Shutdown Earthquake

W_t - Tornado, including wind pressure with missiles

P_a - Pressure from postulated main steam pipe break

Y_r - Pipe anchor force due to postulated pipe break

Y_i - Jet force due to postulated pipe break

Y_m - Missile impact force due to postulated pipe break

TABLE 3.8.4-18 (SHEET 1 of 2)

NORTH STEAM VALVE ROOM
STRUCTURAL STEEL LOADING COMBINATIONS AND ALLOWABLE STRESSES

Loading Combinations	Tension Net Section	Shear on Gross Section	Compression on Gross Section	Bending	Concrete Bearing
Case I DL+LL±OBE	0.60 F_Y	0.40 F_Y	See Note 1	0.66 F_Y to 0.60 F_Y	0.25 f''_c
Case II DL+LL+SSE	0.90 F_Y	$\frac{0.9F_y}{\sqrt{3}}$	See Note 2 See Note 3	0.90 F_Y	0.595 f''_c See Note 3

Note 1 - Varies with slenderness ratio; see AISC "Manual of Steel Constructions," 7th Edition, Table 1-36, page 5-84.
Note 2 - Varies with slenderness ratio:

Main and secondary members where KL/r ≤ Cc:

$$F_a = 0.9F_y\left[1-\frac{(KL/r)^2}{2C_c^2}\right] \quad \text{(A)}$$

Main members where Cc < KL/r < 200:

$$F_a = \frac{0.9\pi^2 E}{(KL/r)^2} \quad \text{(B)}$$

Secondary members where 120< KL/r ≤ 200:

$$F_{as} = \frac{F_a\left[by Formula (A) or (B)\right]}{1.6 - L/200r}$$

where:

$$C_c = \sqrt{\frac{2\pi^2 E}{F_y}}$$

E = Modulus elasticity of steel
Fa = Axial compressive stress permitted in the absence of bending moment
Fas =-Axial compressive stress permitted in the absence of bending moment, for bracing and other secondary members.

TABLE 3.8.4-18 (SHEET 2 of 2)

NORTH STEAM VALVE ROOM
STRUCTURAL STEELLOADING COMBINATIONS AND ALLOWABLE STRESSES

F_Y = Specified minimum yield stress of material (kips per square inch)

f''_c = Compressive strength of concrete (kips per square inch)

K = Effective length factor

L = Actual unbraced length (inches)

r = Governing radius of gyration (inches)

Note 3 - When the supporting surface is wider on all sides than the loaded area, the permissible bearing stress on the loaded area may be multiplied by $\sqrt{A_2/A_1}$ 1 but not more than 2.

Where: A_1 = Loaded area (square inches)

A_2 = Maximum area of the portion of the supporting surface that is geometrically similar to and concentric with the loaded area (square inches).

TABLE 3.8.4-19

ERCW STRUCTURES
LOADS, LOAD COMBINATIONS, ALLOWABLE STRESSES,
AND MATERIAL PROPERTIES

LOADS

D = Dead Load
L = Live Loads (loads which vary in intensity and occurrence)
E = Operating Basis earthquake (one-half safe shutdown earthquake)
E' = Safe Shutdown Earthquake
W_t = Tornado Loading (Wind and Missiles and pressure differential as applicable)
W = Loads Generated by the Design Wind for the Plant
L_c = Construction Live Load
= Structure - Slab and Beams Supporting ERCW Pipes

LOAD COMBINATIONS

Design Cases		ASTM A36 Allowable Stresses
Structure - Slabs Supporting ERCW Pipes		
I	u = 1.4 D + 1.7 L	0.6 yield
II	u = 1.4 D + 1.7 L + 1.9 E	0.6 yield
III	u = D+L+E'	0.9 yield
Structure - ERCW Standpipe Structure		
I	u = 1.4 D + 1.7 L + 1.9 E	
II	u = D+L+E'	
III	u = D+W_t	
IV	u = 1.4D + 1.7L + 1.7W	
Structure - ERCW Discharge Overflow Structure		
I	u = 1.4 D + 1.7 L	
II	u = 1.4 D + 1.7 L + 1.9 E	
III	u = D+L+E'	
IV	u = D+W_t	
V_A	u = 1.4 D + 1.7 L + 1.7 W	
V_B	u = 1.2 D + 1.7 W	
VI	u = 1.4 D + 1.4 L_c	
Structure - ERCW Valve Covers		
I	U = 1.4 D + 1.7 L	
II	U = 1.4 D + 1.7 L + 1.9 E	
III	U = D+L+E'	
IV	U = D+W_t	

MATERIAL PROPERTIES

Concrete: f''_c = 3000 or 4000 psi
w = 145 pcf

Reinforcing Steel: f_y = 60 ksi (ASTM A615, Grade 60)

TABLE 3.8.4-20

REFUELING WATER STORAGE TANK FOUNDATION
LOADS, LOAD COMBINATIONS. AND MATERIAL PROPERTIES

LOADS

D	=	Dead Load
L	=	Live Load Including Soil Pressure
E	=	Operating Basis Earthquake
W	=	Design Wind
E'	=	Safe Shutdown Earthquake
W_t	=	Tornado Loading (Wind and Missile)
Y_j	=	Jet Impingement Associated with High-Energy Pipe Break
Y_m	=	Missile Impact Generated by High-Energy Pipe Break

LOAD COMBINATIONS

Design Cases

1	U = 1.4 D + 1.7 L
2	U = 1.4 D + 1.7 L + 1.9 E
3	U = 1.4 D + 1.7 L + 1.7 W
4	U = 1.2 D + 1.9 E
5	U = 1.2 D + 1.7 W
6	U = D+L+E'
7	U = D+L+Wt
8	U = D+L
9	U = D+L+Y_j+Y_m+1.25 E
10	U = D+L+Y_j+Y_m+E'

MATERIAL PROPERTIES

Concrete: f''c = 3000 psi
w = 145 pcf

Reinforcing Steel: f_y = 60 ksi (ASTM A615, Grade 60)

TABLE 3.8.4-21

SPENT FUEL POOL GATES
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

No.	Load Combinations[(1)]	Allowable Stresses lb/in^2 Loading Conditions: Bending	Shear
1	D+L	0.6 F_y	0.4 F_y
2	D+L+OBE	0.6 F_y	0.4 F_y
3	D+L+W	0.6 F_y	0.4 F_y
4	$D+L+T_o+R_o+SSE$	0.9 F_y	0.6 F_y
5	$D+L+T_o+R_o+W_t$	0.9 F_y	0.6 F_y
6	$D+L+T_a+R_a+P_a$	0.9 F_y	0.6 F_y

Notes:

(1) T_o, R_o, T_a, R_a, $P_a = 0$

WBN

TABLE 3.8.4-22

DELETED

WBN

TABLE 3.8.4-23 (Sheet 1 of 2)

WATERTIGHT EQUIPMENT HATCH COVERS

LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

		Allowable Stresses (psi)		
No.	Load Combination	Tension	Compression*	Shear
Hatch Closed				
I	D + 200 lb/ft^2 live load	$0.6F_y$	$0.6F_y$	$0.4F_y$
II	$D + L_1$	$0.9F_y$	$0.9F_y$	$0.6F_y$
III	$D + L_2$	$0.9F_y$	$0.9F_y$	$0.6F_y$
IV	$D + L_1$ + OBE	$0.6F_y$	$0.6F_y$	$0.4F_y$
V	$D + L_1$ + SSE	$0.9F_y$	$0.9F_y$	$0.6F_y$

Where:

D - Dead Loads or their related internal moments and forces including permanent equipment

L_1 - Live Load due to flood to El 711.0

L_2 - Live Load due to pressure of 3 psi from below

OBE - Loads due to the operating basis earthquake

SSE - Loads due to the safe shutdown earthquake

* The value indicated for the allowable compression stresses is the maximum value permitted when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_y when buckling controls.

$$F_{cr} = F_Y \left[1 - \frac{\left(\frac{Kl}{r}\right)^2}{2\ C_c^2} \right] when\ \frac{Kl}{r} \leq C_c \qquad 1$$

or

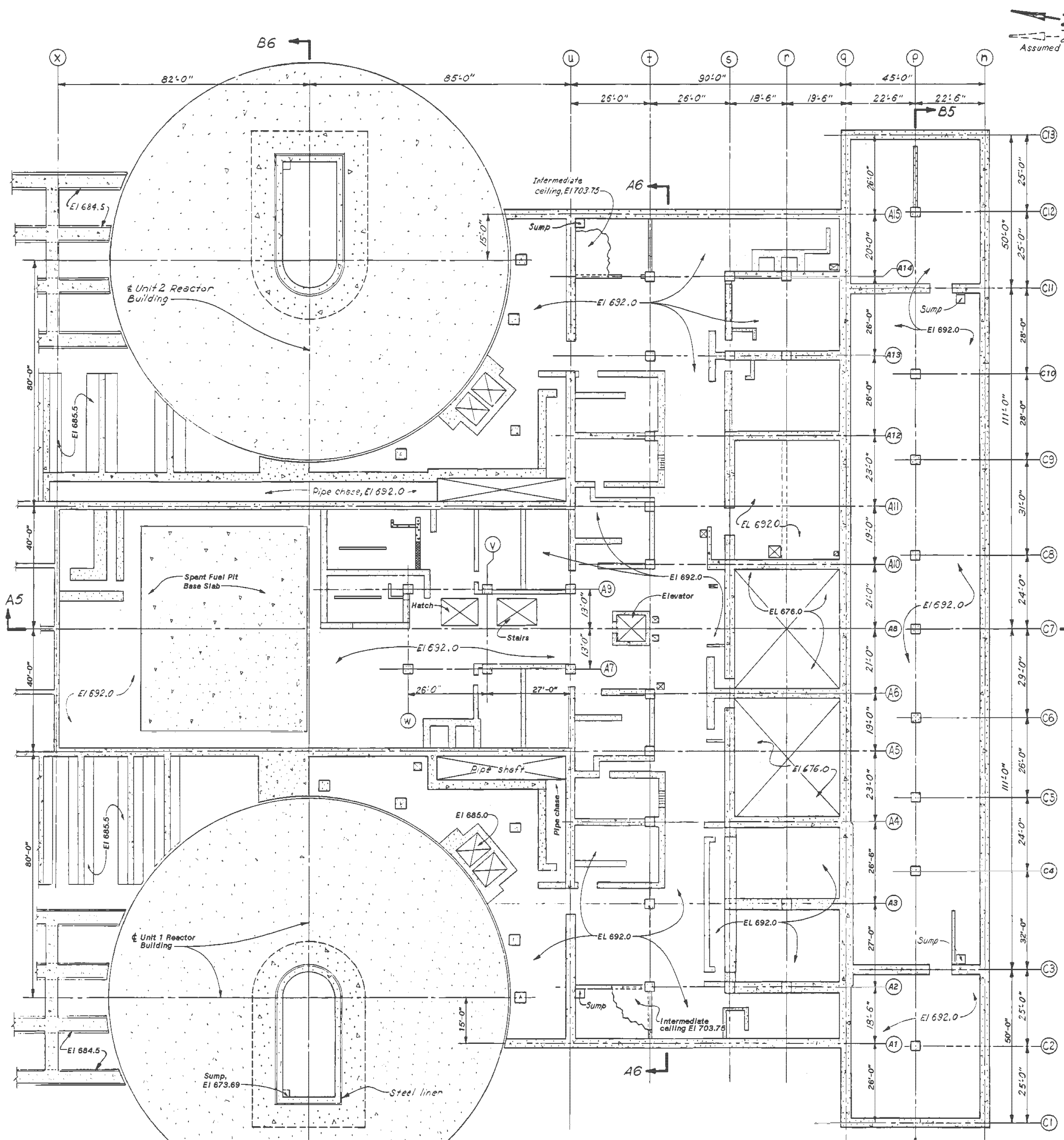
$$F_{cr} = \frac{\pi^2 E}{\left(\frac{Kl}{r}\right)^2}\ when\ \frac{Kl}{r} > C_c \qquad 2$$

TABLE 3.8.4-23 (Sheet 2 of 2)

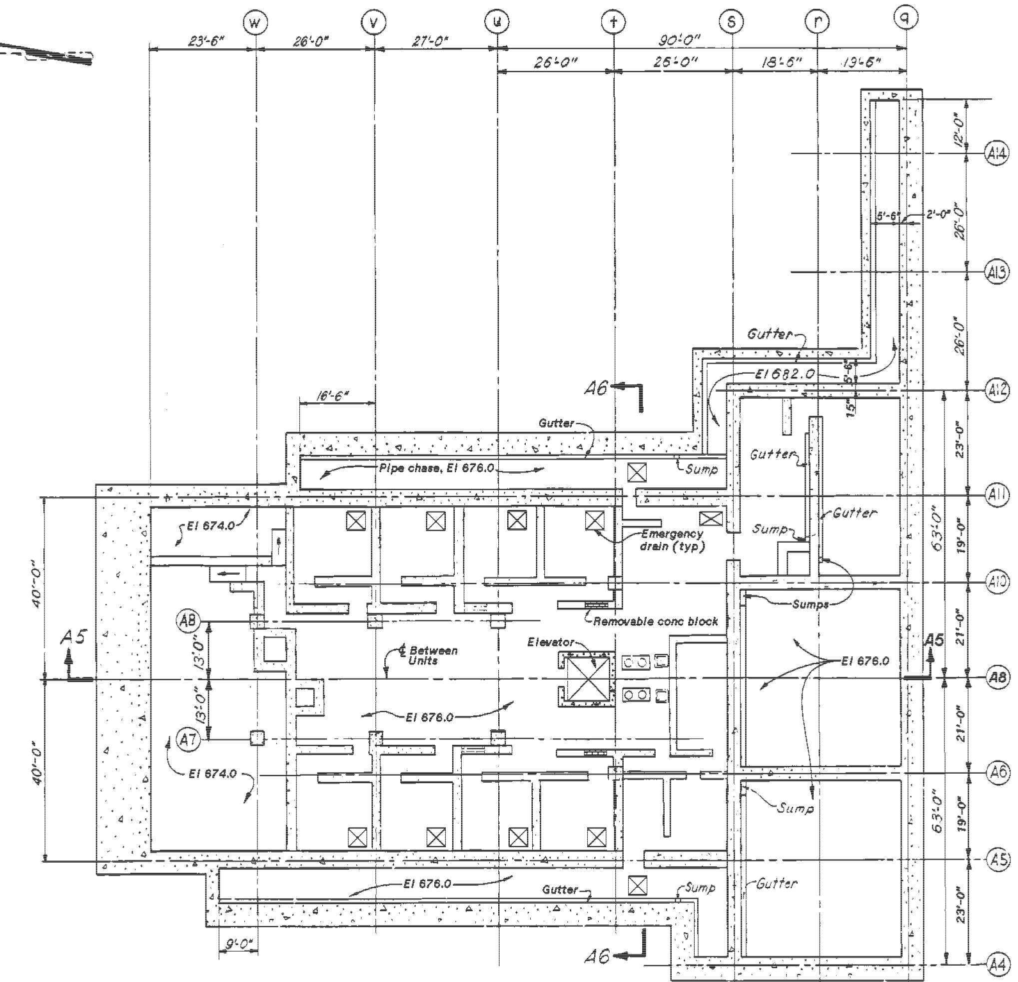
WATERTIGHT EQUIPMENT HATCH COVERS

Material	Serial Designation of the Specifications of the ASTM
Structural Steel	A36
Pipe	A53 or A103 Grade B
Headed Concrete Anchors	1/2" diam x 5-3/16", A108
Steel Screws	A193, Grade B
Seals	Natural or synthetic rubber or combination of natural and synthetic rubber (This is not an ASTM designation)

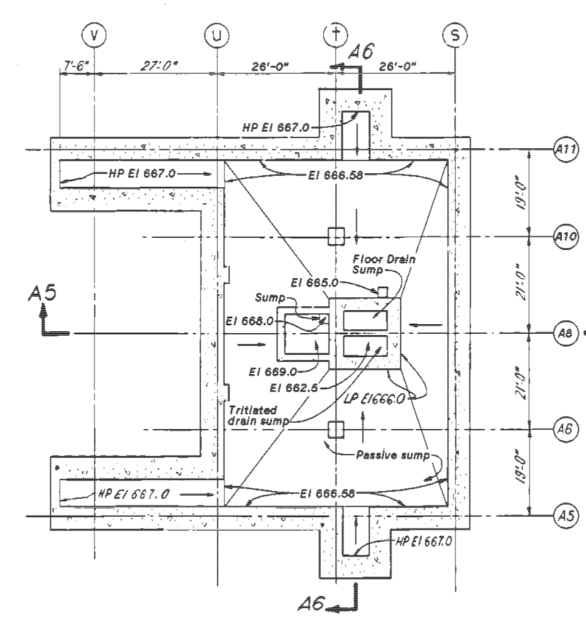
CAD MAINTAINED DRAWING



PLAN - EL 692.0



PLAN - EL 676.0



PLAN - EL 666.0±

NOTES:

1. These drawings are for general information. For details see outline drawings.
2. Construction Specification No. G-2 applies unless otherwise noted. For class of concrete required, see outline drawings.
3. For water seal details, see standard drawing 30B526.
4. Chamfer all exposed corners 3/4", unless otherwise noted.
5. For areas of protective coatings for slabs and walls and surface preparation of concrete required for protective coatings see drawing 46W466-1.
6. Welding of or to reinforcing bars is prohibited, except where specified on the drawings, without prior approval of the Civil Design Branch.
7. Seals in construction joints are PVC, Type I, and shall conform to TVA Specification PF 1026. C seals may be substituted for PVC seals in horizontal construction joints.
8. For location of settlement stations see 10N203-1 & 2.
9. Ground water inleakage where occurring should be controlled to the fullest extent practical by using General Construction Specification G-83 and the other correction methods given in CAQR's WBP 870087 and WBP 870088.
10. NO ATTACHMENTS ARE ALLOWED TO THE REINFORCED CONCRETE PARTITION WALLS OR THE REINFORCED MASONRY BLOCK WALLS WITHOUT A NE-CIVIL APPROVED DESIGN OUTPUT DOCUMENT. DO NOT ATTACH TO REMOVABLE MASONRY BLOCK SHIELD WALLS. PERMANENT SIGNS ARE TO BE PLACED ON THESE WALLS NOTING THESE REQUIREMENTS. THESE SIGNS ARE PART OF THE CORRECTIVE ACTION FOR SCARs WBP870493 R6 AND WBP880766 R4.
11. REPAIR LEAKING EXPANSION AND CONST. JTS. WITH HORNFLEX "L" BY A.C. HORN OR EQUAL TO PROVIDE A WATER TIGHT SEAL IN THE AUXILIARY & CONTROL BUILDINGS.

SYMBOLS:

- Structural walls
- Non-bearing walls (Partition and shielding walls)
- Reinforced conc block walls (all walls not shown)

COMPANION DRAWINGS: 41N700-2 THRU -8

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR, AUX & CONTROL BLDGS

CONCRETE
GENERAL OUTLINE FEATURES

TVA DWG NO. 41N700-1 R8
FIGURE 3.8.4-1

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-2

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-3

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

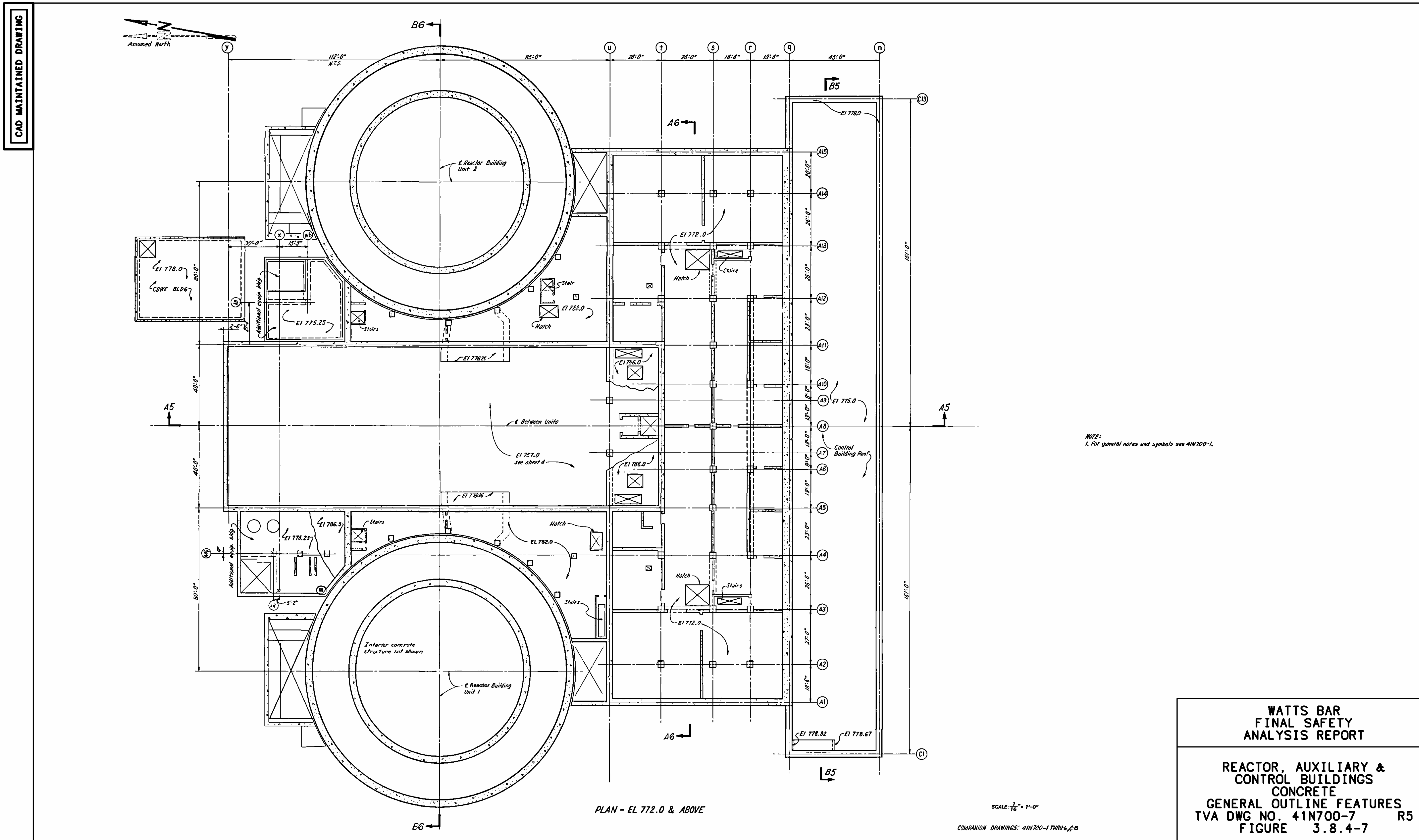
FIGURE 3.8.4-4

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-5

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-6



PLAN - EL 772.0 & ABOVE

NOTE:
1. For general notes and symbols see 41N700-1.

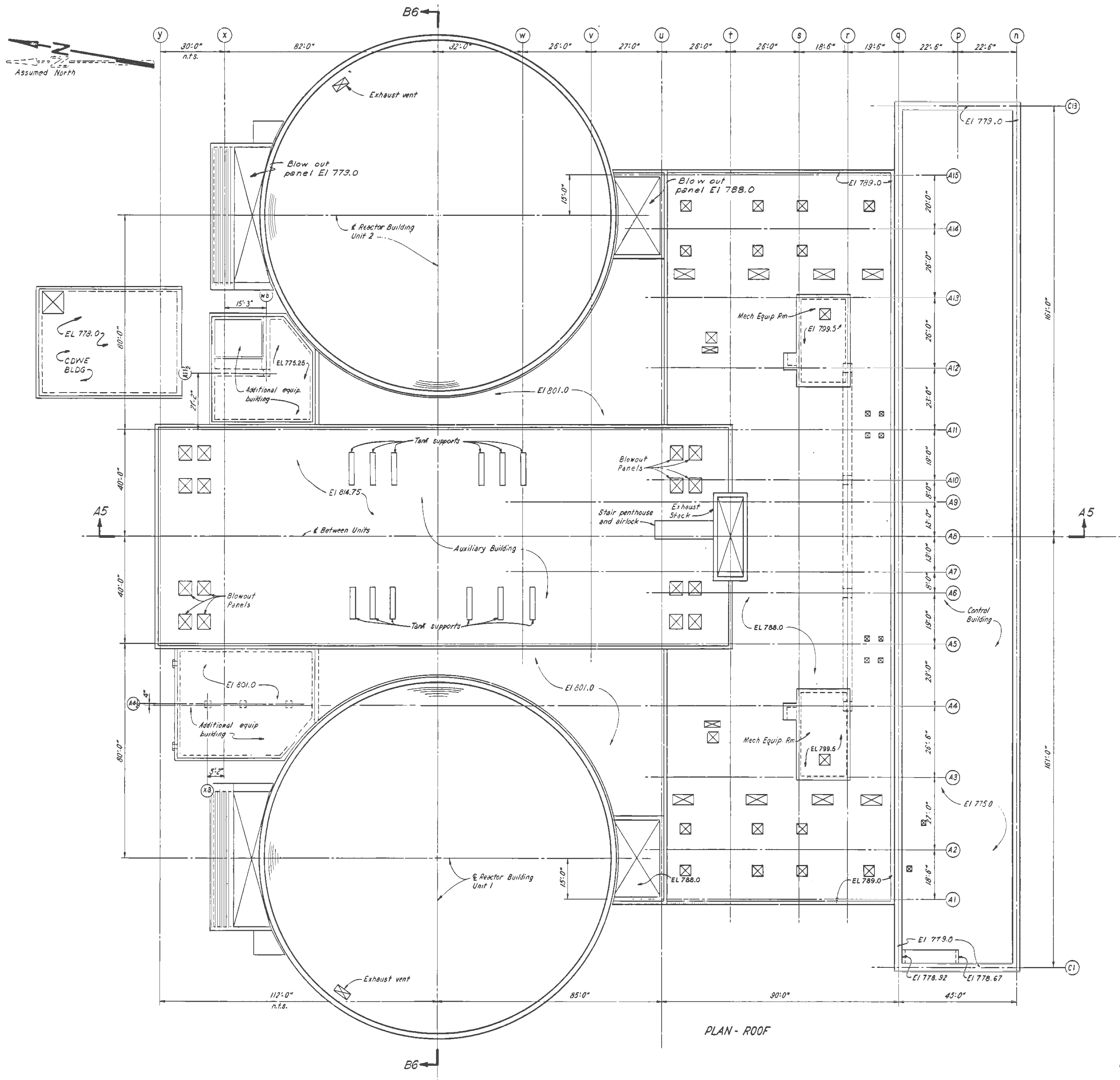
SCALE: 1/16" = 1'-0"

COMPANION DRAWINGS: 41N700-1 THRU 6, & 8

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR, AUXILIARY & CONTROL BUILDINGS
CONCRETE
GENERAL OUTLINE FEATURES
TVA DWG NO. 41N700-7 R5
FIGURE 3.8.4-7

CAD MAINTAINED DRAWING



SCALE: $\frac{1}{16}$" = 1'-0"

COMPANION DRAWINGS: 41N700-1 THRU 7

NOTE:
1. For general notes and symbols see 41N700-1.

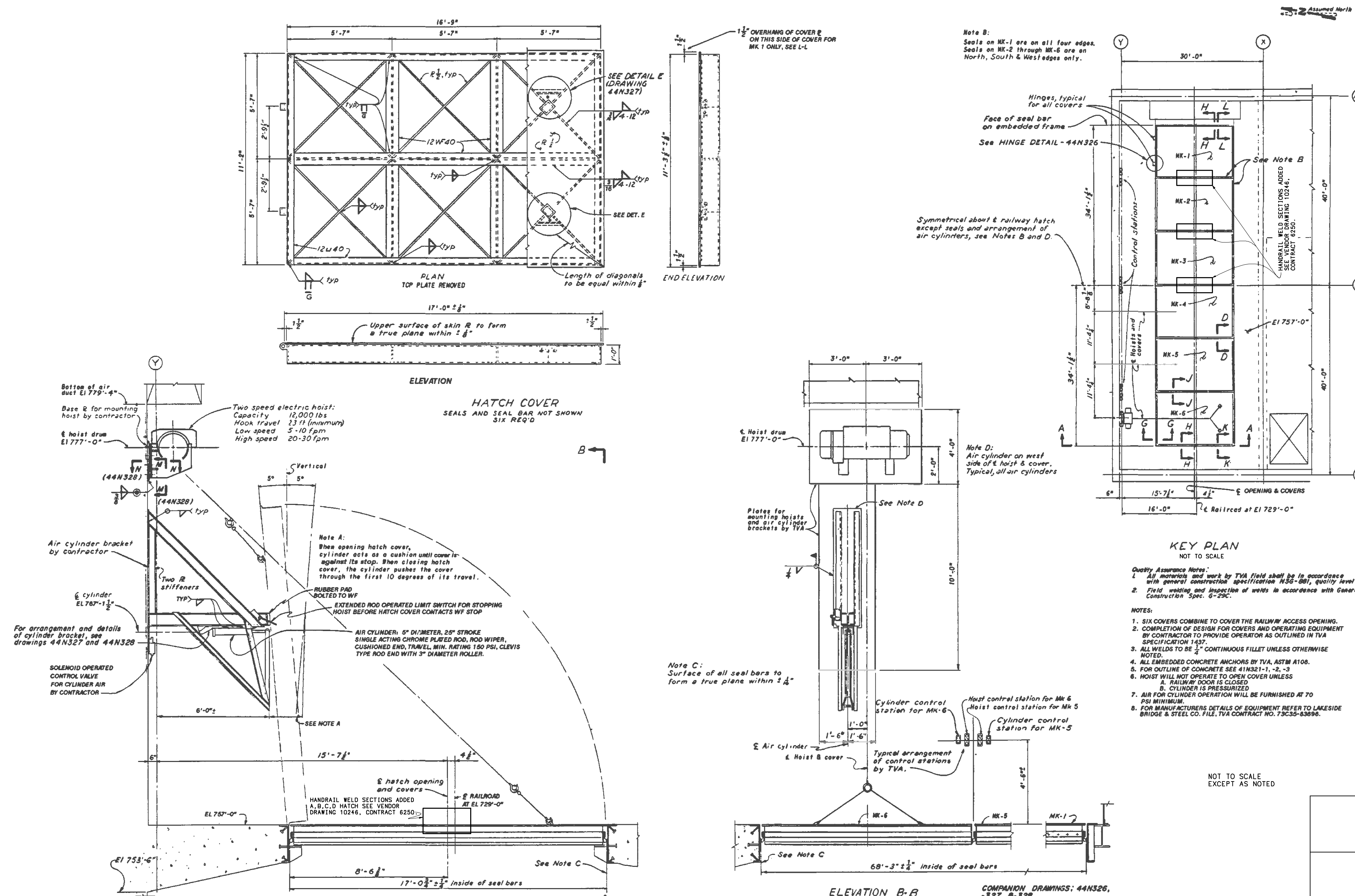
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR, AUXILIARY &
CONTROL BUILDINGS
CONCRETE
GENERAL OUTLINE FEATURES
TVA DWG NO. 41N700-8 R5
FIGURE 3.8.4-8

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-9

CAD MAINTAINED DRAWING



KEY PLAN

NOT TO SCALE

Quality Assurance Notes:

1. All materials and work by TVA field shall be in accordance with general construction specification N3G-881, quality level II.
2. Field welding and inspection of welds in accordance with General Construction Spec. G-29C.

NOTES:

1. SIX COVERS COMBINE TO COVER THE RAILWAY ACCESS OPENING.
2. COMPLETION OF DESIGN FOR COVERS AND OPERATING EQUIPMENT BY CONTRACTOR TO PROVIDE OPERATOR AS OUTLINED IN TVA SPECIFICATION 1437.
3. ALL WELDS TO BE 1/4" CONTINUOUS FILLET UNLESS OTHERWISE NOTED.
4. ALL EMBEDDED CONCRETE ANCHORS BY TVA, ASTM A108.
5. FOR OUTLINE OF CONCRETE SEE 41N321-1, -2, -3
6. HOIST WILL NOT OPERATE TO OPEN COVER UNLESS
   A. RAILWAY DOOR IS CLOSED
   B. CYLINDER IS PRESSURIZED
7. AIR FOR CYLINDER OPERATION WILL BE FURNISHED AT 70 PSI MINIMUM.
8. FOR MANUFACTURERS DETAILS OF EQUIPMENT REFER TO LAKESIDE BRIDGE & STEEL CO. FILE, TVA CONTRACT NO. 73C35-83896.

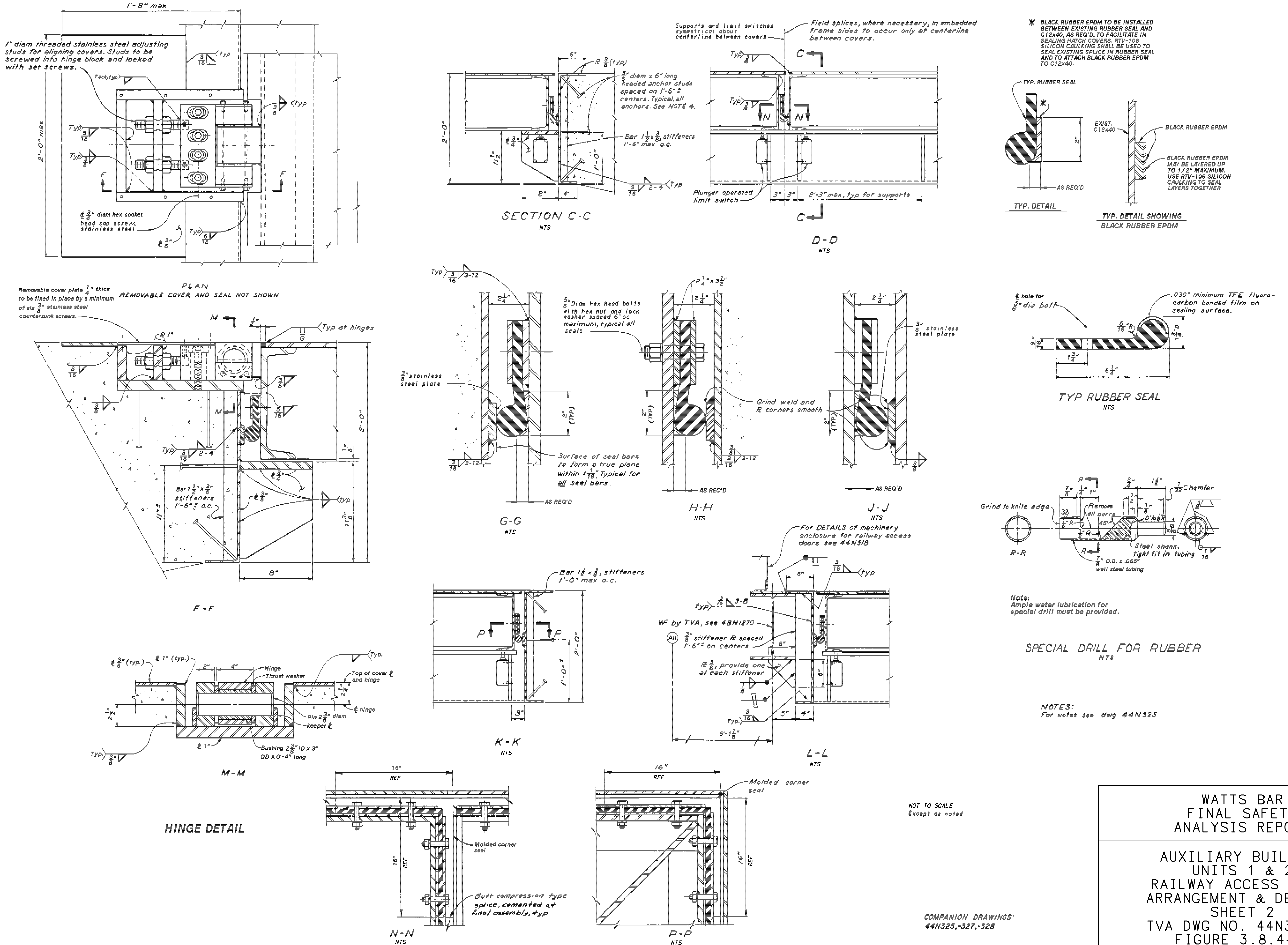
QUALITY ASSURANCE:
HATCH COVERS AND FRAME ARE CLASS 1 SEISMIC EQUIPMENT. ALL MATERIAL FOR CLASS 1 EQUIPMENT REQUIRES COMPLETE QUALITY ASSURANCE DOCUMENTATION. FOR WELDING BY TVA FIELD, WELDING AND INSPECTION OF WELDS DONE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29.

NOT TO SCALE
EXCEPT AS NOTED

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

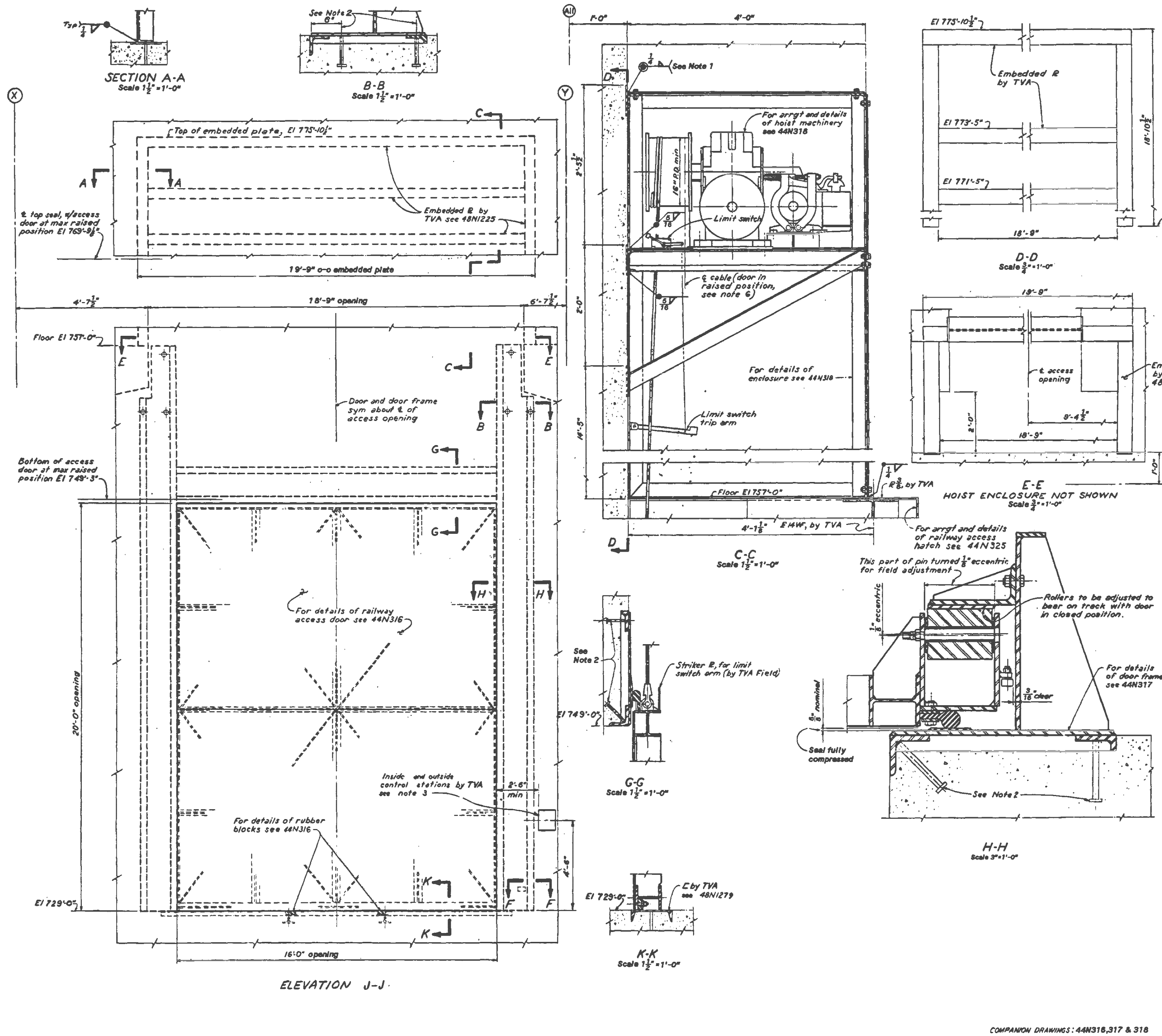
AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS HATCH
ARRANGEMENT & DETAILS
SHEET 1
TVA DWG NO. 44N325 RE
FIGURE 3.8.4-10

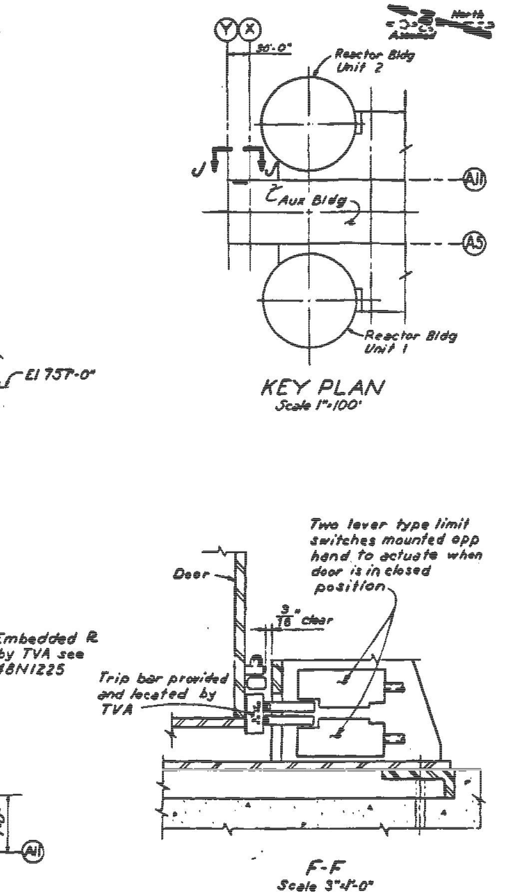


WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS HATCH
ARRANGEMENT & DETAILS
SHEET 2
TVA DWG NO. 44N326 RE
FIGURE 3.8.4-11

CAD MAINTAINED DRAWING





NOTES:

1. MACHINERY ENCLOSURE TO BE WELDED TO EMBEDDED PLATE BY TVA.
2. 5/8" D X 6-3/8" HEADED CONCRETE ANCHORS, ASTM A 108, ARE TO BE LOCATED ON 1'-6"± CENTERS. ANCHORS FURNISHED AND INSTALLED BY TVA FIELD.
3. OUTSIDE STATION TO BE 3-POSITION KEYED SWITCH MOUNTED IN CLOSED BOX BY TVA. INSIDE STATION TO BE FLUSH MOUNTED 4-BUTTON PUSH-BUTTON. KEY to (PLANT OPERATIONS).
4. DOOR IS NOT TO OPERATE UNLESS ALL RAILWAY ACCESS HATCH COVERS AND PERSONNEL ACCESS DOORS TO EL 706'-0" OF AUXILIARY BUILDING ARE CLOSED. ELECTRICAL INTERLOCKING OF DOOR WITH HATCH COVERS AND PERSONNEL BY TVA. SEE 45N775-2.
5. FOR OUTLINE OF CONCRETE SEE DWG 41N321 AND 41N337.
6. CENTERLINE OF CABLE TO PASS THROUGH THE CENTER OF GRAVITY OF DOOR WHEN THE DOOR IS IN THE RAISED POSITION.

QUALITY ASSURANCE:
DOOR AND FRAME ARE CLASS 1 SEISMIC EQUIPMENT. ALL MATERIAL FOR CLASS 1 EQUIPMENT REQUIRES COMPLETE QUALITY ASSURANCE DOCUMENTATION FOR WELDING BY TVA FIELD, WELDING AND INSPECTION OF WELDS DONE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.

ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

FOR MANUFACTURES DETAILS OF EQUIPMENT REFER TO MACHINE PRODUCTS CO. FILE, TVA CONTRACT NO. 73C88-83796-N3H-1.

Scale ½" = 1'-0"
Except as noted

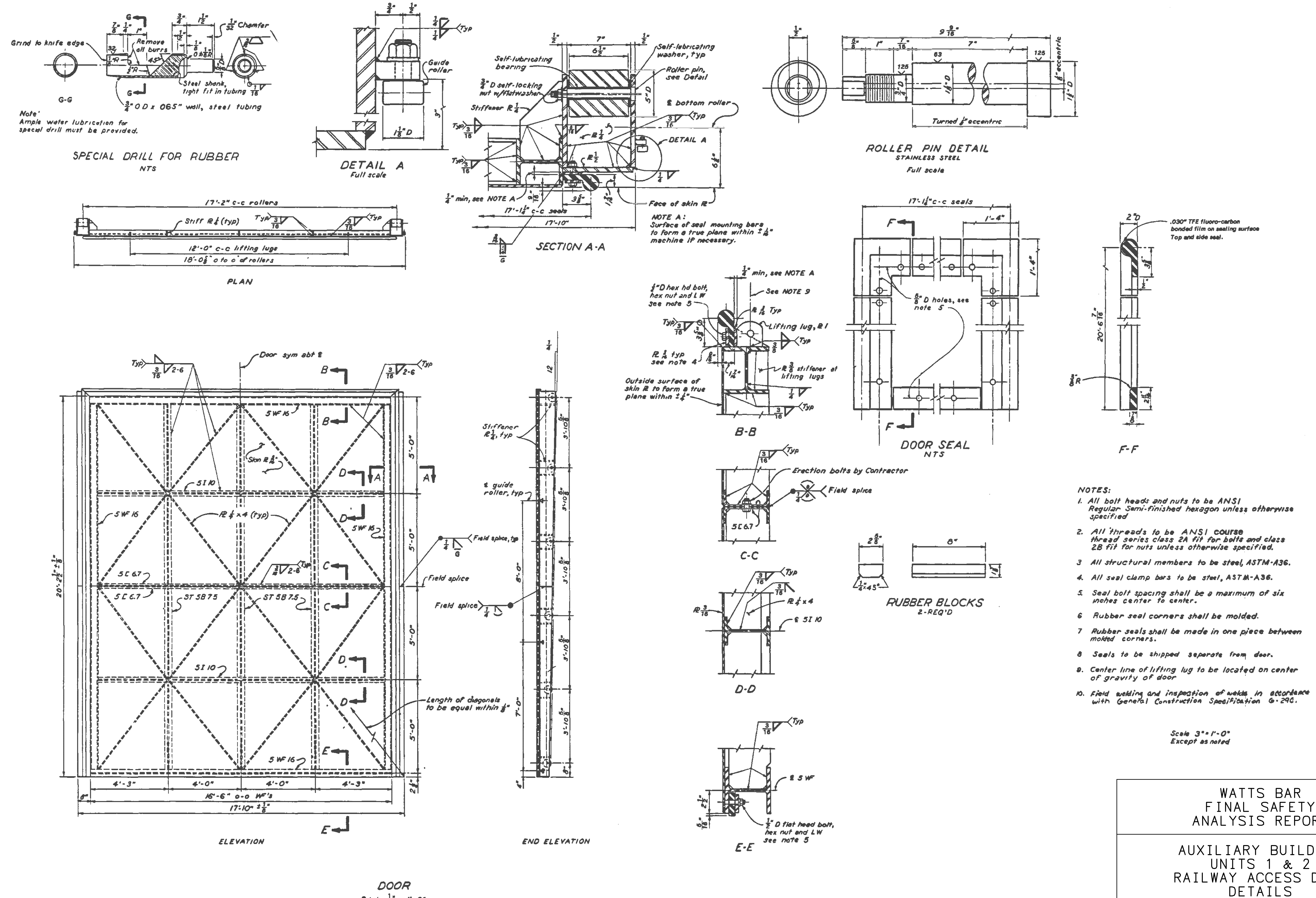
COMPANION DRAWINGS: 44N316,317 & 318

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS DOOR
ARRANGEMENT

TVA DWG NO. 44N315 RC
FIGURE 3.8.4-12

CAD MAINTAINED DRAWING



NOTES:

1. All bolt heads and nuts to be ANSI Regular Semi-finished hexagon unless otherwise specified
2. All threads to be ANSI course thread series class 2A fit for bolts and class 2B fit for nuts unless otherwise specified.
3. All structural members to be steel, ASTM-A36.
4. All seal clamp bars to be steel, ASTM-A36.
5. Seal bolt spacing shall be a maximum of six inches center to center.
6. Rubber seal corners shall be molded.
7. Rubber seals shall be made in one piece between molded corners.
8. Seals to be shipped seperate from door.
9. Center line of lifting lug to be located on center of gravity of door
10. Field welding and inspection of welds in accordance with General Construction Specification G-29C.

Scale 3" = 1'-0"
Except as noted

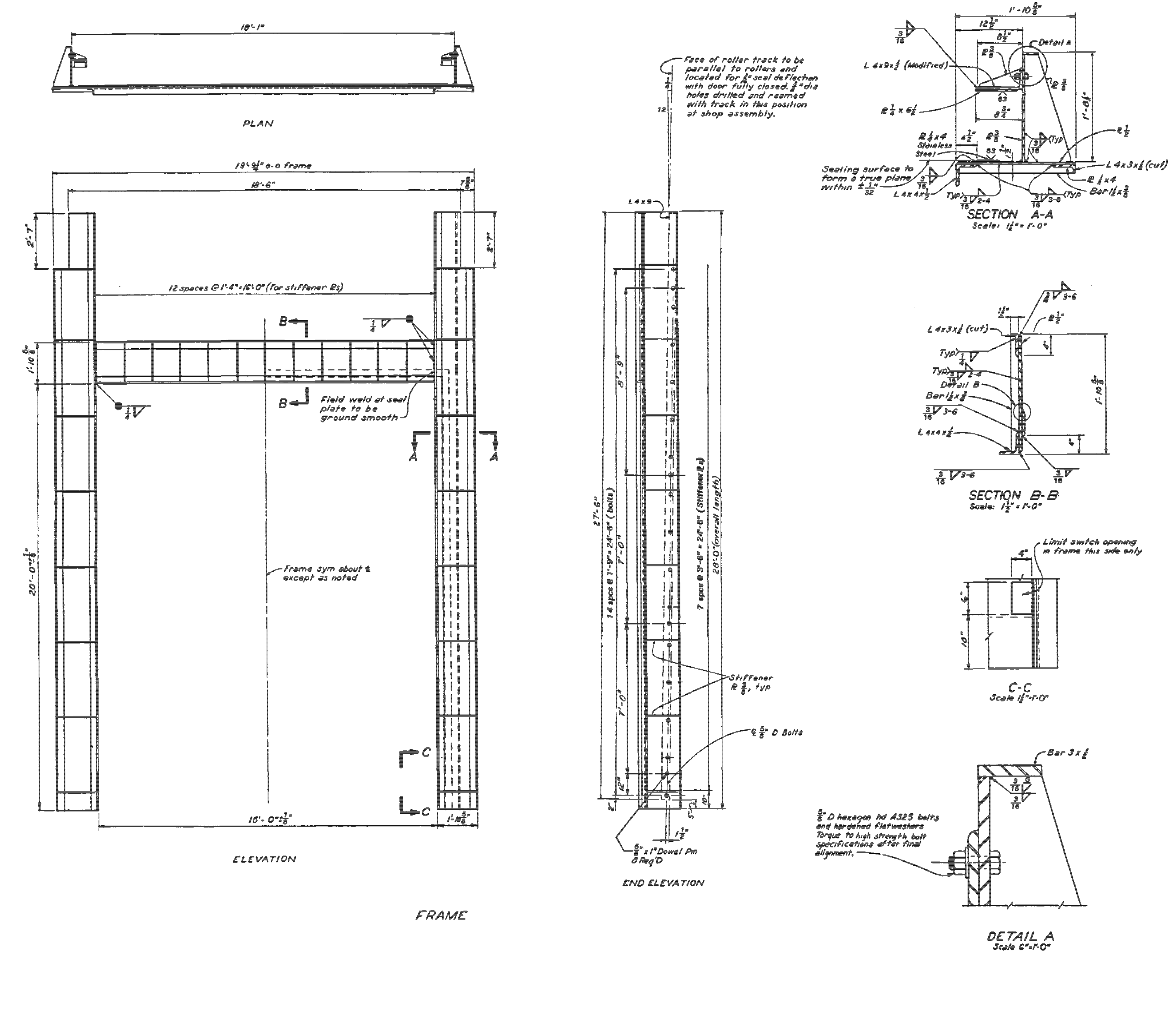
COMPANION DRAWINGS: 44N315, 317 & 318

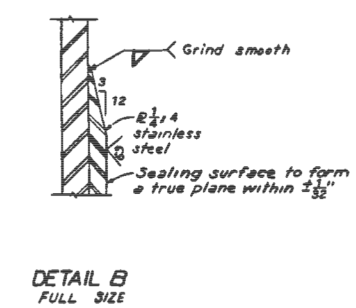
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS DOOR
DETAILS

TVA DWG NO. 44N316 RC
FIGURE 3.8.4-13

CAD MAINTAINED DRAWING





Notes:

1. Field welding and inspection of welds in accordance with General Construction Specification G-29C.
2. Frame to have 16'-0" x 20'-0" clear opening
3. All threads to be ANSI Coarse-Thread Series Class 2A for bolts and Class 2B for nuts unless otherwise noted.
4. All bolt heads and nuts to be ANSI Regular Semi-finished hexagon unless otherwise noted
5. All structural members to be ASTM A36 unless otherwise specified.
6. All materials and work by TVA field shall be in accordance with general construction specification N3G-881, quality level II.

Scale: ¼" = 1'-0"
Except as noted

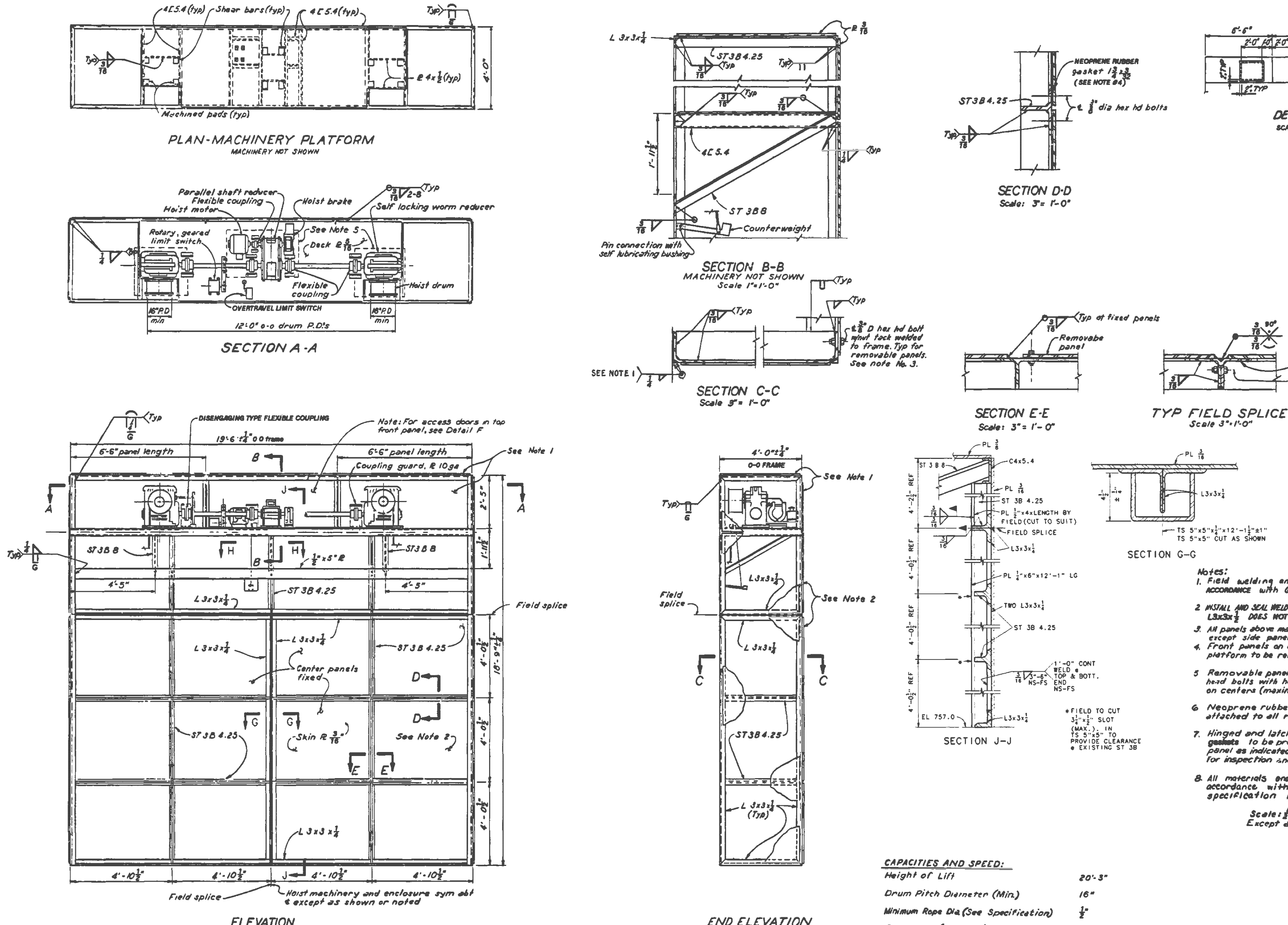
COMPANION DRAWINGS: 44N315, 316, 318

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS DOOR
FRAME DETAILS

TVA DWG NO. 44N317 RC
FIGURE 3.8.4-14

CAD MAINTAINED DRAWING



Notes:

1. Field welding and inspection of welds in accordance with General Construction Specification G-29C.
2. Install and seal weld $\frac{3}{4}$" x $\frac{3}{8}$" bar as required where L3x3x$\frac{1}{4}$ does not interface against embedded plates.
3. All panels above machinery platform to be removable except side panels.
4. Front panels on each side below machinery platform to be removable.
5. Removable panels secured with $\frac{3}{8}$" dia hexagon head bolts with hexagon nuts and lockwasher 9" on centers (maximum).
6. Neoprene rubber gasket, $1\frac{3}{4}$ x $\frac{3}{32}$, permanently attached to all removable and hinged panels.
7. Hinged and latched access doors with neoprene gaskets to be provided in top cover and top front panel as indicated to give access to groups of machinery for inspection and maintenance.
8. All materials and work by TVA field shall be in accordance with general construction specification N3G-881, quality level II.

Scale: $\frac{1}{2}$" = 1'-0"
Except as noted

CAPACITIES AND SPEED:

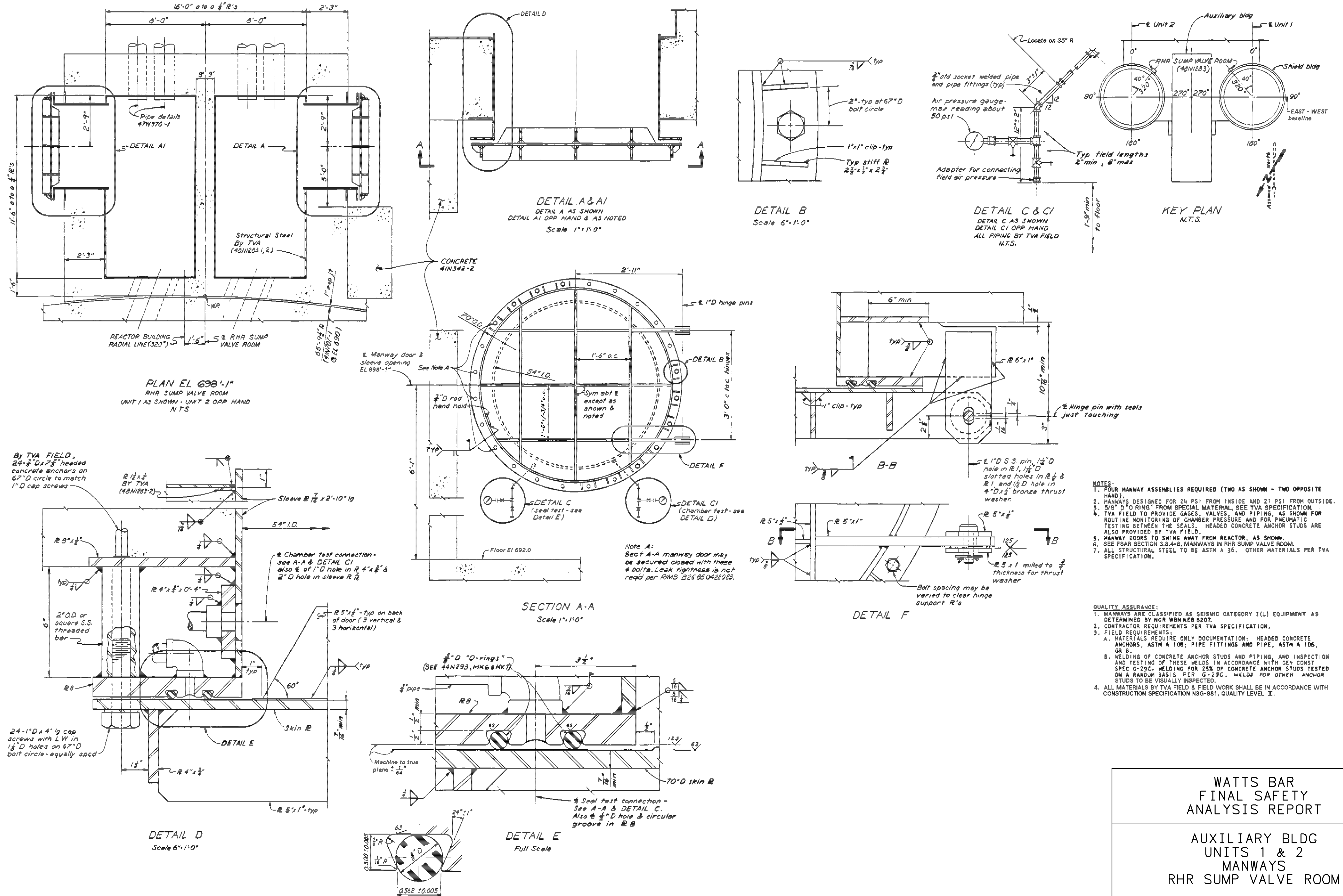
Height of Lift	20'-3"
Drum Pitch Diameter (Min.)	16"
Minimum Rope Dia (See Specification)	$\frac{1}{2}$"
Door Speed (Lowering)	10 fpm (max)
Door Speed (Raising)	9-10 fpm
Motor (Minimum size)	3 HP

COMPANION DWGS: 44N315, -316, -317

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS DOOR
HOIST MACHINERY ENCLOSURE
DETAILS
TVA DWG NO. 44N318 RF
FIGURE 3.8.4-15

CAD MAINTAINED DRAWING



NOTES:
1. FOUR MANWAY ASSEMBLIES REQUIRED (TWO AS SHOWN - TWO OPPOSITE HAND).
2. MANWAYS DESIGNED FOR 24 PSI FROM INSIDE AND 21 PSI FROM OUTSIDE.
3. 5/8" D "O RING" FROM SPECIAL MATERIAL, SEE TVA SPECIFICATION.
4. TVA FIELD TO PROVIDE GAGES, VALVES, AND PIPING, AS SHOWN FOR ROUTINE MONITORING OF CHAMBER PRESSURE AND FOR PNEUMATIC TESTING BETWEEN THE SEALS. HEADED CONCRETE ANCHOR STUDS ARE ALSO PROVIDED BY TVA FIELD.
5. MANWAY DOORS TO SWING AWAY FROM REACTOR, AS SHOWN.
6. SEE FSAR SECTION 3.8.4-6, MANWAYS IN RHR SUMP VALVE ROOM.
7. ALL STRUCTURAL STEEL TO BE ASTM A 36. OTHER MATERIALS PER TVA SPECIFICATION.

QUALITY ASSURANCE:
1. MANWAYS ARE CLASSIFIED AS SEISMIC CATEGORY I(L) EQUIPMENT AS DETERMINED BY NCR WBN NEB 8207.
2. CONTRACTOR REQUIREMENTS PER TVA SPECIFICATION.
3. FIELD REQUIREMENTS:
   A. MATERIALS REQUIRE ONLY DOCUMENTATION: HEADED CONCRETE ANCHORS, ASTM A 108; PIPE FITTINGS AND PIPE, ASTM A 106, GR B.
   B. WELDING OF CONCRETE ANCHOR STUDS AND PIPING, AND INSPECTION AND TESTING OF THESE WELDS IN ACCORDANCE WITH GEN CONST SPEC G-29C. WELDING FOR 25% OF CONCRETE ANCHOR STUDS TESTED ON A RANDOM BASIS PER G-29C. WELDS FOR OTHER ANCHOR STUDS TO BE VISUALLY INSPECTED.
4. ALL MATERIALS BY TVA FIELD & FIELD WORK SHALL BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BLDG
UNITS 1 & 2
MANWAYS
RHR SUMP VALVE ROOM

TVA DWG NO. 44N355 RJ
FIGURE 3.8.4-16

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-17

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

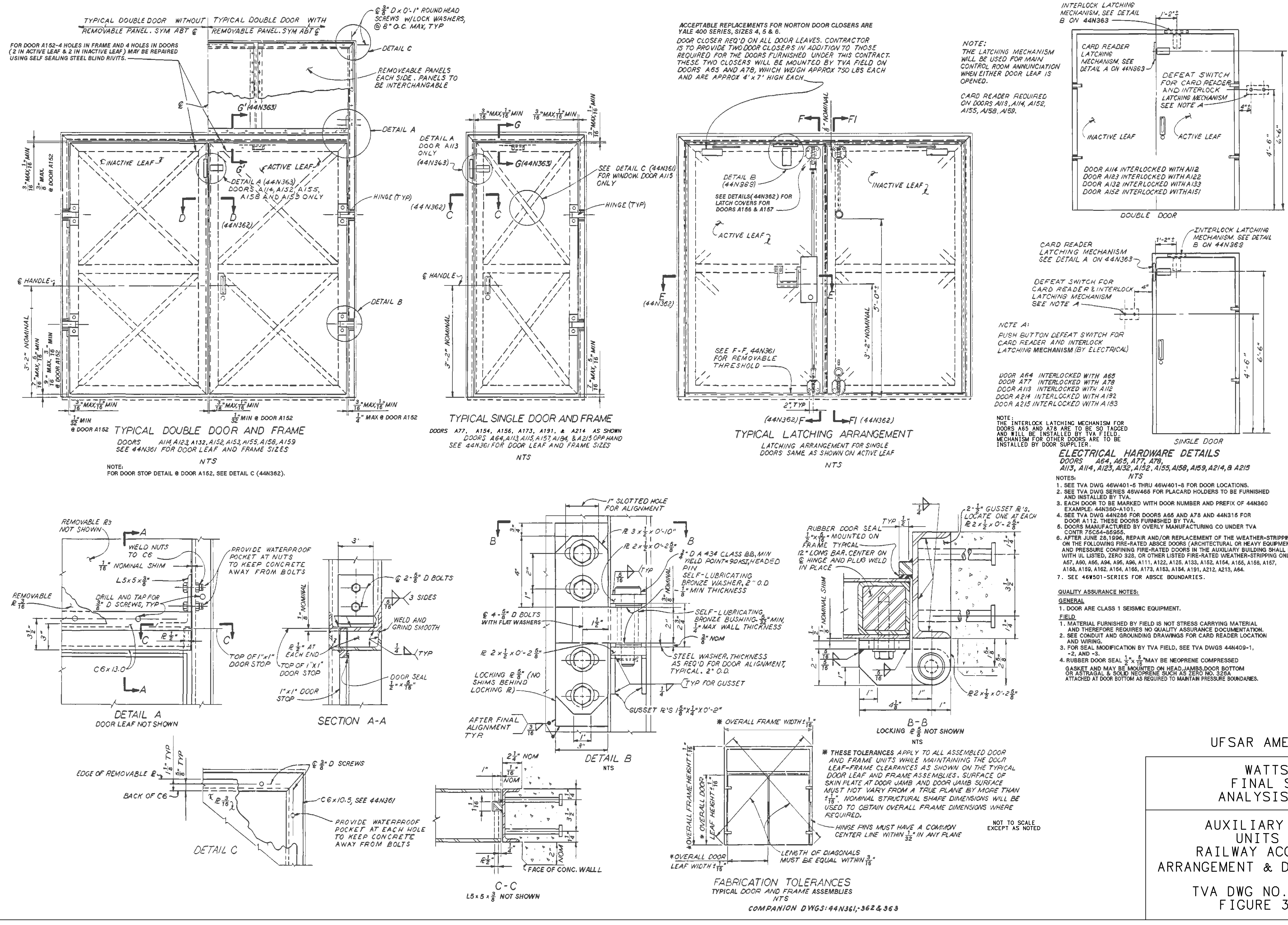
FIGURE 3.8.4-18

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-19

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-20



ELECTRICAL HARDWARE DETAILS

DOORS A64, A65, A77, A78, A113, A114, A123, A132, A152, A155, A158, A159, A214, & A215

NTS

NOTES:

1. SEE TVA DWG 46W401-5 THRU 46W401-8 FOR DOOR LOCATIONS.
2. SEE TVA DWG SERIES 46W465 FOR PLACARD HOLDERS TO BE FURNISHED AND INSTALLED BY TVA.
3. EACH DOOR TO BE MARKED WITH DOOR NUMBER AND PREFIX OF 44N360 EXAMPLE: 44N360-A101.
4. SEE TVA DWG 44N286 FOR DOORS A65 AND A78 AND 44N315 FOR DOOR A112. THESE DOORS FURNISHED BY TVA.
5. DOORS MANUFACTURED BY OVERLY MANUFACTURING CO UNDER TVA CONTR 75C54-85955.
6. AFTER JUNE 28,1996, REPAIR AND/OR REPLACEMENT OF THE WEATHER-STRIPPING ON THE FOLLOWING FIRE-RATED ABSCE DOORS (ARCHITECTURAL OR HEAVY EQUIPMENT) AND PRESSURE CONFINING FIRE-RATED DOORS IN THE AUXILIARY BUILDING SHALL BE WITH UL LISTED, ZERO 328, OR OTHER LISTED FIRE-RATED WEATHER-STRIPPING ONLY: A57, A60, A65, A94, A95, A96, A111, A122, A125, A133, A152, A154, A155, A156, A157, A158, A159, A162, A164, A165, A173, A183, A184, A191, A212, A213, A64.
7. SEE 46W501-SERIES FOR ABSCE BOUNDARIES.

QUALITY ASSURANCE NOTES:

GENERAL

1. DOOR ARE CLASS 1 SEISMIC EQUIPMENT.

FIELD

1. MATERIAL FURNISHED BY FIELD IS NOT STRESS CARRYING MATERIAL AND THEREFORE REQUIRES NO QUALITY ASSURANCE DOCUMENTATION.
2. SEE CONDUIT AND GROUNDING DRAWINGS FOR CARD READER LOCATION AND WIRING.
3. FOR SEAL MODIFICATION BY TVA FIELD, SEE TVA DWGS 44N409-1, -2, AND -3.
4. RUBBER DOOR SEAL $\frac{1}{2}$" x $\frac{5}{16}$" MAY BE NEOPRENE COMPRESSED GASKET AND MAY BE MOUNTED ON HEAD, JAMBS, DOOR BOTTOM OR ASTRAGAL & SOLID NEOPRENE SUCH AS ZERO NO. 326A ATTACHED AT DOOR BOTTOM AS REQUIRED TO MAINTAIN PRESSURE BOUNDARIES.

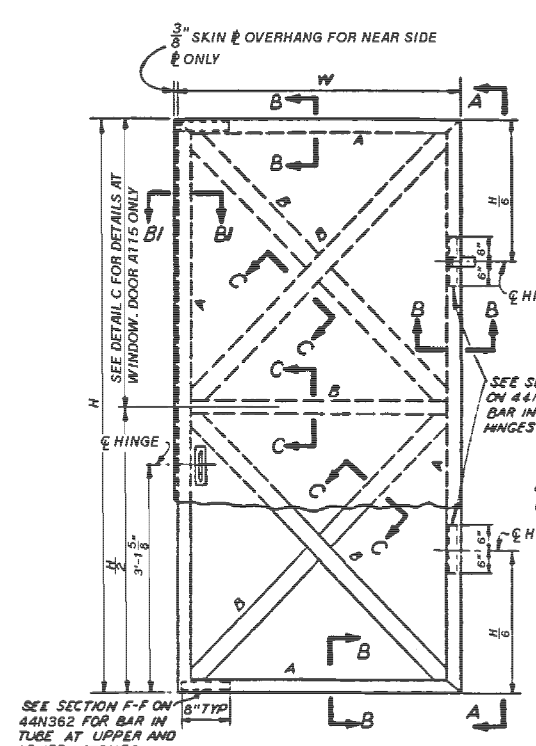
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

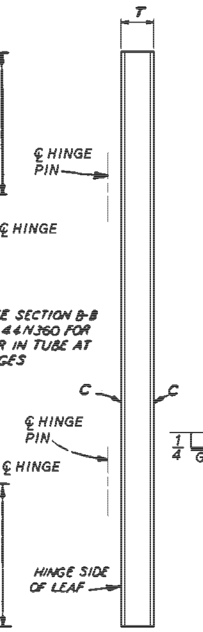
AUXILIARY BUILDING
UNITS 1 & 2
RAILWAY ACCESS HATCH
ARRANGEMENT & DETAILS-SHEET 1

TVA DWG NO. 44N360 RL
FIGURE 3.8.4-21

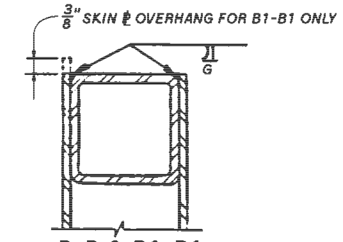
CAD MAINTAINED DRAWING



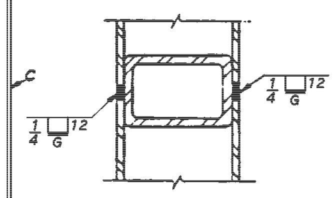
TYPICAL DOOR LEAF
SEE "O" LINE IN SCHEDULE FOR ORIENTATION
NTS



A-A



B-B & B1-B1
B1-B1 AS NOTED



C - C

DOOR LEAF SCHEDULE

DOOR NO.		GROUP 10				GROUP 20					
			*A155	*A152 *A159	*A158	*A123 * A132	*A153	A77, A154, A156 A173, A191, A214	A64, A113, A115 A157, A184, A215	*A114	
SIZE	H		9'-7 $\frac{1}{2}$"	8'-9 $\frac{1}{2}$"	9'-7 $\frac{1}{2}$"	7'-1 $\frac{1}{2}$"	7'-1 $\frac{1}{2}$"	7'-1 $\frac{1}{2}$"	7'-1 $\frac{1}{2}$"	7'-1 $\frac{1}{2}$"	
	W		4'-3 $\frac{7}{16}$"	3'-11 $\frac{7}{16}$"	3'-11 $\frac{7}{16}$"	3'-11 $\frac{7}{16}$"	2'-3 $\frac{7}{16}$"	2'-11 $\frac{3}{8}$"	2'-11 $\frac{3}{8}$"	2'-11 $\frac{7}{16}$"	
	T		4 $\frac{3}{8}$"	4 $\frac{3}{8}$"	4 $\frac{3}{8}$"	3 $\frac{1}{4}$"	3 $\frac{1}{4}$"	3 $\frac{1}{4}$"	3 $\frac{1}{4}$"	3 $\frac{1}{4}$"	
	O		AO	AO	AO	AO	AO	AS	OH	AO	
MATERIAL	A	TUBE 4 X 3 X $\frac{3}{16}$"				TUBE 3 X 3 X $\frac{3}{16}$"					
	B	TUBE 4 X 2 X $\frac{3}{16}$"				TUBE 3 X 2 X $\frac{3}{16}$"					
	C	℄ $\frac{3}{16}$				10 GA SHEET					

NOTES:

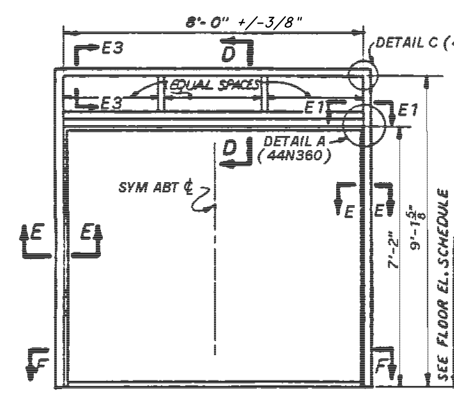
* DENOTES DOUBLE DOORS. ACTIVE LEAF AND INACTIVE LEAF REQ'D FOR THESE DOORS
AO - ACTIVE LEAF AS SHOWN, INACTIVE LEAF OPPOSITE HAND
AS - AS SHOWN
OH - OPPOSITE HAND

FRAME TYPE II SCHEDULE

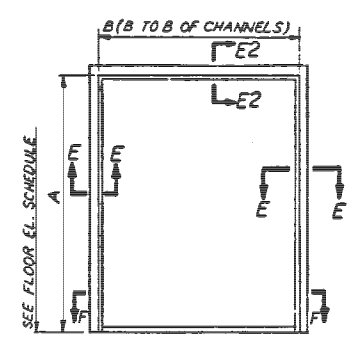
		A173 A184 A191	A152 A159	A155	A158	A64, A77, A113, A115, A154, A156 A157, A214, A215	A153	A114	
A		7'-2"	8'-10"	9'-8"	9'-8"	7'-2"	7'-2"	7'-2"	
B		3'-0"	8'-0"	8'-8"	8'-0"	3'-0"	4'-8"	6'-0"	

FLOOR ELEVATION SCHEDULE

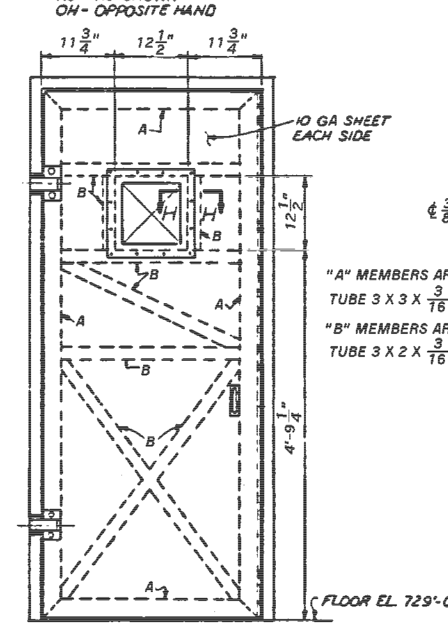
FLOOR EL	713'-0"	729'-0"	737'-0"	757'-0"	772'-0"
DOOR NUMBER	A64		A123	A152	A184
	A77		A132	A153	A191
		A113	A214	A154	
		A114	A215	A155	
		A115		A156	
				A157	
				A158	
				A159	
				A173	



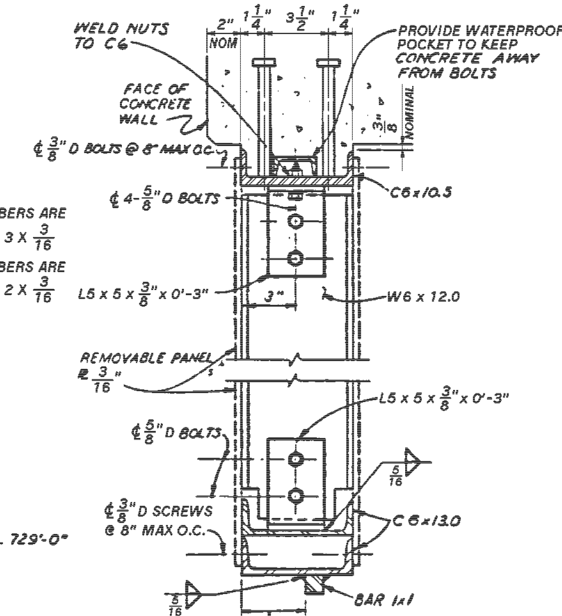
FRAME TYPE I
DOORS A123 & A132 ONLY
REMOVABLE PANELS NOT SHOWN
SCALE: $\frac{1}{2}$" = 1'-0"



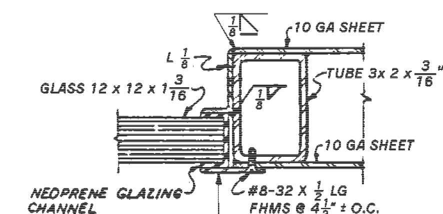
FRAME TYPE II
SEE SCHEDULE
NTS



DETAIL C
WINDOW - DOOR A115 ONLY
SCALE 1" = 1'-0"



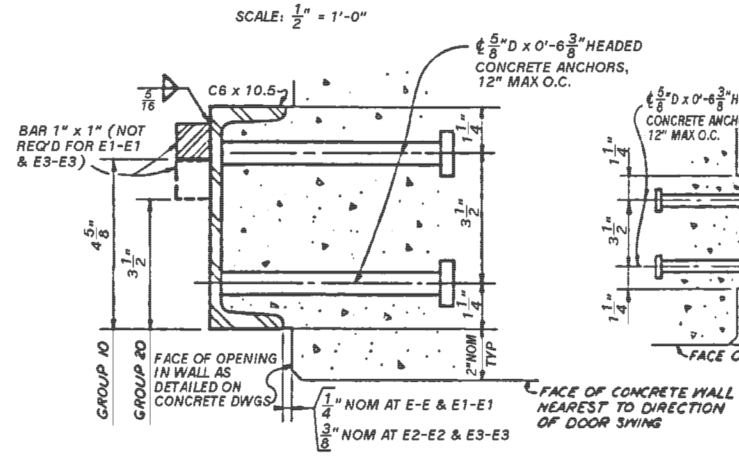
D-D



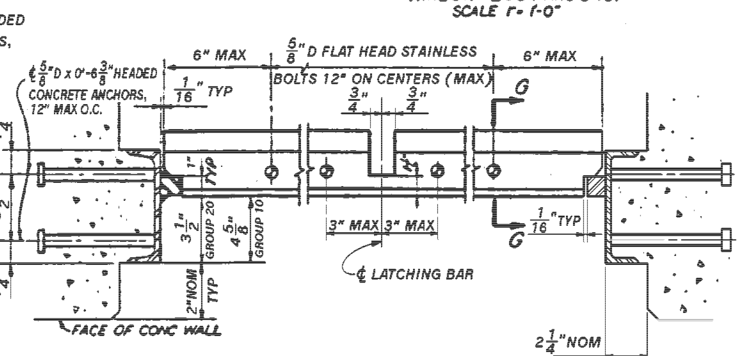
H-H
SCALE: 6" = 1'-0"

NOTES:

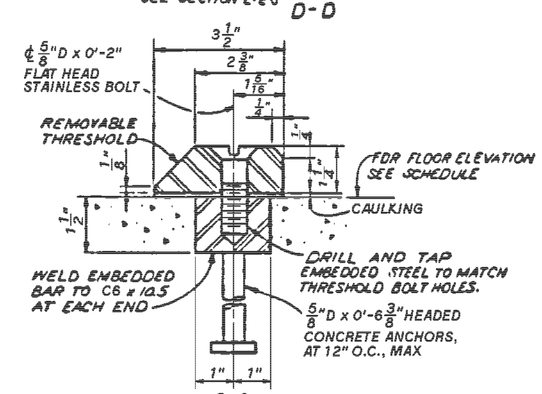
1. SEE TVA DRAWINGS 46W401-5 THRU 46W401-8 FOR DOOR LOCATIONS.
2. SEE 46W501-SERIES FOR ABSCE BOUNDARIES.



E-E, E1-E1, E2-E2, & E3-E3
E1-E1, E2-E2, & E3-E3 AS NOTED
SCALE: 6" = 1'-0"



F-F
(THRESHOLD)



G-G
SCALE: 6" = 1'-0"

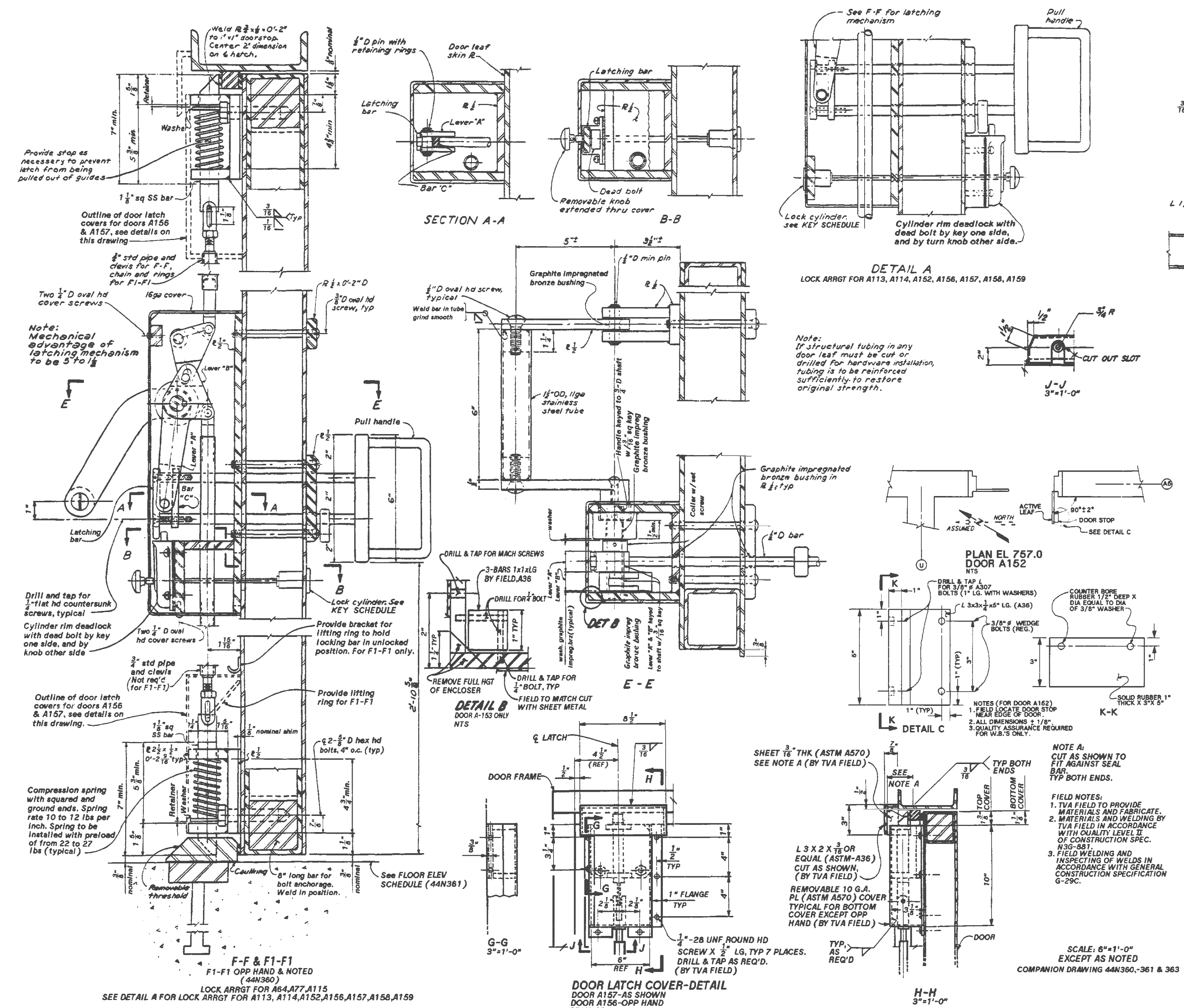
SCALE: 3" = 1'-0"
EXCEPT AS NOTED

COMPANION DRAWINGS: 44N360, -362 & -363

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
AUXILIARY BUILDING
PRESSURE CONFINING
PERSONNEL DOORS
ARRANGEMENT & DETAILS-SHEET 2
TVA DWG NO. 44N361 RL
FIGURE 3.8.4-22

CAD MAINTAINED DRAWING



KEY SCHEDULE			
DOOR NO.	KEY	DOOR NO.	KEY
A64	11b-1	A156	11b-1
A77	11b-2	A157	11b-2
		A158	14b
		A159	17b
		A173	None
		A184	None
A113	1a	A191	None
A114	1a	A214	None
A115	1a	A215	None
A123	None		
A132	None		
A152	17b		
A153	None		
A154	None		
A155	None		

LOCK AND KEY NOTES:

ALL LOCKS KEYED 1a ARE TO BE KEYED ALIKE AND SUBJECT TO GRAND MASTER KEY A AND GREAT GRAND MASTER.

ALL LOCKS KEYED 11b-1 ARE TO BE KEYED SEPARATELY AND SUBJECT TO MASTER B-1 AND GRAND MASTER B.

ALL LOCKS KEYED 11b-2 ARE TO BE KEYED SEPARATELY AND SUBJECT TO MASTER B-2 AND GRAND MASTER B.

ALL LOCKS KEYED 14b ARE TO BE KEYED SEPARATELY AND SUBJECT TO GRAND MASTER B.

ALL LOCKS KEYED 17b ARE TO BE KEYED ALIKE IN GROUPS A. SUBJECT TO GRAND MASTER B.

CYLINDER RIM DEADLOCKS ARE NOT REQUIRED FOR DOORS A123, A132, A153, A154, A155, A173, A184, A191, A214, AND A215.

TVA will furnish name of vendor and manufacturer providing key 1A, grand master key A, master B-1, master B-2, grand master B, and great grand master when this information becomes available.

NOTE A:
CUT AS SHOWN TO FIT AGAINST SEAL BAR.
TYP BOTH ENDS.

FIELD NOTES:
1. TVA FIELD TO PROVIDE MATERIALS AND FABRICATE.
2. MATERIALS AND WELDING BY TVA FIELD IN ACCORDANCE WITH QUALITY LEVEL II OF CONSTRUCTION SPEC. N3G-881.
3. FIELD WELDING AND INSPECTING OF WELDS IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION G-29C.

SCALE: 6"=1'-0"
EXCEPT AS NOTED
COMPANION DRAWING 44N360,-361 & 363

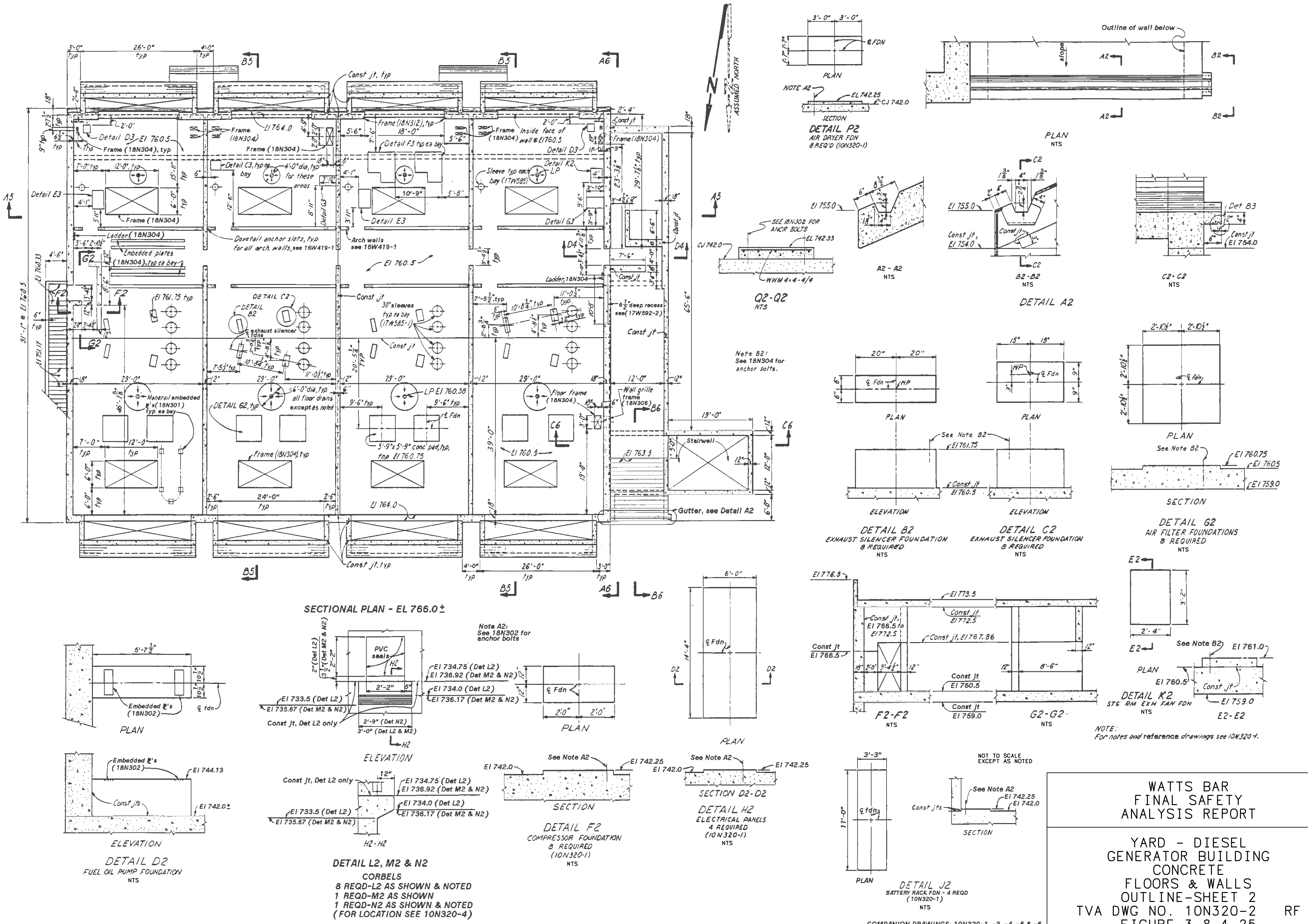
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
AUXILIARY BUILDING
PRESSURE CONFINING
PERSONNEL DOORS
ARRANGEMENT & DETAILS - SHEET 3
TVA DWG NO. 44N362 RF
FIGURE 3.8.4-23

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

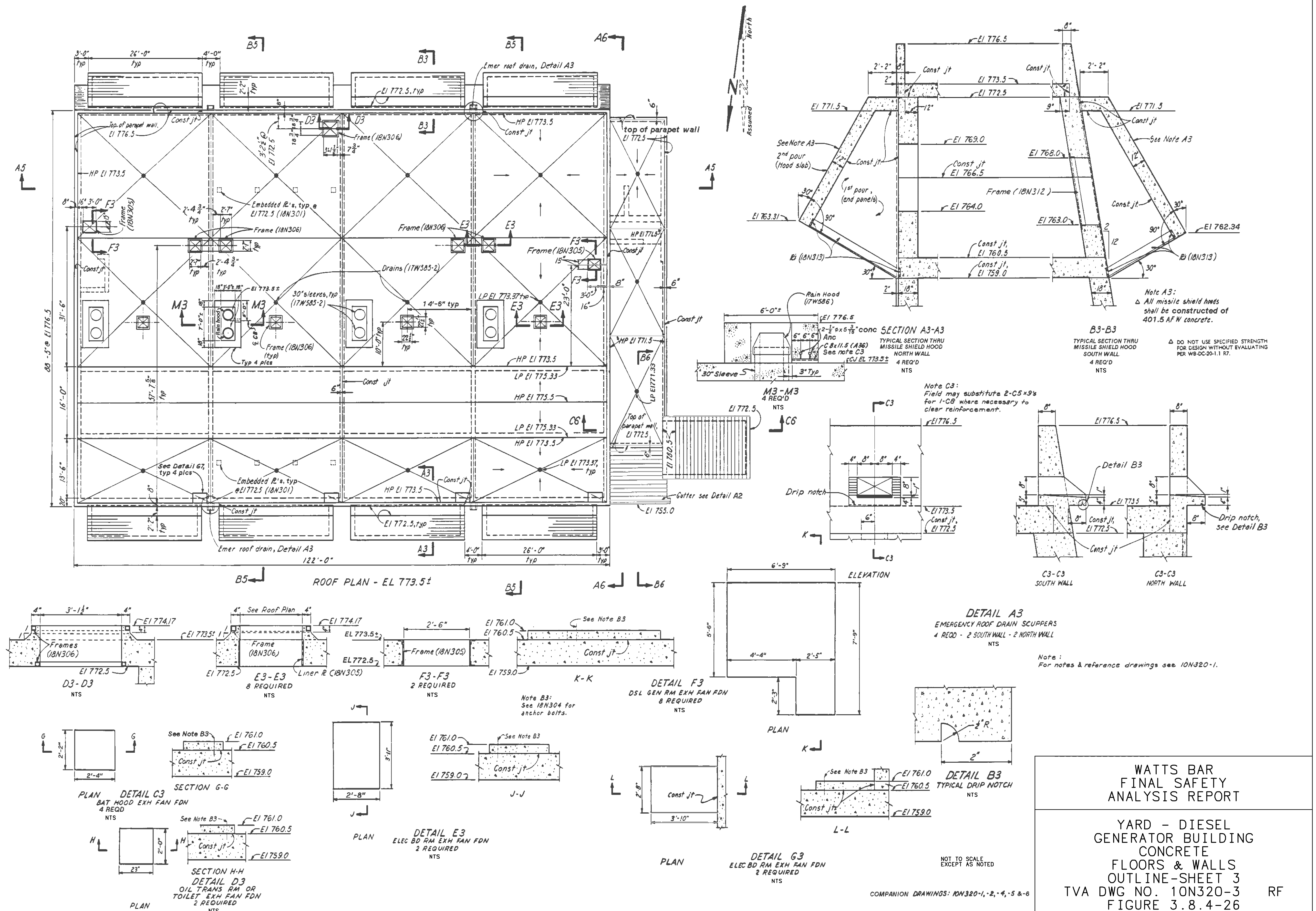
FIGURE 3.8.4-24

CAD MAINTAINED DRAWING



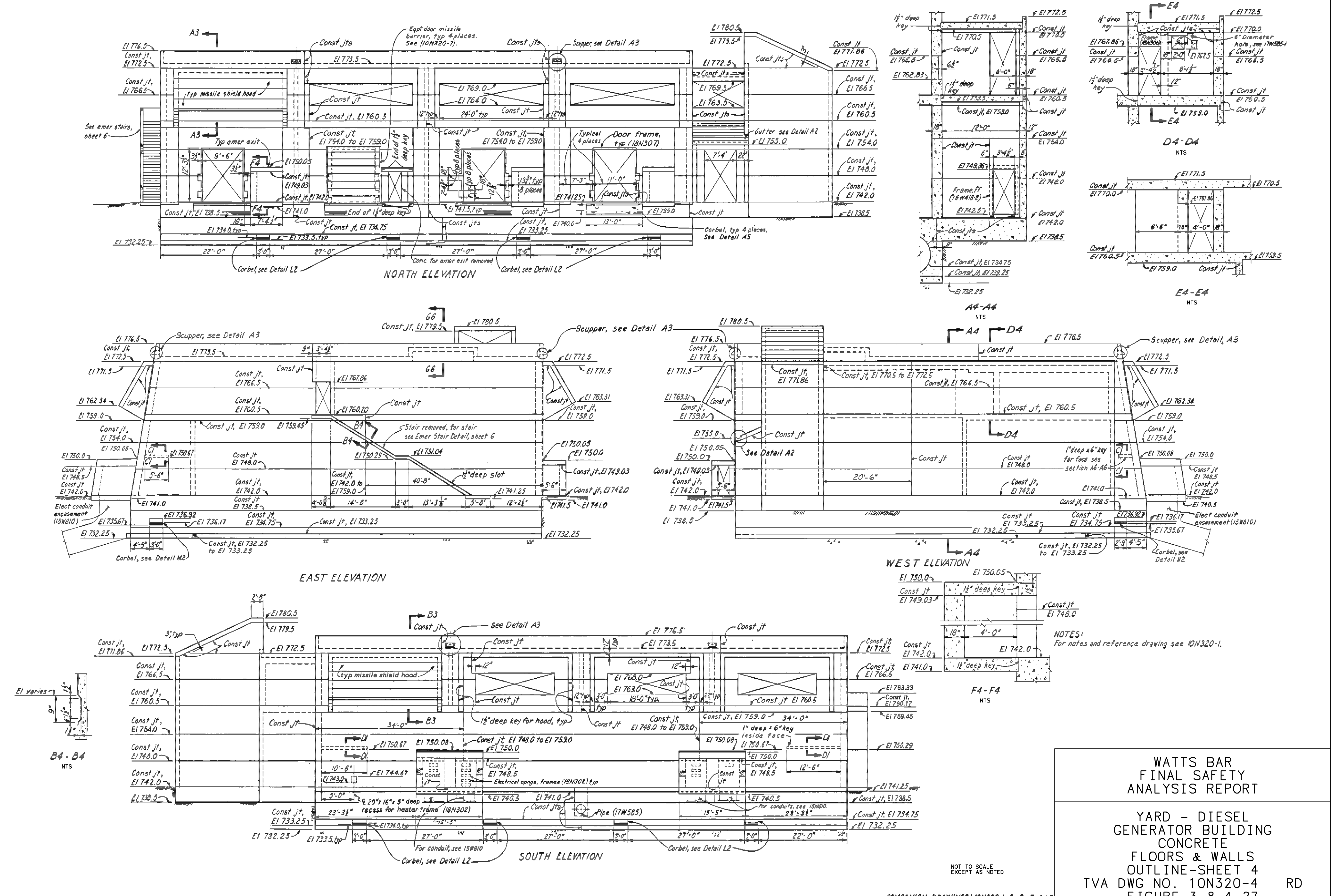
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD - DIESEL
GENERATOR BUILDING
CONCRETE
FLOORS & WALLS
OUTLINE-SHEET 2
TVA DWG NO. 10N320-2 RF
FIGURE 3.8.4-25



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD - DIESEL
GENERATOR BUILDING
CONCRETE
FLOORS & WALLS
OUTLINE-SHEET 3
TVA DWG NO. 10N320-3 RF
FIGURE 3.8.4-26



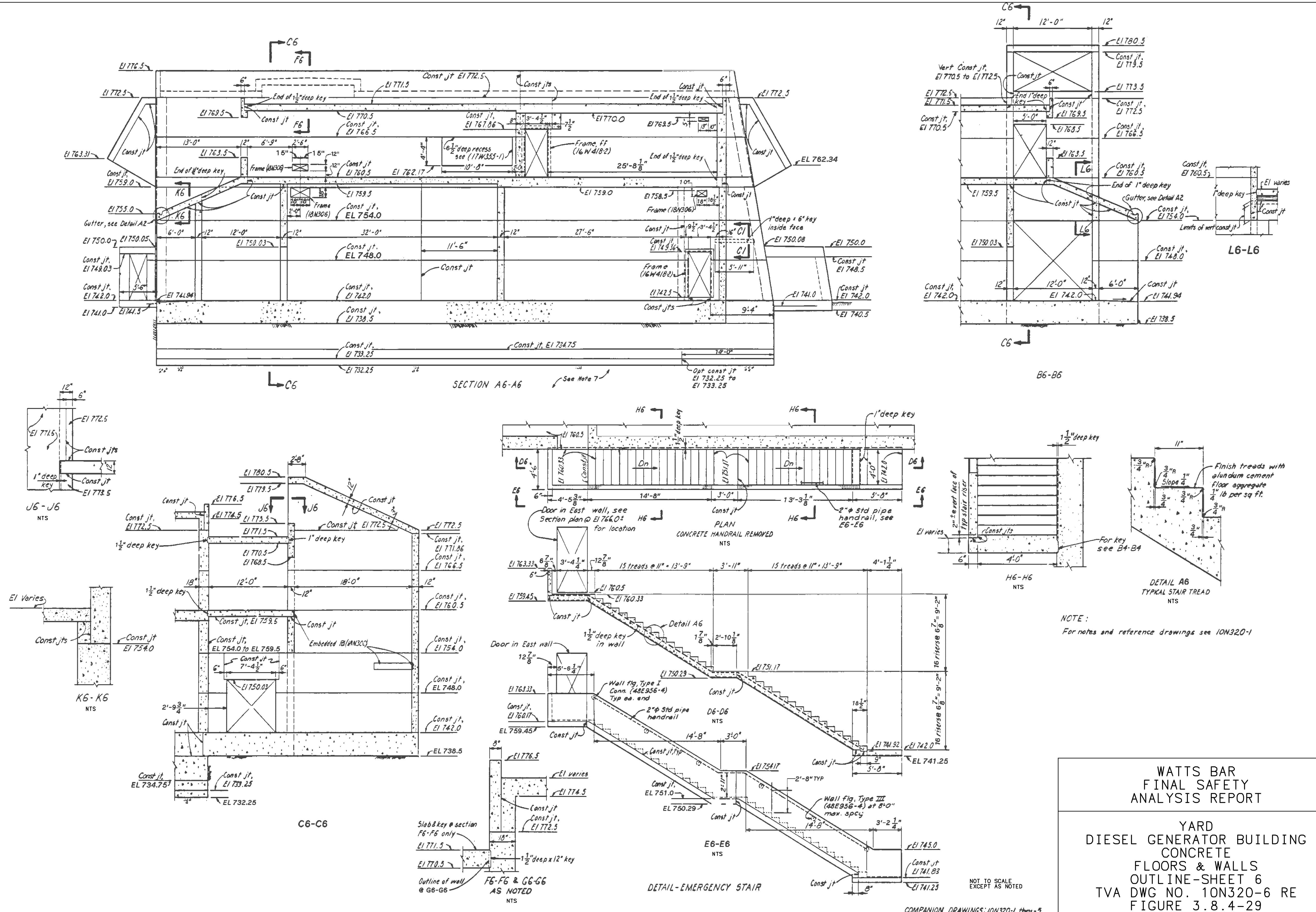
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD - DIESEL
GENERATOR BUILDING
CONCRETE
FLOORS & WALLS
OUTLINE-SHEET 4
TVA DWG NO. 10N320-4 RD
FIGURE 3.8.4-27

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-28

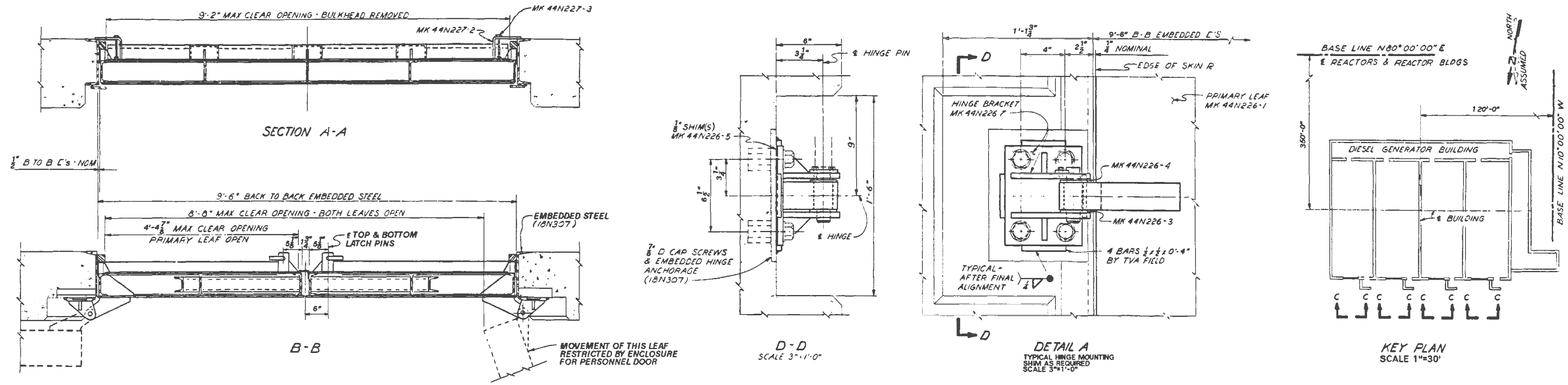
CAD MAINTAINED DRAWING

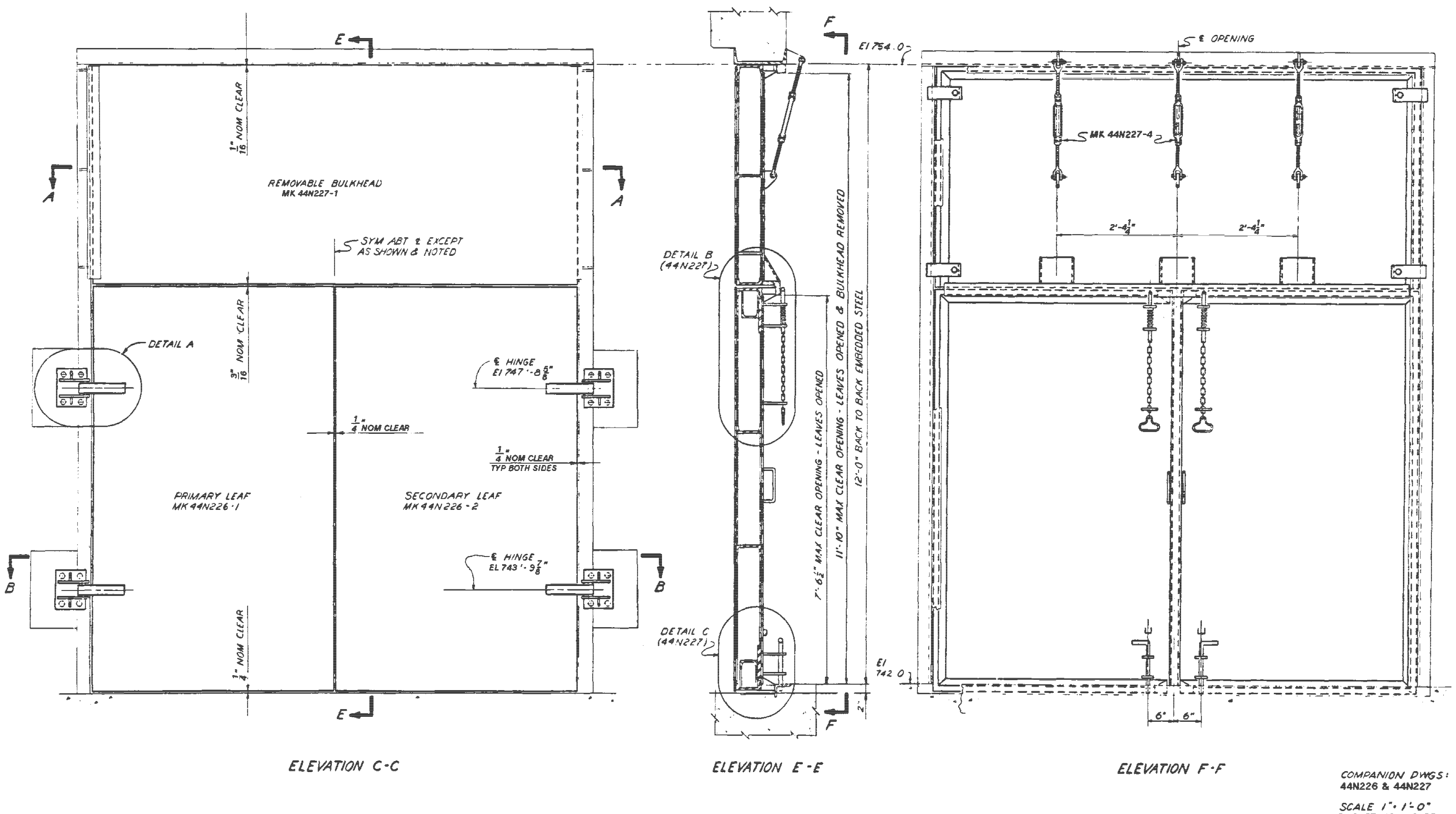


WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
DIESEL GENERATOR BUILDING
CONCRETE
FLOORS & WALLS
OUTLINE-SHEET 6
TVA DWG NO. 10N320-6 RE
FIGURE 3.8.4-29

CAD MAINTAINED DRAWING





COMPANION DWGS:
44N226 & 44N227

SCALE 1"=1'-0"
EXCEPT AS NOTED

NOTES:
1. FOUR DOUBLE-DOOR AND BULKHEAD ASSEMBLIES REQUIRED.
2. ALL MATERIAL STRUCTURAL STEEL EXCEPT AS OTHERWISE NOTED.
3. ALL WELDS MADE WITH AWS A5.1, E70 ELECTRODES.

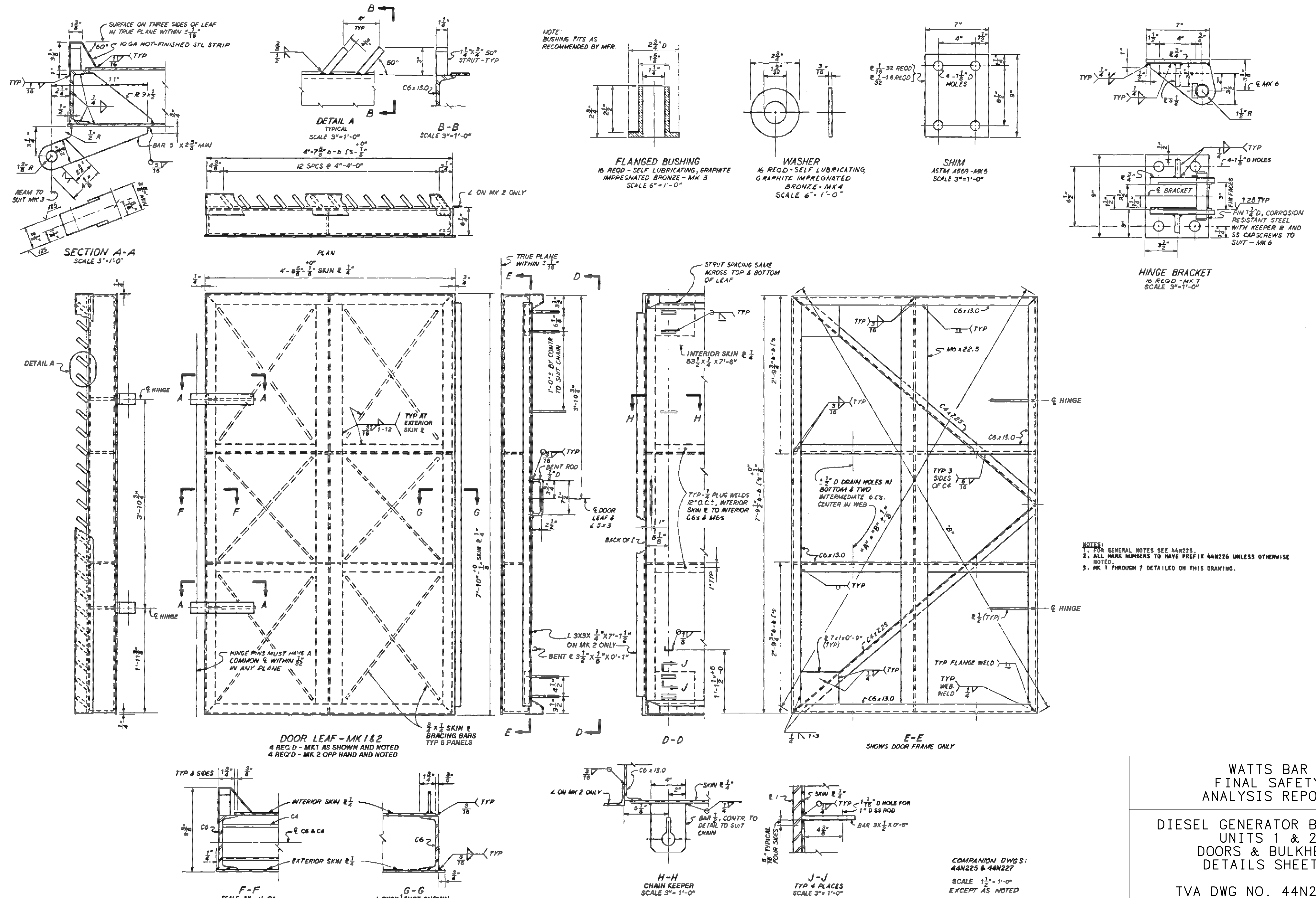
QUALITY ASSURANCE

1. DOORS AND BULKHEADS ARE CATEGORY I EQUIPMENT.
2. CONTRACTOR REQUIREMENTS IN ACCORDANCE WITH TVA SPEC 2409.
3. FIELD REQUIREMENTS:
   WELDING AND VISUAL INSPECTION OF WELDS IN ACCORDANCE WITH GEN CONST SPEC. G-29C
4. ALL MATERIALS BY TVA FIELD & FIELD WORK SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

DIESEL GENERATOR BUILDING
UNITS 1 & 2
DOORS & BULKHEAD
ARRANGEMENT

TVA DWG NO. 44N225 RD
FIGURE 3.8.4-30

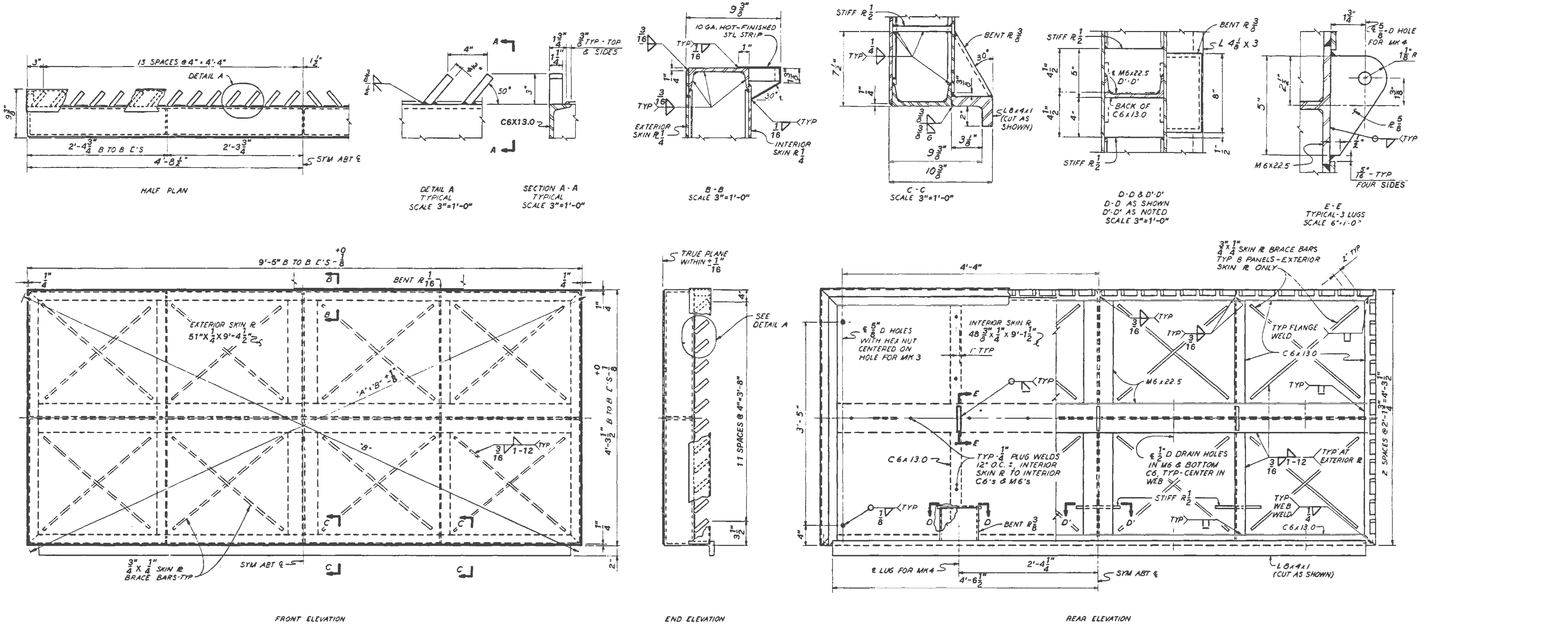


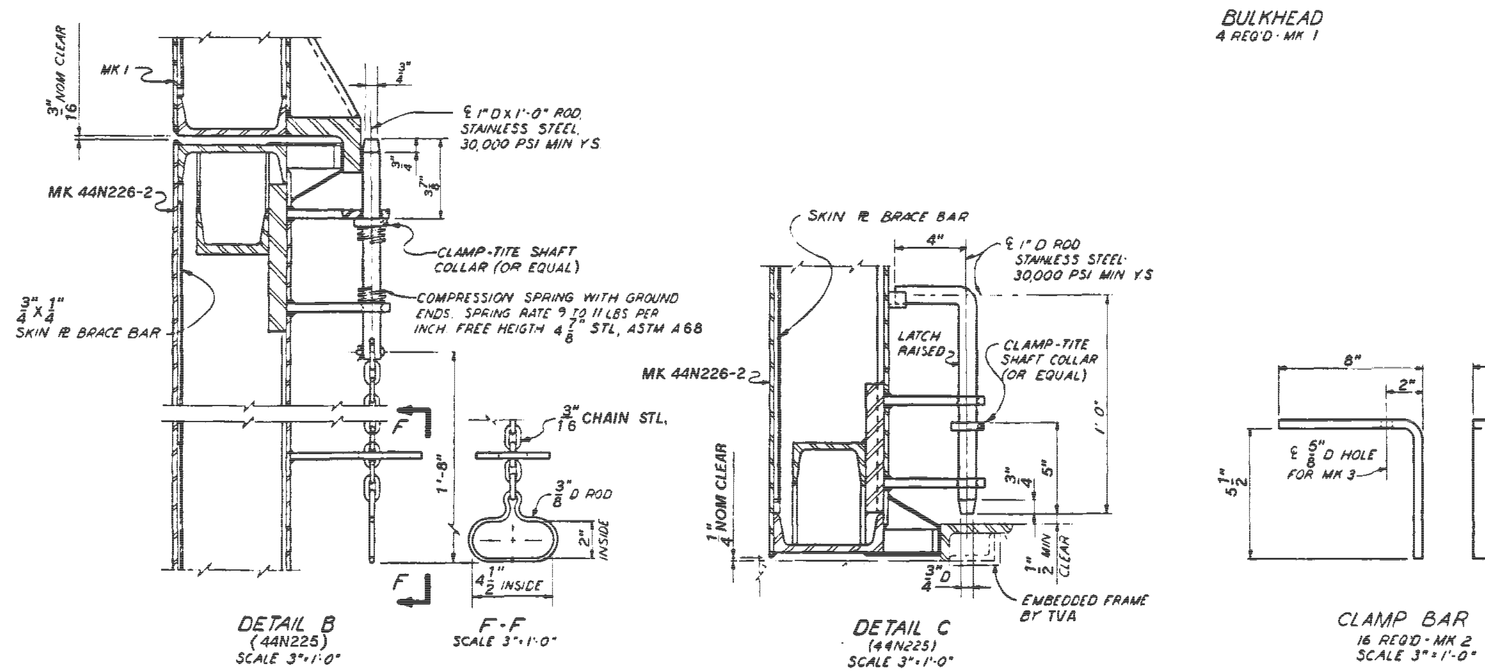
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

DIESEL GENERATOR BUILDING
UNITS 1 & 2
DOORS & BULKHEAD
DETAILS SHEET 1

TVA DWG NO. 44N226 RD
FIGURE 3.8.4-31

CAD MAINTAINED DRAWING





NOTES:
1. FOR GENERAL NOTES SEE 44N225.

PARTS:
ALL MARK NUMBERS TO HAVE PREFIX 44N227 UNLESS OTHERWISE NOTED.
MK 1 AND 2 DETAILED ON THIS DRAWING.
MK 3 - 1/2" D X 7-3/4" LONG CORROSION-RESISTANT HEX HEAD BOLT AND HEX NUT - 16 REQD.
MK 4 - 5/8" D TURNBUCKLE, JAW AND JAW ENDS, WITH 1/2" D PINS, C-TO-C PINS TO INCLUDE 24"-31" RANGE. MINIMUM BREAKING STRENGTH OF 15.0 K - 12 REQD.

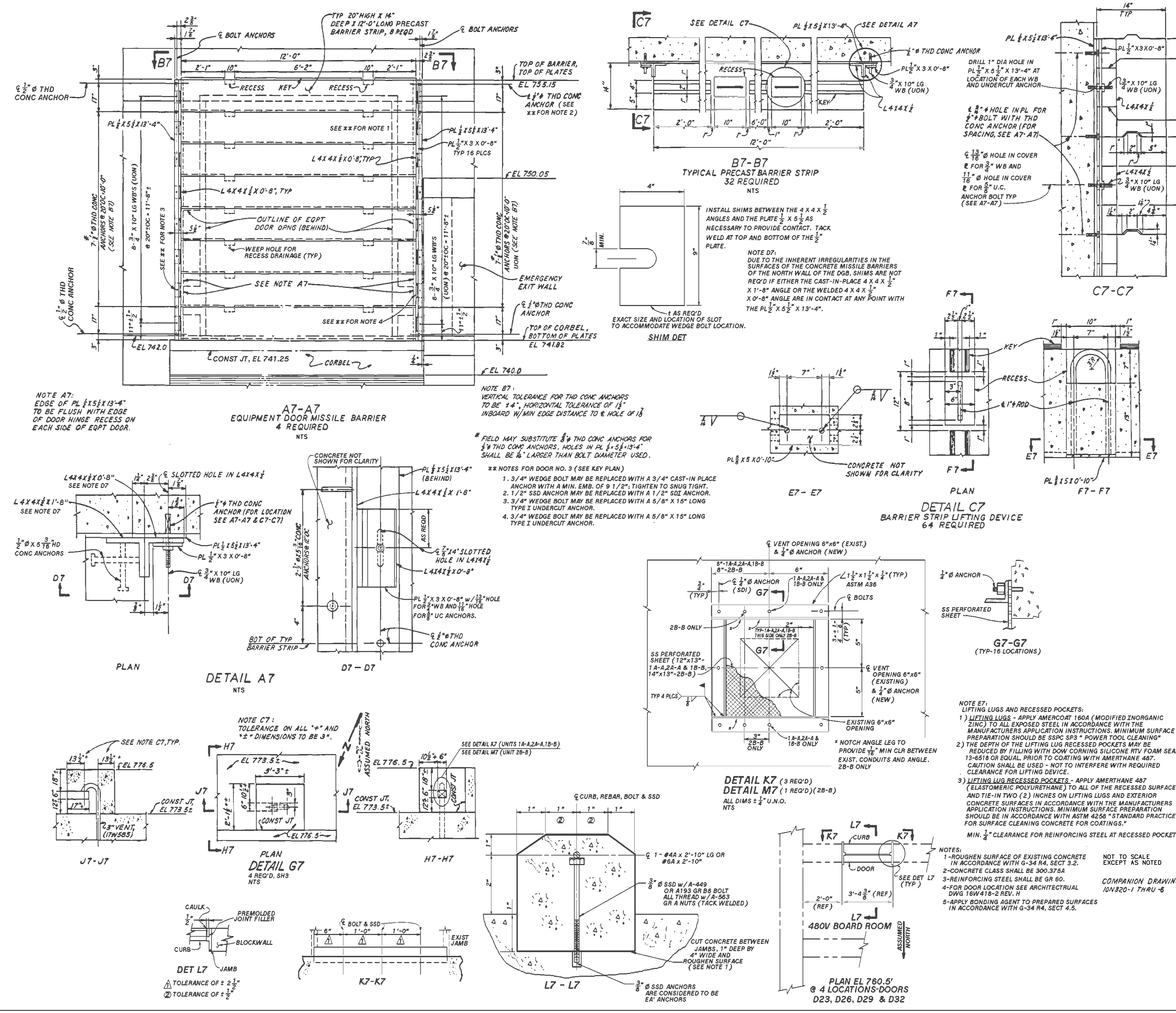
COMPANION DWGS: 44N225 & 44N226
SCALE $1\frac{1}{2}$"=1'-0" EXCEPT AS NOTED

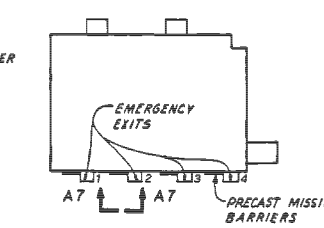
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

DIESEL GENERATOR BUILDING
UNITS 1 & 2
DOORS & BULKHEAD
DETAILS SHEET 2

TVA DWG NO. 44N227 RD
FIGURE 3.8.4-32

CAD MAINTAINED DRAWING





KEY PLAN
NTS

NOTES
1. FABRICATION AND INSTALLATION OF MATERIAL TO BE QA LEVEL II IN ACCORDANCE WITH N3G-881 EXCEPT HEADED CONCRETE ANCHORS AND STUD BOLTS.
2. ALL MATERIAL SHALL BE FABRICATED AND ERECTED IN ACCORDANCE WITH AISC CODE.
3. ALL MATERIAL AND FABRICATION BY TVA FIELD.
4. ALL PLATE MATERIAL TO BE ASTM A36.
5. ALL HEADED CONCRETE ANCHORS AND THREADED STUDS TO BE ASTM A108 AND REQUIRE ONLY CERTIFICATION THAT THE MATERIAL CONFORMS TO THE APPLICABLE ASTM SPECIFICATION.
6. ALL ANCHOR BOLTS AND NUTS TO BE ASTM A307.
7. ALL STUD BOLTS TO BE NELSON STUD ANCHORS OR EQUAL.
8. STUD BOLTS ARE NOT TO BE ATTACHED TO PLATES UNTIL BARRIER STRIP LOCATIONS ARE VERIFIED AND MATCH MARKED.
9. LIFTING DEVICE 1"øRODS TO BE ASTM A36.
10. CONCRETE BOLT ANCHORS TO BE PHILLIPS REDHEADS OR EQUAL.
11. BARRIER STRIP LIFTING EQUIPMENT TO BE PROVIDED BY HEAVY EQUIPMENT, AS LISTED:
(A) FORK LIFT TRUCK WITH 170" FORK HEIGHT.
(B) SPECIAL SHORT TANG FORKS.
(C) SPECIAL LIFT BEAM ATTACHMENT FOR USE WITH SHORT TANG FORKS.
12. ALL WELDING AND INSPECTION SHALL BE PERFORMED IN ACCORDANCE WITH TVA GENERAL CONSTRUCTION SPECIFICATION G-29C
13. ALL WELDING ELECTRODES TO BE AWS A5.1, E70 SERIES.
14. FOR ADDITIONAL NOTES SEE 10N320-1.
15. PROTECTIVE SCREENS ON THE CONCRETE OPENINGS FOR THE DIESEL OIL VENTS ON THE DGB ROOF TO BE PERFORATED SHEETS- 20 GAGE TYPE 304 SS WITH ROUND HOLES 0.25" WITH 58% EFFECTIVE OPEN AREA AS FURNISHED BY McMASTER-CARR OR EQUAL.
16. PROTECTIVE COATING ON THE CARBON STEEL SHAPES SHALL USE COATING SYSTEM WBNP-N-989 (AMERLOC 400 PRIMER AND AMERSHIELD AS TOP COAT). THEY SHALL BE APPLIED IN ACCORDANCE WITH ENGINEERING SPECIFICATION N3A-932.
17. ANCHORS SHALL BE 1/4" DIA. SDI INSTALLED IN ACCORDANCE WITH GENERAL ENGINEERING SPECIFICATION G-32.

NOT TO SCALE EXCEPT AS NOTED

COMPANION DRAWINGS:
10N320-1 THRU -6

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
DIESEL GENERATOR BUILDING
CONCRETE
FLOORS & WALLS
OUTLINE-SHEET 7
TVA DWG NO. 10N320-7 RF
FIGURE 3.8.4-33

FIGURE 3.8.4-34

DELETED

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-35

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-36

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-36

UNIT 2

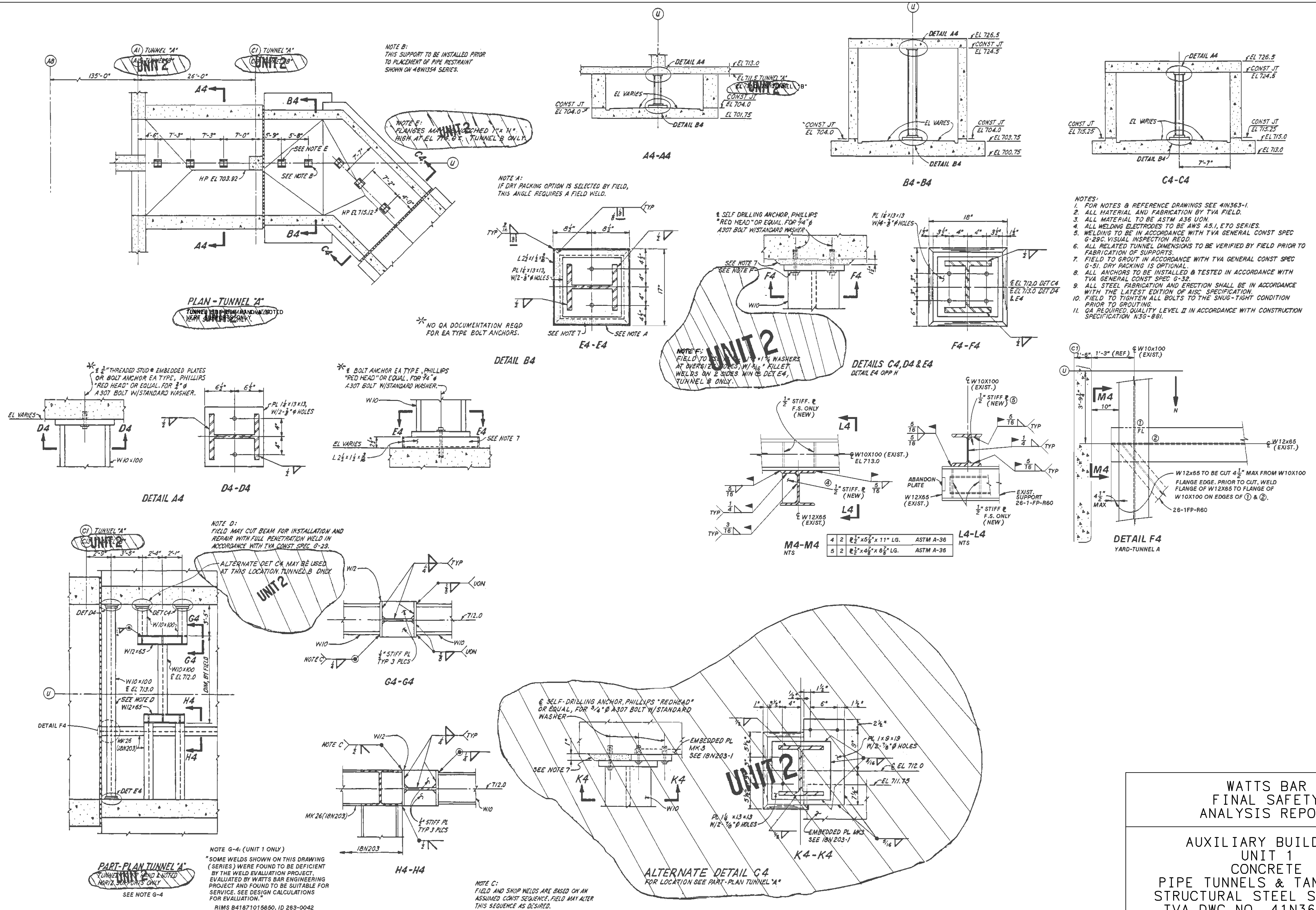
SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-36A

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-36A

UNIT 2

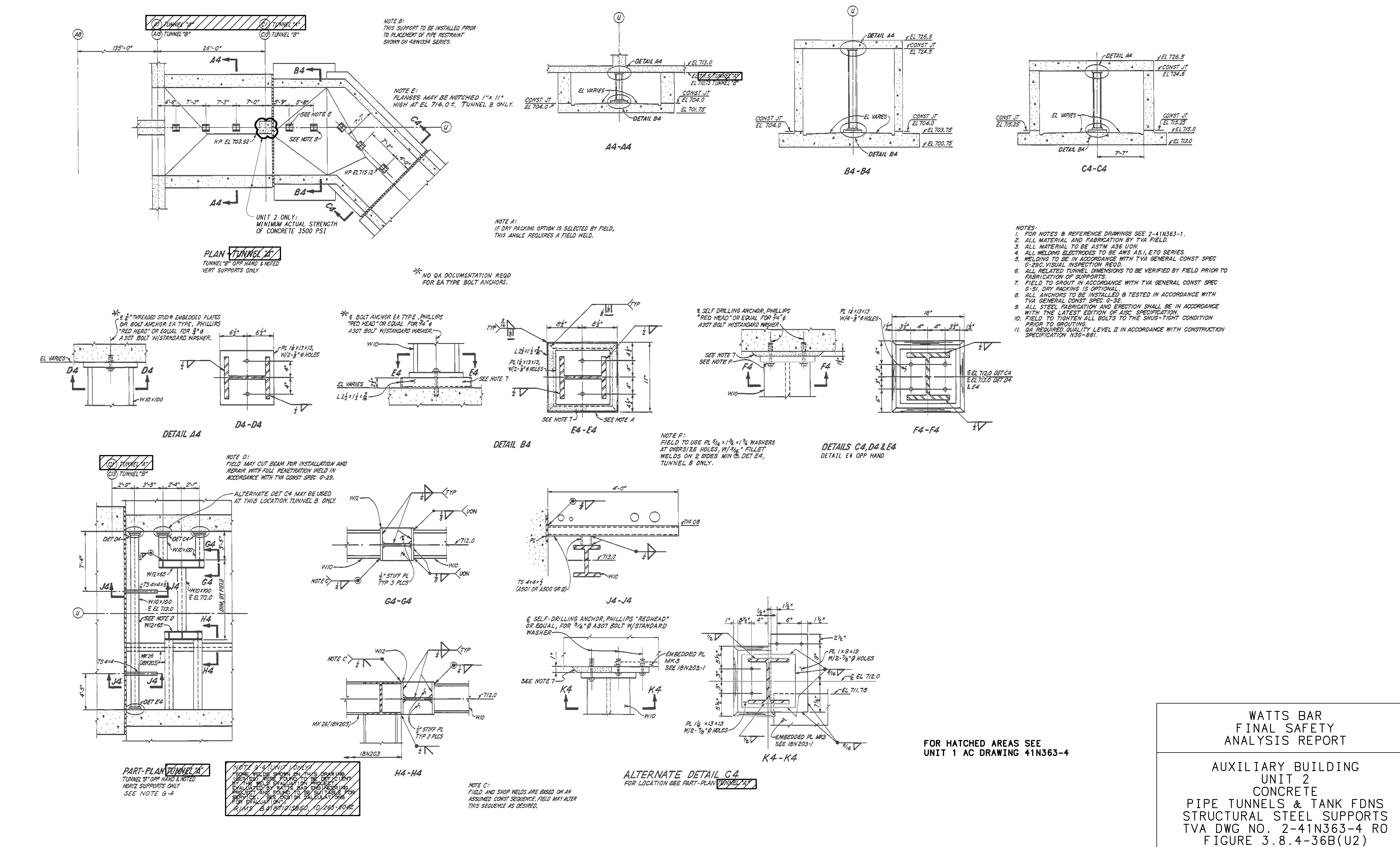


NOTES:
1. FOR NOTES & REFERENCE DRAWINGS SEE 41N363-1.
2. ALL MATERIAL AND FABRICATION BY TVA FIELD.
3. ALL MATERIAL TO BE ASTM A36 UON.
4. ALL WELDING ELECTRODES TO BE AWS A5.1, E70 SERIES.
5. WELDING TO BE IN ACCORDANCE WITH TVA GENERAL CONST SPEC G-29C. VISUAL INSPECTION REQD.
6. ALL RELATED TUNNEL DIMENSIONS TO BE VERIFIED BY FIELD PRIOR TO FABRICATION OF SUPPORTS.
7. FIELD TO GROUT IN ACCORDANCE WITH TVA GENERAL CONST SPEC G-51. DRY PACKING IS OPTIONAL.
8. ALL ANCHORS TO BE INSTALLED & TESTED IN ACCORDANCE WITH TVA GENERAL CONST SPEC G-32.
9. ALL STEEL FABRICATION AND ERECTION SHALL BE IN ACCORDANCE WITH THE LATEST EDITION OF AISC SPECIFICATION.
10. FIELD TO TIGHTEN ALL BOLTS TO THE SNUG-TIGHT CONDITION PRIOR TO GROUTING.
11. QA REQUIRED, QUALITY LEVEL II IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3G-881.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
CONCRETE
PIPE TUNNELS & TANK FDNS
STRUCTURAL STEEL SUPPORTS
TVA DWG NO. 41N363-4 RD
FIGURE 3.8.4-36B

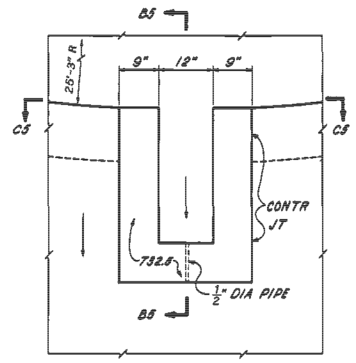
CAD MAINTAINED DRAWING



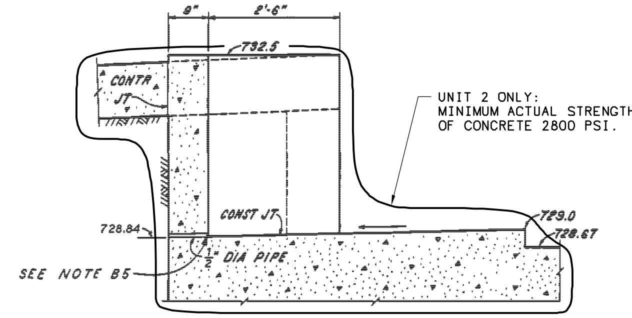
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
CONCRETE
PIPE TUNNELS & TANK FDNS
STRUCTURAL STEEL SUPPORTS
TVA DWG NO. 2-41N363-4 R0
FIGURE 3.8.4-36B(U2)

CAD MAINTAINED DRAWING

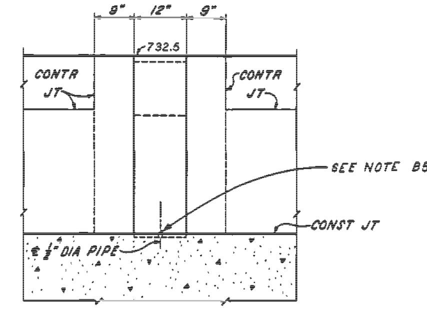


PLAN



B5-B5

DETAIL A5
4 REQD TANK FDN A
4 REQD TANK FDN B
FOR ORIENTATION SEE 41N363-3



C5-C5

NOTE B5
FIELD TO PLUG EXISTING 1/2" DIA PIPES WITH GROUT

NOTES:
FOR LIST OF NOTES AND REFERENCE DRAWINGS SEE 41N363-1.

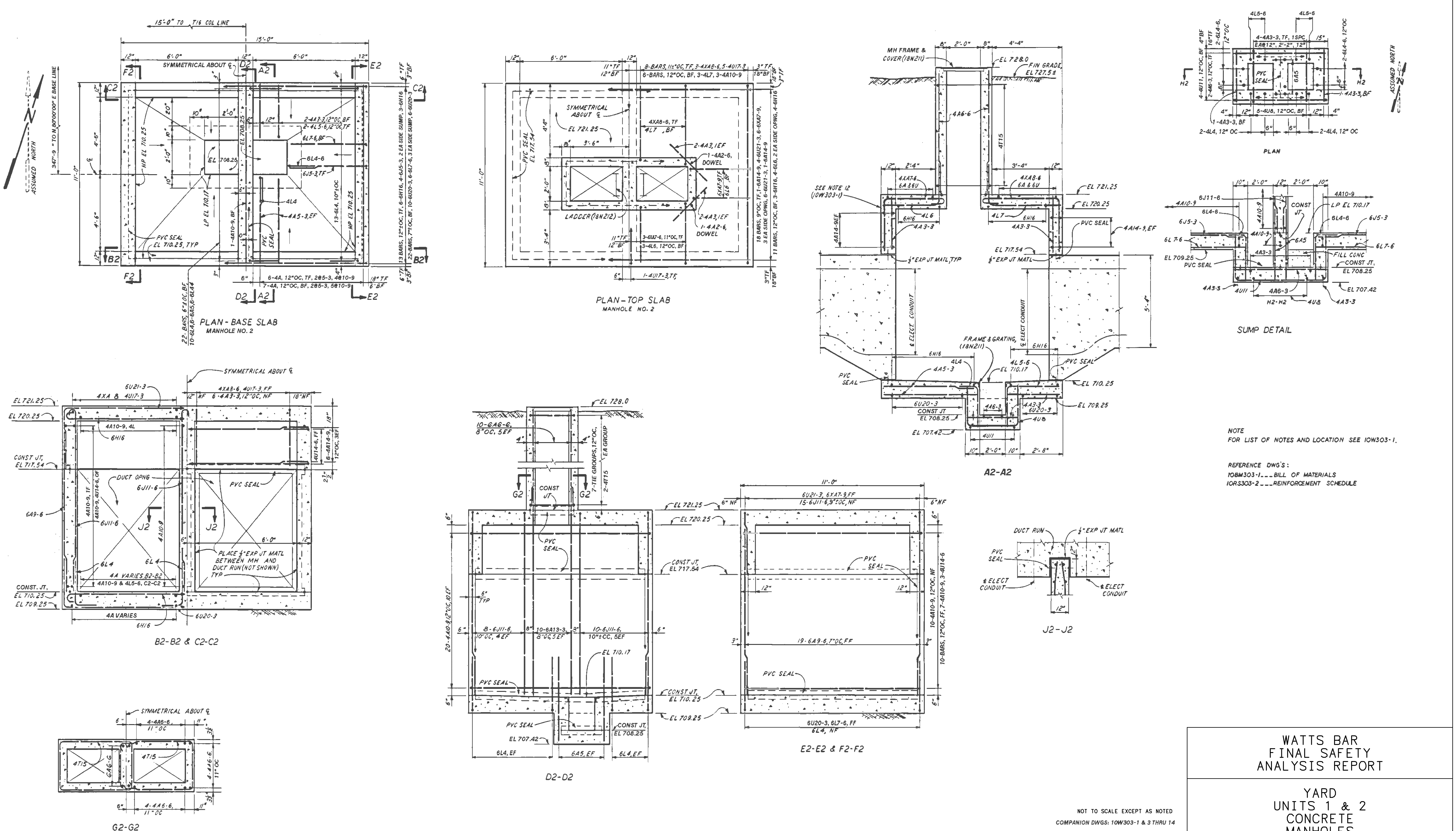
COMPANION DRAWINGS: 41N363-1 THRU -4

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
CONCRETE
PIPE TUNNELS & TANK FDNS
OUTLINE
TVA DWG NO. 41N363-5 RF
FIGURE 3.8.4-36C

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-37



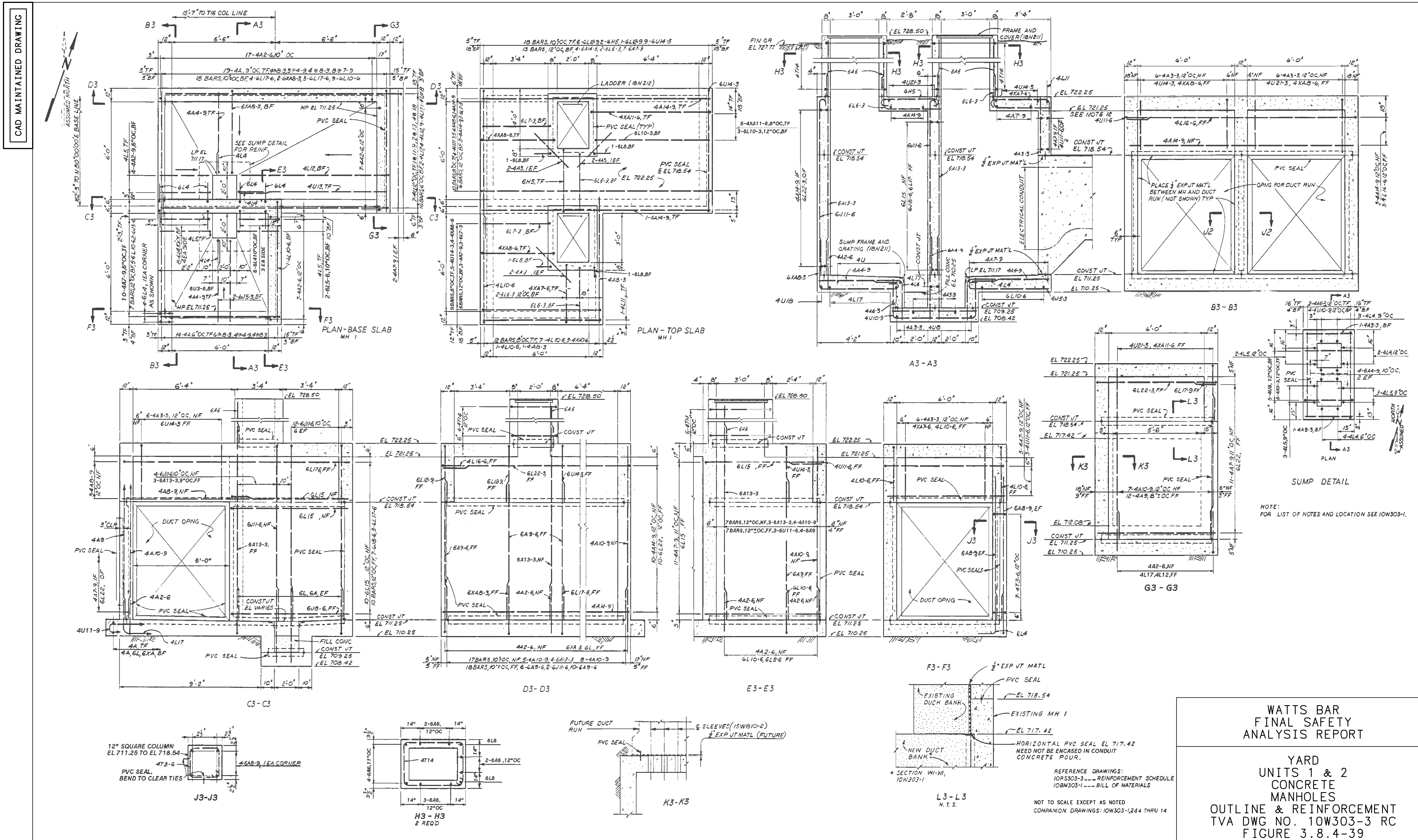
NOTE
FOR LIST OF NOTES AND LOCATION SEE 10W303-1.

REFERENCE DWG'S:
10BM303-1 --- BILL OF MATERIALS
10RS303-2 --- REINFORCEMENT SCHEDULE

NOT TO SCALE EXCEPT AS NOTED
COMPANION DWGS: 10W303-1 & 3 THRU 14

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

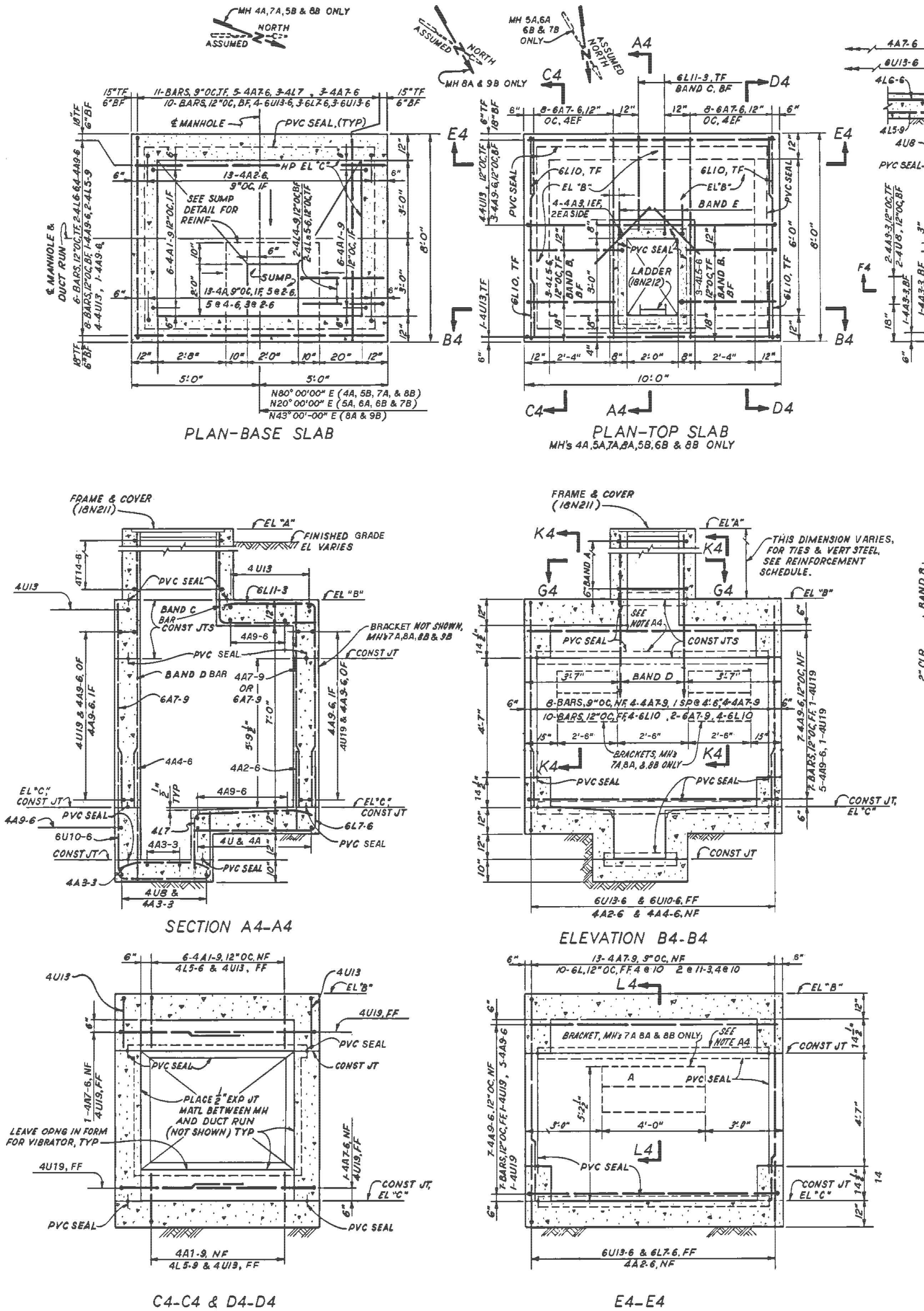
YARD
UNITS 1 & 2
CONCRETE
MANHOLES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-2 RC
FIGURE 3.8.4-38



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-3 RC
FIGURE 3.8.4-39

CAD MAINTAINED DRAWING



PLAN-BASE SLAB

PLAN-TOP SLAB
MH's 4A,5A,7A,8A,5B,6B & 8B ONLY

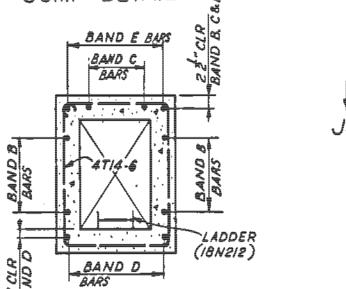
PLAN
SUMP DETAIL

SECTION A4-A4

ELEVATION B4-B4

C4-C4 & D4-D4

E4-E4



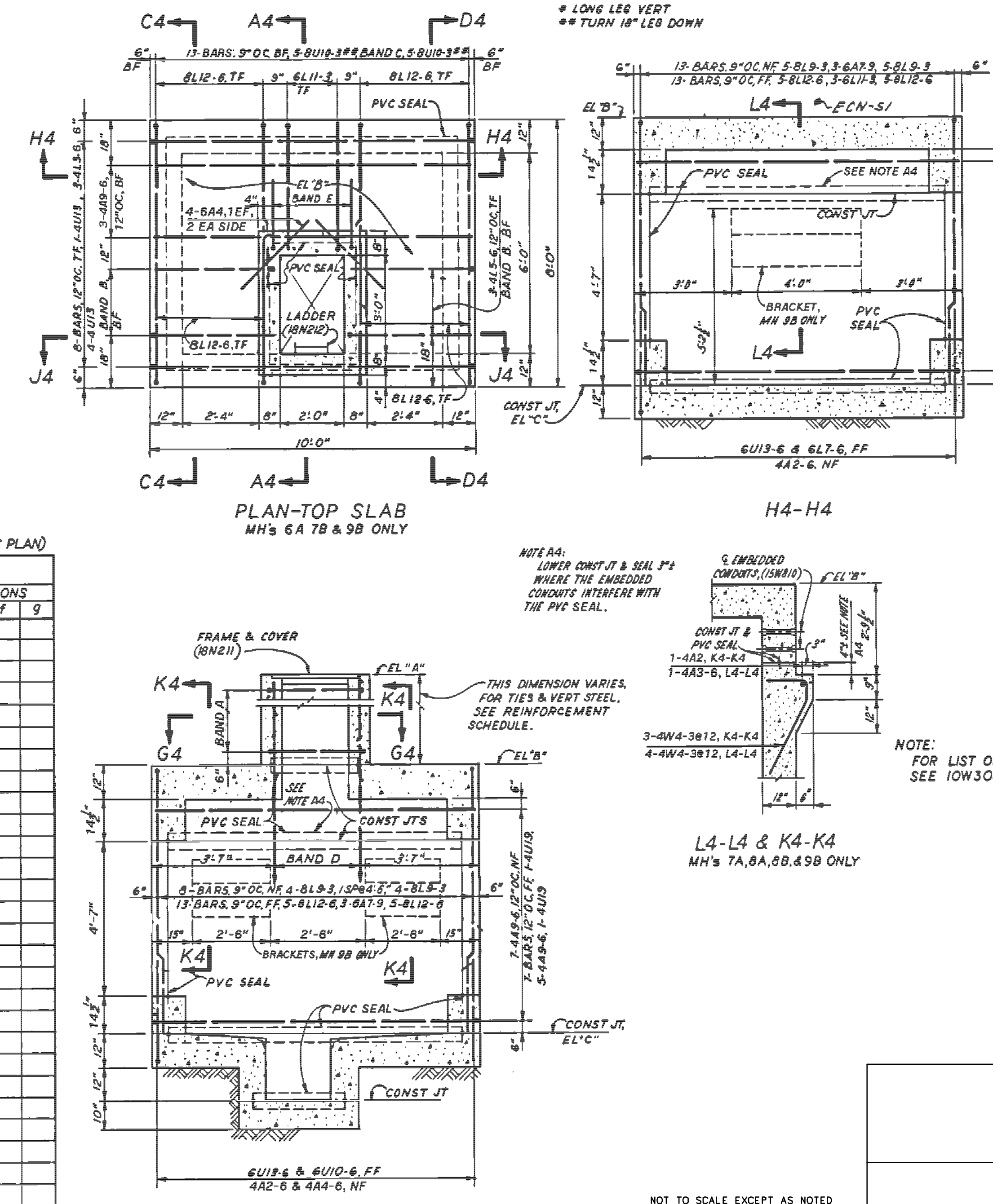
G4-G4 (ORIENTED SAME AS PLAN)

MANHOLE ELEVATIONS & REINFORCEMENT SCHEDULE								
MH#	EL"A"	EL"B"	EL"C"	BAND A	BAND B	BAND C	BAND D	BAND E
4A	728.25	725.17	717.17	3-4T14-6,12"OC	3-6La7-6,12"OC	3-6La7-6, 6"OC	5-6A10-9, 9"±OC	2-6L4-6,2'8"OC
5B	728.25	726.17	718.17	2-4T14-6,12"OC	3-6L6-6,12"OC	3-6L6-6,6"OC	5-6A9-9,9"±OC	2-6L3-6,2'8"OC
5A	728.25	725.25	717.25	3-4T14-6,12"OC	#3-6L7,12"OC	#3-6L7, 6"OC	5-6A10-9,9"±OC	2-6L4-6,2'8"OC
6A	724.25	713.25	705.25	11-4T14-6,12"OC	3-8L15,12"OC	3-8L15,9"OC	5-8A18-9,9"±OC	2-8L12-9,2'8"OC
6B	728.25	725.25	717.25	3-4T14-6,12"OC	#3-6L7,12"OC	#3-6L7, 6"OC	5-6A10-9,9"±OC	2-6L4-6,2'8"OC
7A	722.25	719.0	711.0	3-4T14-6,12"OC	#3-6L7-3,12"OC	#3-6L7-3,6"OC	5-6A11, 9"± OC	2-6L4-9,2'8"OC
7B	723.75	713.25	705.25	10-4T14-6,12"OC	3-8L14-3,12"OC	3-8L14-3,9"OC	5-8A18-3,9"±OC	2-8L12-3,2'8"OC
8A	714.25	706.25	698.25	8-4T14-6,12"OC	3-6L12,12"OC	3-6L12,6"OC	5-6A15-9,9"±OC	2-6L9-6,2'8"OC
8B	723.25	719.0	711.0	4-4T14-6,12"OC	3-6L8-3,12"OC	3-6L8-3, 6"OC	5-6A12,9"± OC	2-6L5-9,2'8"OC
9B	714.25	706.25	698.25	8-4T14-6,12"OC	3-8L11-9,12"OC	3-8L11-9,9"OC	5-8A15-9,9"±OC	2-8L10, 2'8"OC

LONG LEG VERT
TURN 18" LEG DOWN

BENT BAR LIST CONT							
BAR MARK	NO. REQD	BENDING DIMENSIONS					
		a	b	c	e	f	g
4W4-3	40	2-7	0-7	EX	1-2		

BENT BAR LIST							
BAR MARK	NO. REQD	BENDING DIMENSIONS					
		a	b	c	e	f	g
8L11-9	9	3-6	EX				
8L10	2	8-8	EX				
8L12-9	2	11-7	EX				
6L4-9	2	3-11	EX				
6U13-6	70	3-0	7-8	EX			
6L11-3	22	3-7	EX				
6L10	128	7-10	EX				
6U10-6	40	2-5	3-6	EX			
6L7-6	30	3-0	EX				
6La7-6	9	3-8	EX				
4U19	40	4-10	9-6	EX			
4T14-6	55	4-0	3-0				
4U13	90	1-9	9-8	EX			
6L5-9	2	1-1	EX				
4U8	20	2-5	3-4	EX			
4L7	30	2-6	EX				
4L5-9	20	1-9	EX				
4L5-6	60	1-10	EX				
4L4-9	20	1-9	EX				
6L6-6	9	3-8	EX				
6L4-6	6	3-9	EX				
6L3-6	2	2-9	EX				
6L8-3	9	3-6	EX				
4L6-6	20	2-6	EX				
4La5-6	20	2-6	EX				
6L7-3	9	3-6	EX				
8L15	9	3-6	EX				
6L9-6	2	1-1	EX				
8L12-3	2	11-2	EX				
6L12	9	3-6	EX				
8L14-3	9	3-6	EX				
8L9-3	36	7-10	EX				
8U10-3	20	1-6	7-8	EX			
8L12-6	40	7-10	EX				
6L7	18	3-6	EX				



PLAN-TOP SLAB
MH's 6A 7B & 9B ONLY

H4-H4

J4-J4

L4-L4 & K4-K4
MH's 7A,8A,8B,& 9B ONLY

NOTE A4:
LOWER CONST JT & SEAL 3"± WHERE THE EMBEDDED CONDUITS INTERFERE WITH THE PVC SEAL.

NOTE:
FOR LIST OF NOTES, LOCATION AND ORIENTATION, SEE 10W303-1.

NOT TO SCALE EXCEPT AS NOTED

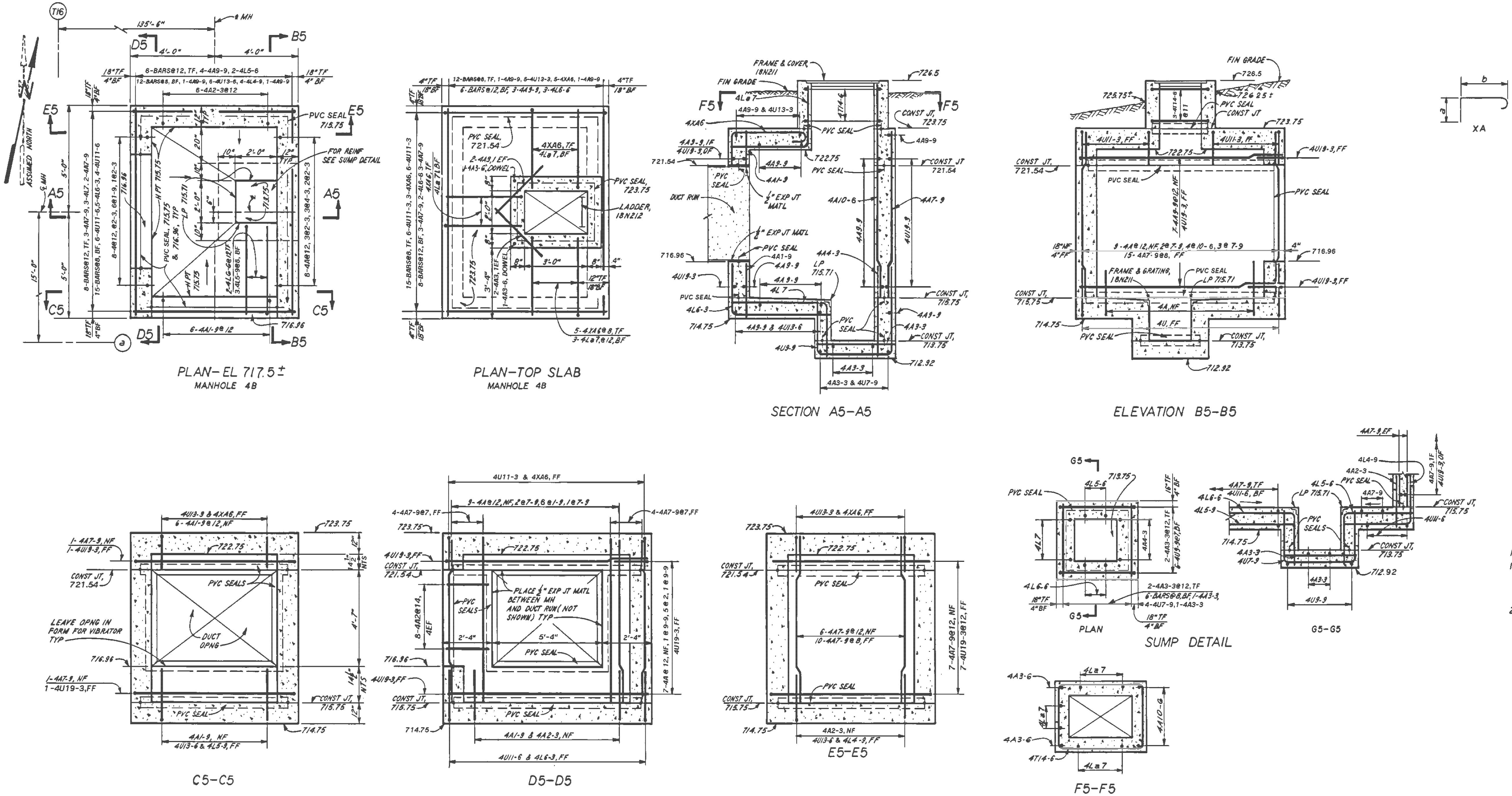
COMPANION DRAWINGS:
10W303-1,2,3 & 5THRU 11

REFERENCE DRAWINGS:
10BM303-1 ...... BILL OF MATERIAL

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLES 4A-8A & 5B-9B
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-4 RD
FIGURE 3.8.4-40

CAD MAINTAINED DRAWING



BENT BAR LIST							
BAR MARK	NO. REQD	BENDING DIMENSIONS					
		a	b	c	e	f	g
4L4-9	4	1-9	EX				
4L5-9	3	1-9	EX				
4XA6	13	1-10	3-8				
4L5-6	2	2-6	EX				
4L6-3	5	1-9	EX				
4L6-6	2	2-6	EX				
4La7	8	3-8	EX				
4L7	3	2-6	EX				
4U7-9	4	2-3	3-4	EX			
4U9-9	6	2-5	3-6	EX			
4U11-3	12	1-10	7-8	EX			
4U11-6	10	2-3	7-8	EX			
4U13-3	5	1-10	9-8	EX			
4U13-6	6	1-10	9-8	EX			
4T14-6	3	4-0	3-0				
4U19-3	9	9-8	7-7	EX			

NOTES:

1. THE FIRST AND LAST BAR IN A REINFORCING BILLING, UNLESS OTHERWISE NOTED, ARE POSITIONED ONE HALF OF THE DESIGNATED BAR SPACING OR 6 INCHES, WHICHEVER IS LESS, FROM THE NEAREST OUTLINE FEATURE DEFINING THE AREA BEING REINFORCED.
2. FOR ADDITIONAL NOTES AND LOCATION, SEE 10W303-1.

NOT TO SCALE EXCEPT AS NOTED

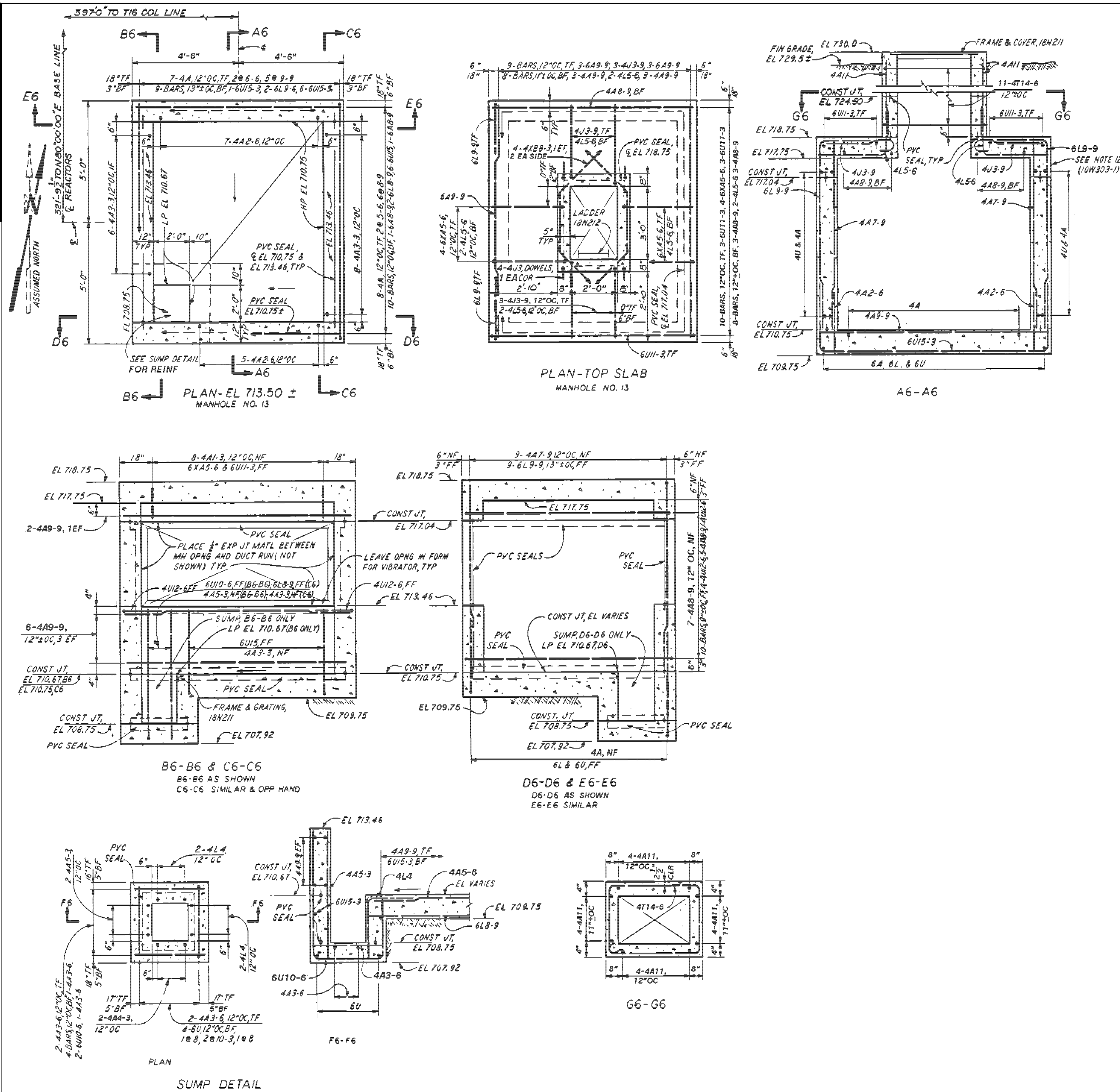
COMPANION DRAWINGS:
10W303-1 THRU 4 & 6 THRU Later

REFERENCE DRAWINGS:
10BM303-1___BILL OF MATERIAL

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLE 4B
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-5 RD
FIGURE 3.8.4-41

CAD MAINTAINED DRAWING



NOTE:
FOR LIST OF NOTES AND LOCATION OF MANHOLES, SEE 10W303-1.

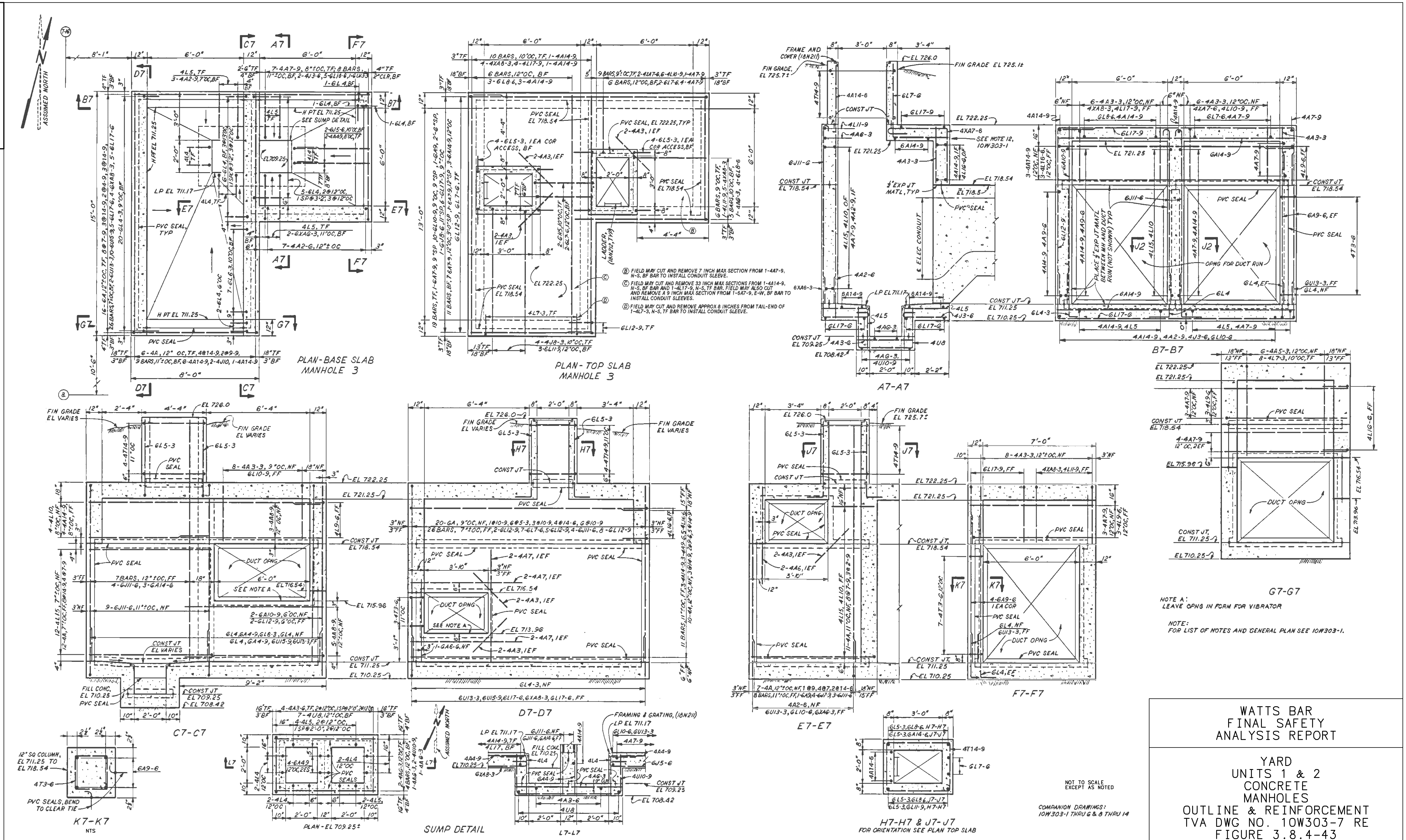
REFERENCE DRAWINGS:
10BM303-1___BILL OF MATERIALS
10RS303-6___REINFORCEMENT SCHEDULE

NOT TO SCALE
EXCEPT AS NOTED

COMPANION DWGS: 10W303-1 THRU 5, 7 THRU 14

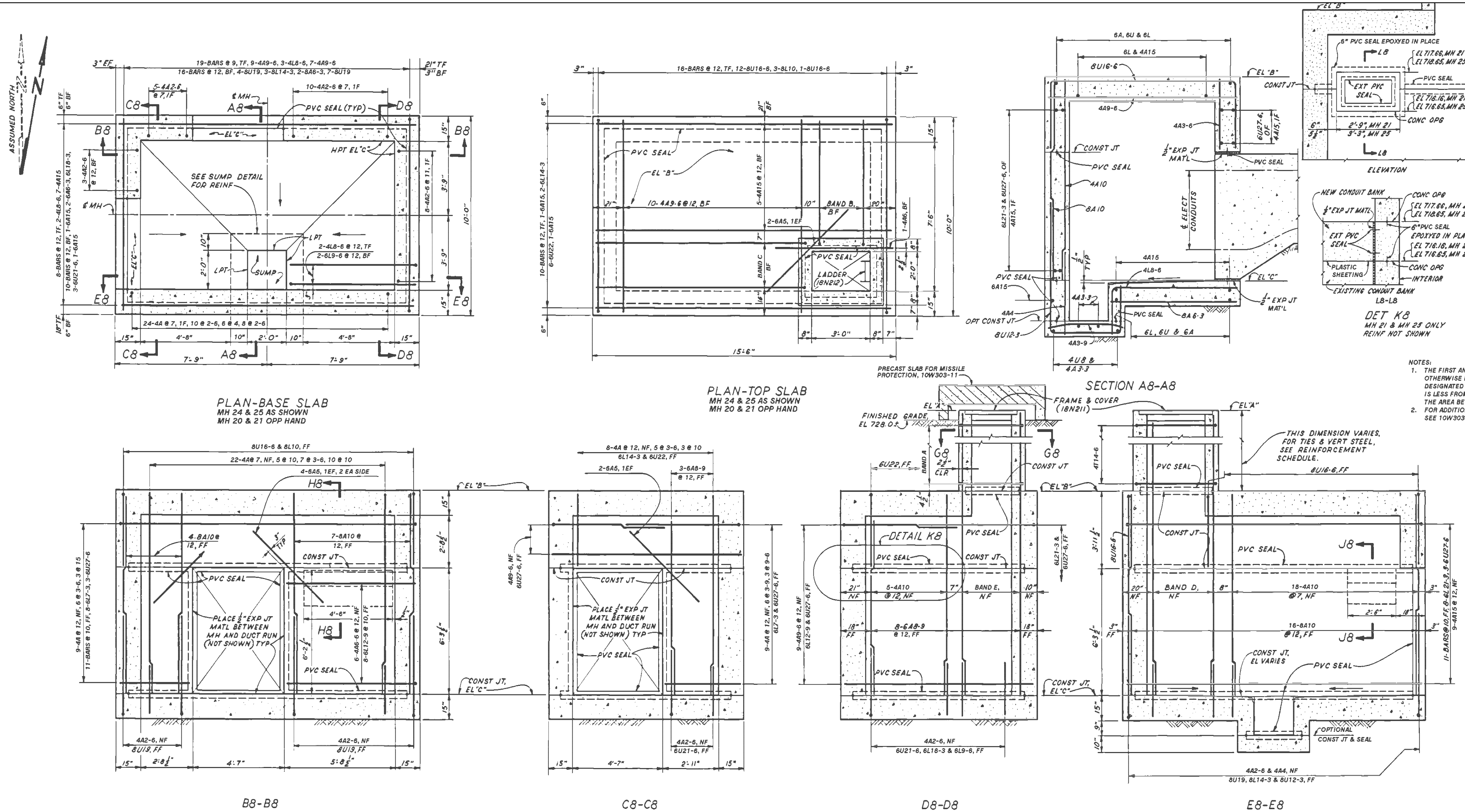
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-6 RD
FIGURE 3.8.4-42



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

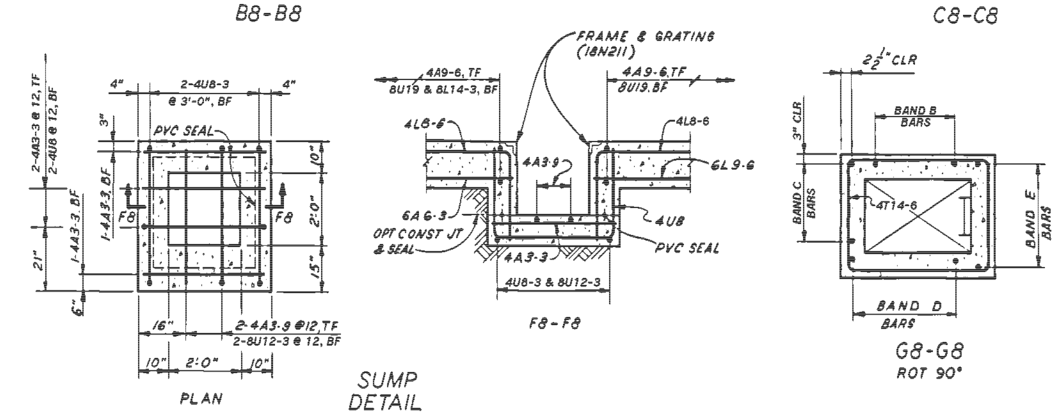
YARD
UNITS 1 & 2
CONCRETE
MANHOLES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-7 RE
FIGURE 3.8.4-43



BENT BAR LIST

BAR MARK	NO. REQD	BENDING DIMENSIONS a	b	c	e	f	g
8U19	44	4-11	9-8	EX			
8U16-6	52	3-7	9-8	EX			
8L14-3	12	9-8	EX				
8U12-3	8	6-6	3-9	EX			
8L10	12	6-5	EX				
6U27-6	24	6-5	15-0	EX			
6U22	24	3-6	15-2	EX			
6U21-6	12	3-4	15-2	EX			
6L21-3	32	15-1	EX				
6L19-9	10	10-8	EX				
6L18-3	12	15-1	EX				
6L17-9	10	10-8	EX				
6L15-3	8	6-2	EX				
6L14-3	8	10-10	EX				
6L13-3	8	6-2	EX				
6L12-9	32	6-6	EX				
6L9-6	8	6-5	EX				
6L7-3	32	3-7	EX				
4T14-6	32	4-0	3-0				
4L8-6	28	6-3	EX				
4U8-3	8	2-4	3-9	EX			
4U8	8	2-5	3-4	EX			
4W5	32	3-2	0-6	EX		2-10	

NOTES:
1. THE FIRST AND LAST BAR IN A REINFORCING BILLING, UNLESS OTHERWISE NOTED, ARE POSITIONED ONE HALF OF THE DESIGNATED BAR SPACING OR 6 INCHES, WHICHEVER IS LESS FROM THE NEAREST OUTLINE FEATURE DEFINING THE AREA BEING REINFORCED.
2. FOR ADDITIONAL NOTES, LOCATIONS AND ORIENTATION, SEE 10W303-1.



MANHOLE ELEVATIONS & REINFORCEMENT SCHEDULE

MH	EL"A"	EL"B"	EL"C"	BAND A	BAND B	BAND C	BAND D	BAND E
20	728.5	719.75	709.5	9-4T14-6@12	4-6L15-3@9	5-6L19-9@7	5-6A18-9@9	6-6A18-9@7
21	728.5	719.75	709.5	9-4T14-6@12	4-6L15-3@9	5-6L19-9@7	5-6A18-9@9	6-6A18-9@7
24	728.5	721.75	711.5	7-4T14-6@12	4-6L13-3@9	5-6L17-9@7	5-6A16-9@9	6-6A16-9@7
25	728.5	721.75	711.5	7-4T14-6@12	4-6L13-3@9	5-6L17-9@7	5-6A16-9@9	6-6A16-9@7

NOT TO SCALE EXCEPT AS NOTED

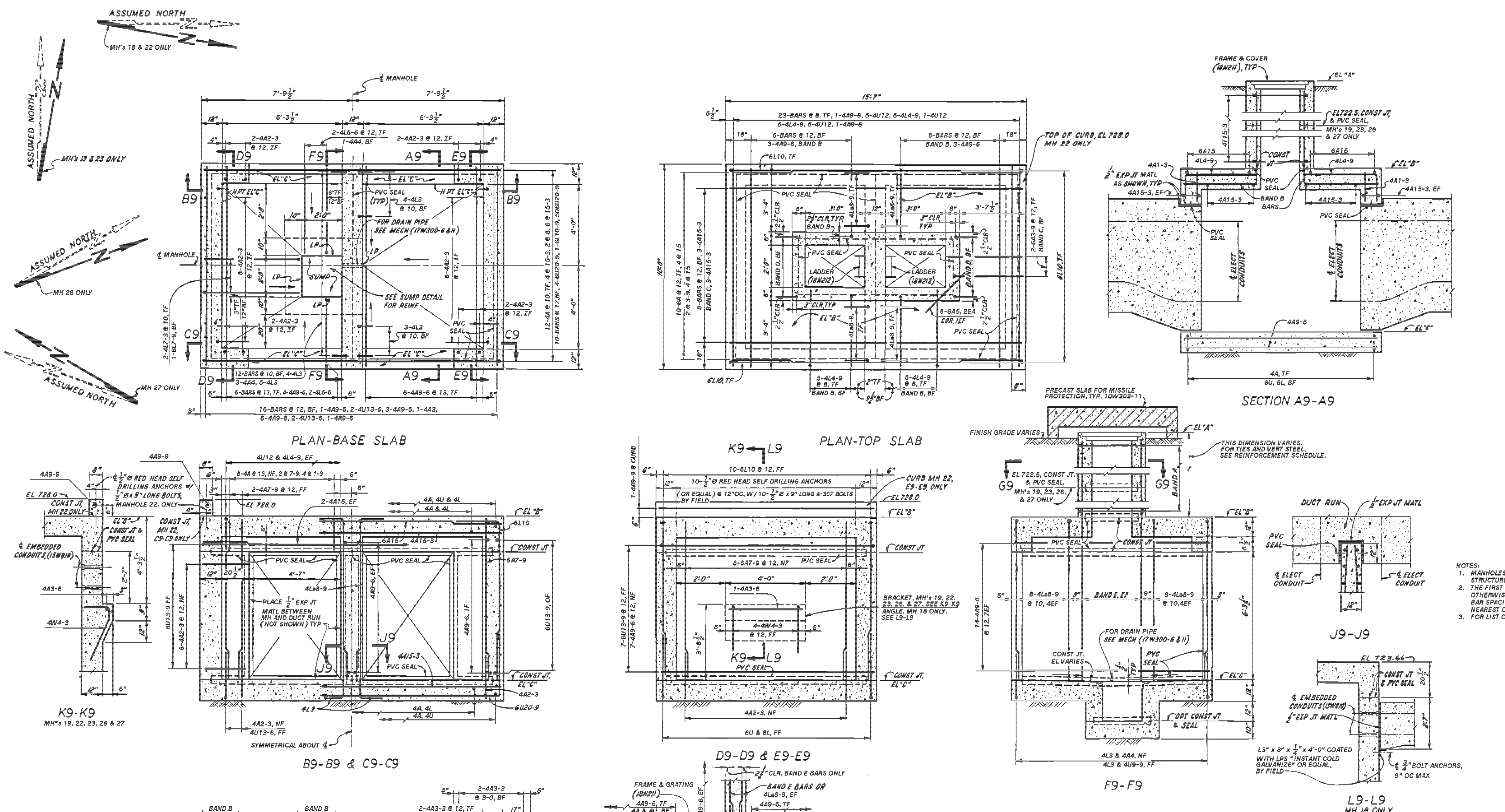
COMPANION DRAWINGS:
10W303-1 THRU 7 & 9 THRU 11

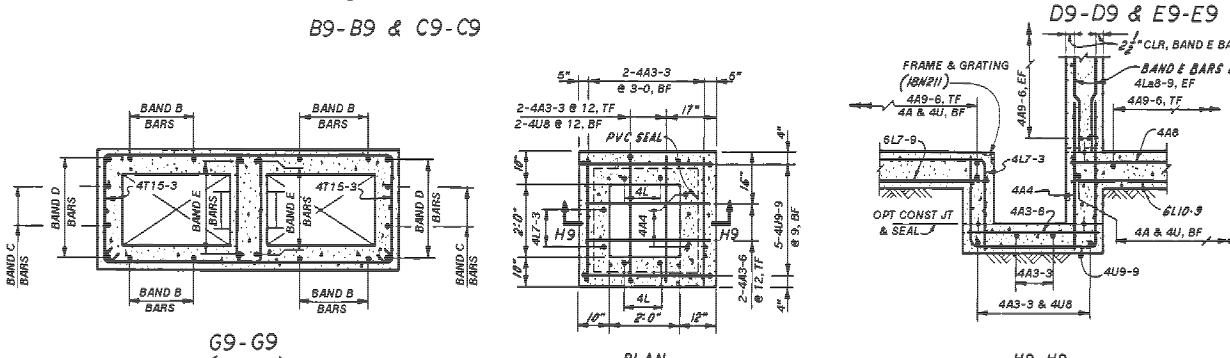
REFERENCE DRAWINGS:
10BM303-1.........BILL OF MATERIAL

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLES 20, 21, 24 & 25
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-8 RD
FIGURE 3.8.4-44

CAD MAINTAINED DRAWING





BENT BAR LIST

BAR MARK	NO. REQD	BENDING DIMENSIONS a	b	c	e	f	g
6U20-9	54	2-11	15-3	EX			
6U13-9	84	2-3	9-8	EX			
6L12-9	4	1-3	EX				
6L10-9	6	7-11	EX				
6L10	120	7-10	EX				
6L11-6	4	10-7	EX				
6L7-9	6	4-11	EX				
6L6-3	4	5-3	EX				
6L2-9	4	1-9	EX				
4T15-3	88	3-0	4-4				
4L15-3	4	3-8	EX				
4La15	12	3-4	EX				
4U13-6	24	2-1	9-6	EX			
4U12	66	1-3	9-7	EX			
4L14-3	4	3-8	EX				
4La13-9	12	3-4	EX				
4U9-9	30	2-5	3-6	EX			
4L8-9	4	3-8	EX				
4La8-9	96	7-10	EX				
4L8-6	12	3-4	EX				
4U8	12	2-5	3-4	EX			
4L7-3	12	4-11	EX				
4L6-6	12	2-5	EX				
4L5-6	12	2-5	EX				
4L5-3	4	3-8	EX				
4L5	12	3-4	EX				
4L4-9	120	1-3	EX				
4L3	96	1-0	EX				
6L12-3	4	1-2	EX				
6L11-3	4	1-3	EX				
4L15	4	3-8	EX				
4L14-6	12	3-4	EX				
4L13-9	4	3-8	EX				
4L13-6	12	3-4	EX				
4W4-3	48	2-7	0-7	EX	1-2		

NOTES:
1. MANHOLES 18, 19, 22, 23, 26 & 27 ARE SEISMIC CATEGORY I STRUCTURES AND QUALITY ASSURANCE IS REQUIRED.
2. THE FIRST AND LAST BAR IN A REINFORCING BILLING, UNLESS OTHERWISE NOTED, IS POSITIONED ONE HALF OF THE DESIGNATED BAR SPACING OR 6 INCHES, WHICHEVER IS LESS, FROM THE NEAREST OUTLINE FEATURE DEFINING THE AREA BEING REINFORCED.
3. FOR LIST OF ADDITIONAL NOTES AND LOCATION SEE 10W303-1.

MANHOLE ELEVATIONS & REINFORCEMENT SCHEDULE

MH	EL "A"	EL "B"	EL "C"	BAND A	BAND B	BAND C	BAND D	BAND E
18	728.25	723.66	715.66	10-4T15-3@12,5 GROUPS OF 2	3-4L8-6 @ 12	2-4L8-9 @ 12	2-6L6-3 @ 2-10	5-6A12-3 @ 8
19	729.58	719.5	711.5	20-4T15-3@12,10 GROUPS OF 2	3-4La13-9 @ 12	2-4L14-3 @ 12	2-6L11-6 @ 2-10	5-11A17-9 @ 8
22	728.25	727.16	719.16	2-4T15-3@12,1 GROUP OF 2	3-4L5 @ 12	2-4L5-3 @ 12	2-6L2-9 @ 2-10	5-4A8-9 @ 8
23	729.58	718.5	710.5	22-4T15-3@12,11 GROUPS OF 2	3-4La15 @ 12	2-4L15-3 @ 12	2-6L12-9 @ 2-10	5-11A18-9 @ 8
26	728.08	718.5	710.5	20-4T15-3@12,10 GROUPS OF 2	3-4L13-6 @ 12	2-4L13-9 @ 12	2-6L11-3 @ 2-10	5-11A17-3@ 8
27	728.25	717.5	709.5	22-4T15-3@12,11 GROUPS OF 2	3-4L14-6 @ 12	2-4L15 @ 12	2-6L12-3 @ 2-10	5-11A18-6 @ 8

NOT TO SCALE EXCEPT AS NOTED

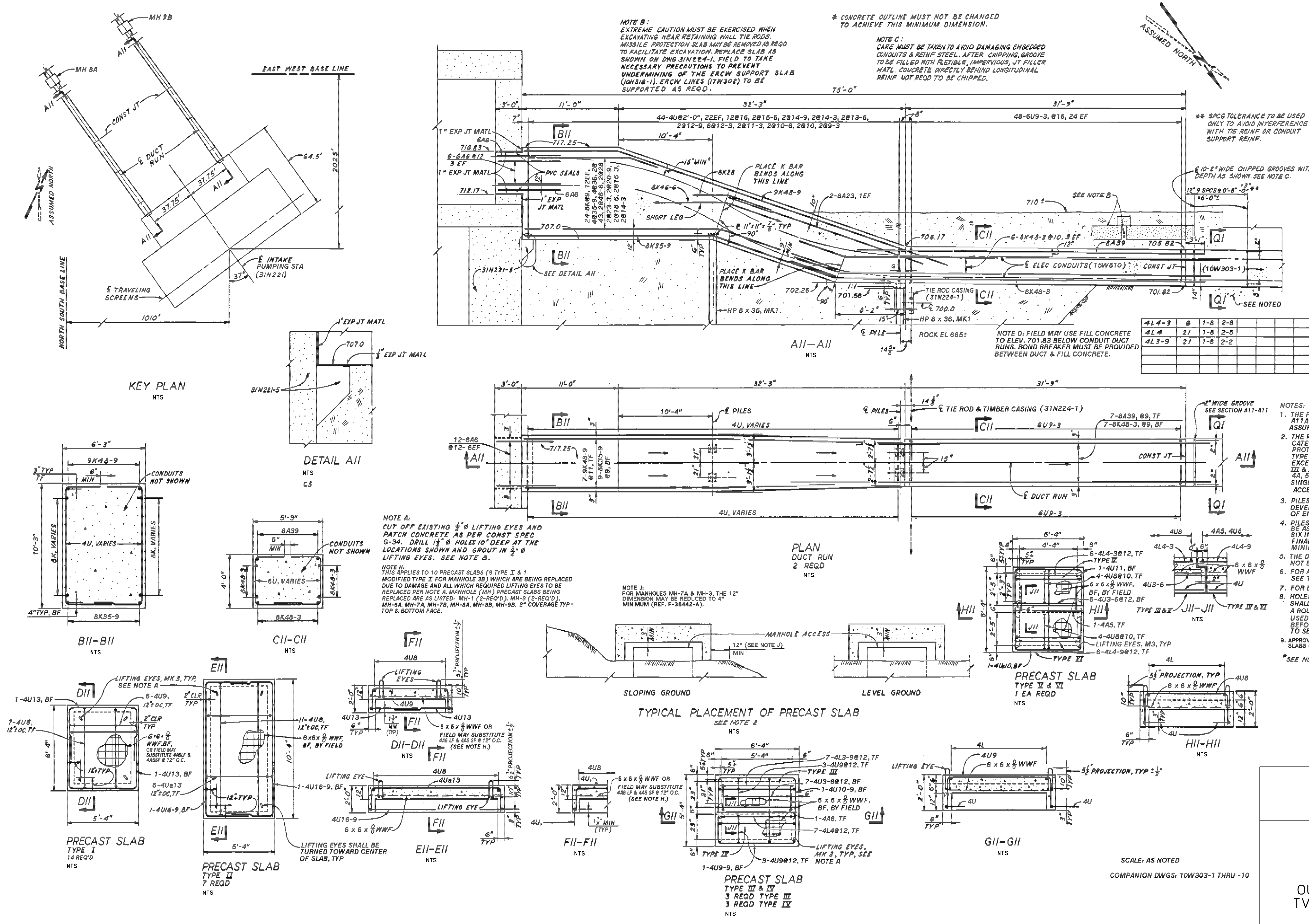
COMPANION DRAWINGS:
10W303-1 THRU 8 & 10, 11

REFERENCE DRAWINGS:
10BM303-1_____BILL OF MATERIAL

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLES 18, 19, 22, 23, 26 & 27
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-9 RD
FIGURE 3.8.4-45

CAD MAINTAINED DRAWING



4L4-3	6	1-8	2-8	
4L4	21	1-8	2-5	
4L3-9	21	1-8	2-2	

BENT BAR LIST

BAR MARK	NO. REQD	BENDING DIMENSIONS a	b	c	e	f	g
8K48-3	26	5-6	EX		1-9		
9K48-9	14	10-9	EX		3-6		
8K46-6	4	23-11	EX		7-10		
4U10-9	3	2-6	5-11	2-6			
8K43	4	21-6	EX		7-1		
8K36	8	14-6	EX		4-9		
8K35-9	26	14-7	EX		4-9		
8K28	4	5-1	EX		1-10		
4U13	30	4-1	5-0	4-1			
8K23-3	4	2-0	EX		0-8		
8K20-9	4	2-0	EX		0-8		
8K18-6	4	1-11	EX		0-8		
8K16-3	4	1-11	EX		0-8		
4Ua10	1	2-7	5-0	2-7			
8K14-3	4	2-0	EX		0-8		
4U11	1	3-1	5-0	3-1			
4U16-9	14	3-6	10-0	3-5			
6U9-3	96	2-2	3-7	EX			
4Ua13	42	1-8	9-11	1-7			
4U16	24	2-8	9-10	EX			
4U15-6	4	2-8	9-3	EX			
4U14-9	4	2-8	8-7	EX			
4U14-3	4	2-8	7-11	EX			
4U13-6	4	2-8	7-3	EX			
4U12-9	4	2-8	6-6	EX			
4U9-9	3	2-0	5-11	2-0			
4U12-3	12	2-6	6-4	EX			
4U12	16	2-4	6-4	EX			
4U11-3	4	2-2	5-10	EX			
4U10-6	4	2-2	5-1	EX			
4U10	4	2-2	4-6	EX			
4U9-3	4	2-2	3-10	EX			
4U3-6	27	1-9	0-3	EX	(USE 4d PIN)		
4U9	108	1-8	5-11	1-7			
4U8	190	1-8	4-11	1-7			
4L4-9	6	1-8	3-2				

NOTES:

1. THE PRECAST SLABS AND THE DUCT RUN OF SECTION A11 ARE CATEGORY I STRUCTURES AND QUALITY ASSURANCE IS REQUIRED (QUALITY LEVEL II PER N3G-881).
2. THE PRECAST SLABS SHALL BE PLACED OVER THE CATEGORY I MANHOLE# ACCESSES TO PROVIDE PROTECTION FROM TORNADO GENERATED MISSILES. TYPE I SLABS ARE FOR ALL SINGLE ACCESSES EXCEPT THOSE @ MH'S 4A, 5A, 5B, & 6B. TYPES III & IV ARE LOCATED ON THE SINGLE ACCESSES OF MH'S 4A, 5A & 5B WHILE TYPE V & VI ARE FOR MH 6B'S SINGLE ACCESS. TYPE II SLABS ARE FOR DOUBLE ACCESSES (MH'S 2, 18 ETC).
3. PILES SHALL BE DRIVEN BY AN APPROVED HAMMER DEVELOPING NOT LESS THAN 15,000 FOOT-POUNDS OF ENERGY/BLOW.
4. PILES SHALL BE DRIVEN TO REFUSAL. PILES SHALL BE ASSUMED DRIVEN TO REFUSAL WHEN THE LAST SIX INCHES AVERAGE TWENTY BLOWS/INCH. THE FINAL INCH OF PENETRATION SHALL REQUIRE A MINIMUM OF TWENTY BLOWS.
5. THE DEVIATION TOLERANCE IN PILE LOCATION SHALL NOT EXCEED THREE INCHES.
6. FOR ADDITIONAL NOTES AND REFERENCE DRAWINGS SEE 10W303-1.
7. FOR LIFTING OF PRECAST COVER SEE 14W211.
8. HOLES FOR BARS THAT ARE TO BE GROUTED IN PLACE SHALL BE MADE BY PERCUSSION DRILLING TO PROVIDE A ROUGH SURFACE CORE DRILLING SHALL NOT BE USED. HOLES SHALL BE CLEAN AND FREE FROM WATER BEFORE FILLING WITH GROUT. GROUT SHALL CONFORM TO SECTION 2.4 OR 2.5 OF CONST G-32.
9. APPROVED SITE CRUSHED STONE MAY BE USED UNDER MISSILE SLABS (REF. F-38442-A).

#SEE NOTE 4, 10W303-1

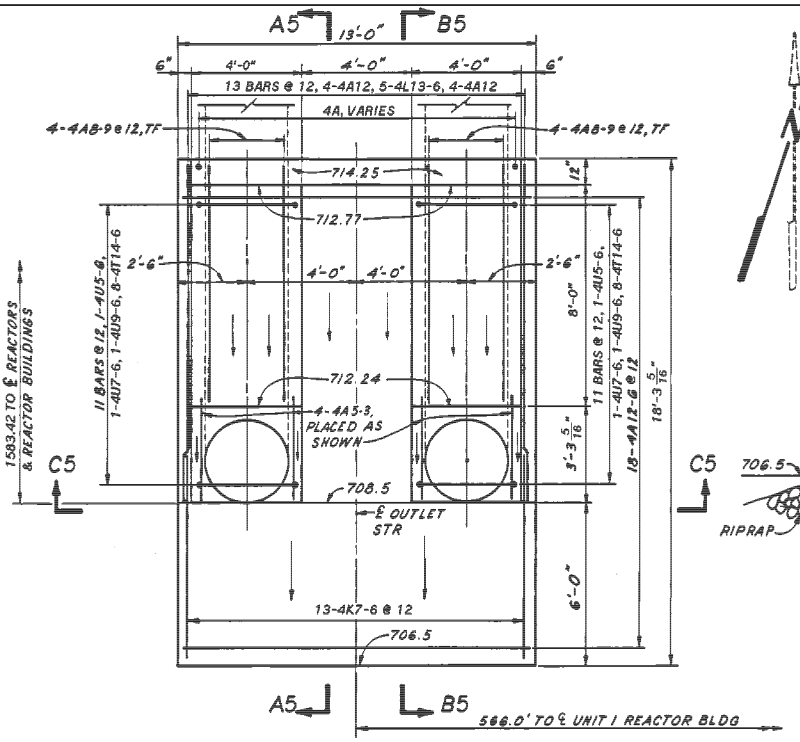
SCALE: AS NOTED

COMPANION DWGS: 10W303-1 THRU -10

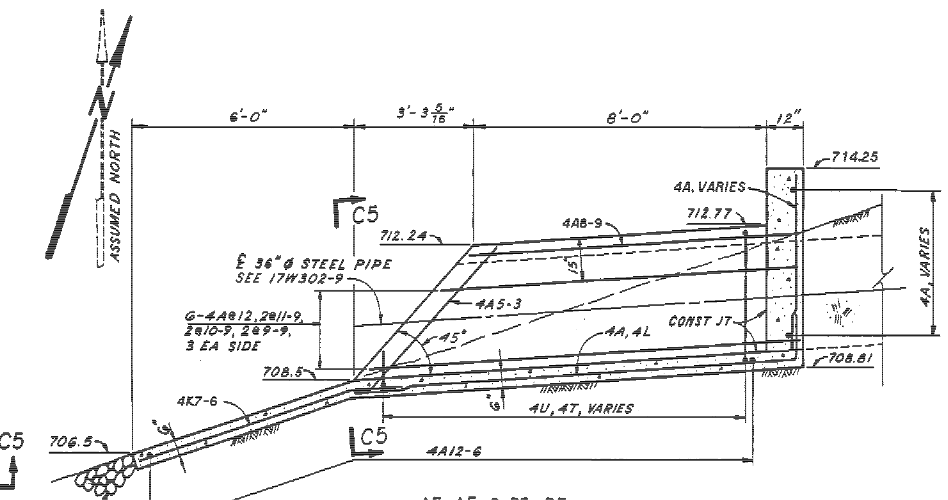
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
MANHOLES & DUCT RUNS
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W303-11 RE
FIGURE 3.8.4-46

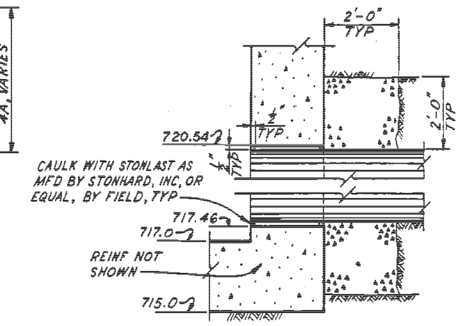
CAD MAINTAINED DRAWING



PLAN
ERCW OUTLET STR

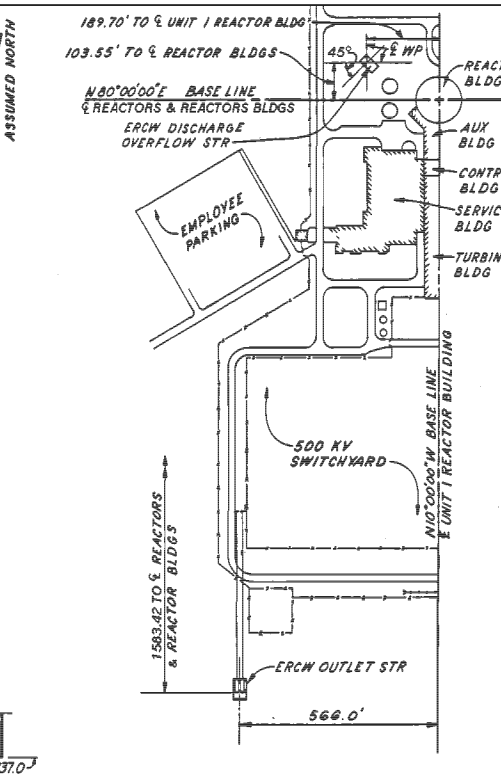


A5-A5 & B5-B5 OPP HAND
NTS



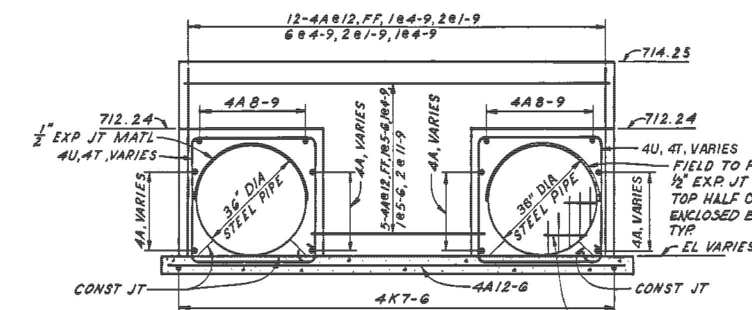
J5-J5
NTS

BENT BAR LIST							
BAR MARK	NO. REQD	BENDING DIMENSIONS					
		a	b	c	e	f	g
8U40-9	17	14-10	11-6	14-10			
8U39-6	14	14-1	11-8	14-2			
8U34-6	44	11-8	11-8	11-7			
8U28-9	12	5-10	17-6	5-10			
8XA28	18	22-7	4-7				
8H24-6	18	22-8					
8U22-9	12	5-10	11-6	5-10			
8L20-3	12	5-8	14-9				
8H19-6	34	17-8					
8U17-3	6	3-0	11-8	3-0			
8J15-9	40	14-10					
8U14-3	10	8-7	4-8	1-5			
8H13-6	66	11-8					
8J10-9	24	9-8					
8H6-6	18	4-8					
8L7	34	6-0	1-2½				
4T14-6	16	3-6	3-6				
4L13-6	5	1-7	12-0				
4U9-6	2	3-1	3-6	3-1			
4U7-6	2	2-1	3-6	2-1			
4K7-6	13	1-4	6-2		0-4		
4U5-6	2	1-1	3-6	1-1			



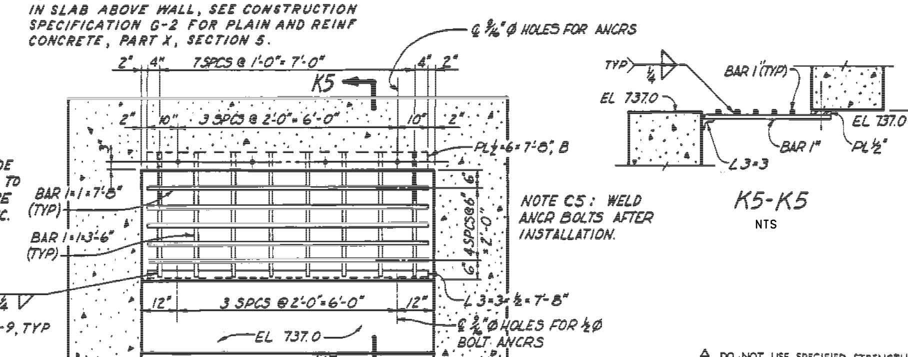
KEY PLAN
NTS

NOTE A5:
THE DISCHARGE OVERFLOW STRUCTURE SHALL BE SUPPORTED ON GRANULAR FILL ABOVE FIRM GRAVEL. ANY FINE GRAINED SOILS ABOVE THE FIRM GRAVEL SHALL BE EXCAVATED AND REPLACED WITH GRANULAR FILL TO THE BASE OF THE STR. THE GRANULAR FILL SHALL CONFORM TO SECTIONS 1032.01 AND 1032.02 OF TVA HIGHWAY SPECIFICATION T-1 AND SHALL BE COMPACTED TO 85 % RELATIVE DENSITY AS DETERMINED BY ASTM D2049.



C5-C5
NTS

NOTE B5:
FOR DELAY BEFORE PLACING CONCRETE IN SLAB ABOVE WALL, SEE CONSTRUCTION SPECIFICATION G-2 FOR PLAIN AND REINF CONCRETE, PART X, SECTION 5.



PART PLAN - EL 737.0
NTS

NOTE C5: WELD ANCR BOLTS AFTER INSTALLATION.

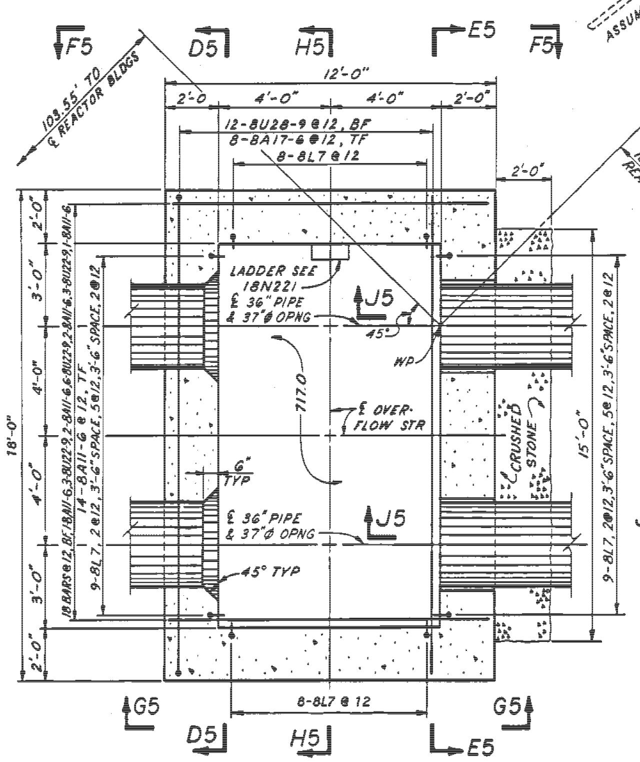
K5-K5
NTS

△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

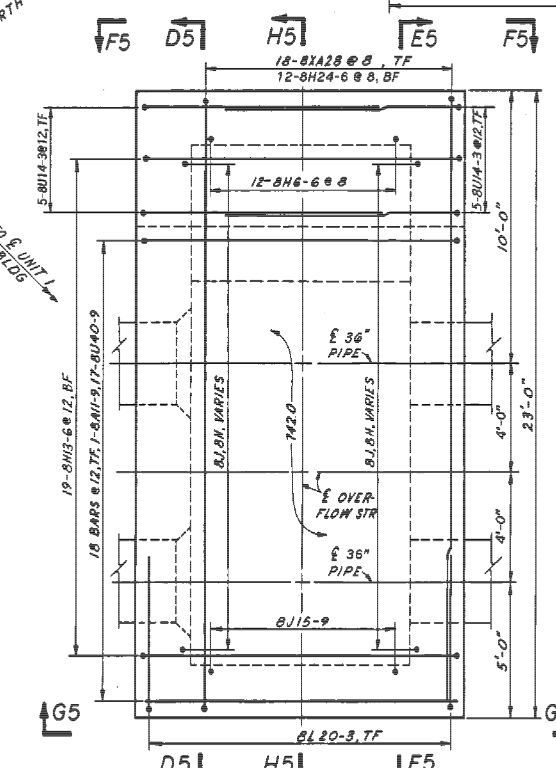
NOTES:
1. THE ERCW OUTLET STRUCTURE IS A NON-CATEGORY I STR. THE ERCW DISCHARGE OVERFLOW STRUCTURE IS A CATEGORY I STRUCTURE & QUALITY ASSURANCE IS REQD.
2. CHAMFER ALL EXPOSED EDGES ¾ INCHES UNLESS OTHERWISE NOTED.
3. THE FIRST AND LAST BAR IN A REINFORCING BILLING, UNLESS NOTED OTHERWISE, ARE POSITIONED ONE HALF OF THE DESIGNATED BAR SPACING OR 6 INCHES, WHICHEVER IS LESS, FROM THE NEAREST OUTLINE FEATURE DEFINING THE AREA BEING REINFORCED.
4. DIMENSIONS SHOWN ARE TO THE CENTERLINE OF REINFORCING BARS, UNLESS OTHERWISE NOTED.
5. CONCRETE CLEAR COVER DIMENSIONS ARE AS FOLLOWS: 3 INCHES FOR FACES CAST AGAINST EARTH OR ROCK; 2 INCHES FOR ALL OTHER FACES.
6. THE DISCHARGE OVERFLOW STRUCTURE SHALL NOT BE BACKFILLED UNTIL THE CONCRETE IN THE WALLS HAS ATTAINED A MINIMUM COMPRESSIVE STRENGTH OF 2000 PSI.
7. BACKFILL SHALL BE COMPOSED OF EARTH MATERIAL IN ACCORDANCE WITH CONSTRUCTION SPEC G-9 FOR ROLLED EARTHFILL FOR DAMS AND POWER PLANTS. EARTH MATERIAL SHALL BE PLACED AND COMPACTED IN LAYERS NOT EXCEEDING 6 INCHES TO A MINIMUM OF 95% OF THE MAXIMUM DRY DENSITY AS DETERMINED BY ASTM D-698 PROCEDURES. MOISTURE CONTENT OF THE BACKFILL SHALL BE WITHIN ±2% OF OPTIMUM.
8. SEALS IN CONSTRUCTION, CONTRACTION AND EXPANSION JOINTS ARE PVC, TYPE I, UNLESS OTHERWISE NOTED. PVC SEALS SHALL CONFORM TO SPECIFICATION PF1026.
9. ALL SEALS, EXPANSION JOINT MAT'L AND CAULKING COMPOUND BY FIELD.

△ 10. CONCRETE SHALL BE 301.5AFW AND PLACED IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-2.

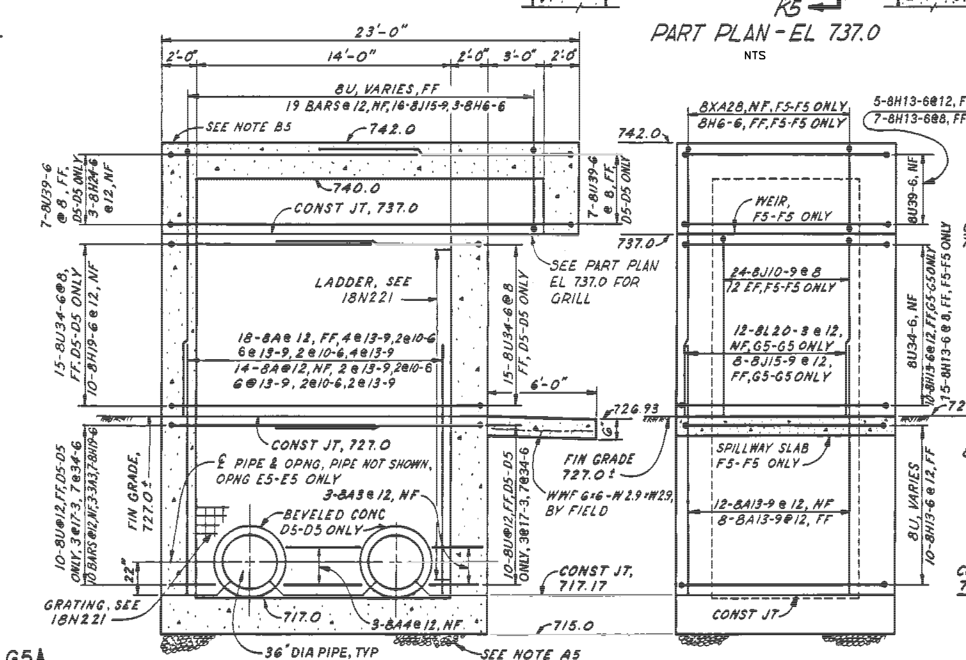
11. FOR ADDITIONAL NOTES SEE 10N318-1.



PLAN-EL 719±
ERCW DISCHARGE OVERFLOW STR

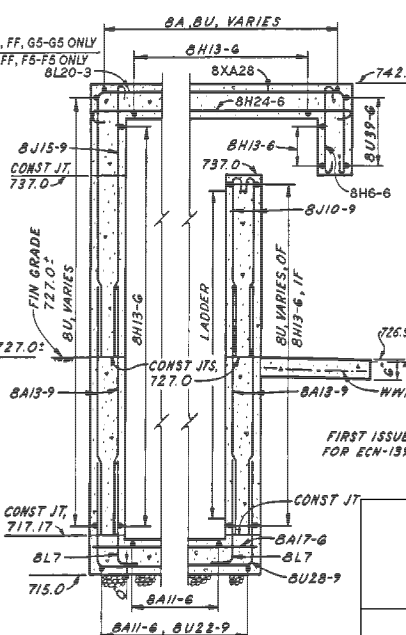


PLAN-EL 742
ERCW DISCHARGE OVERFLOW STR



D5-D5 & E5-E5 OPP HAND
NTS

F5-F5 & G5-G5 AS NOTED
NTS

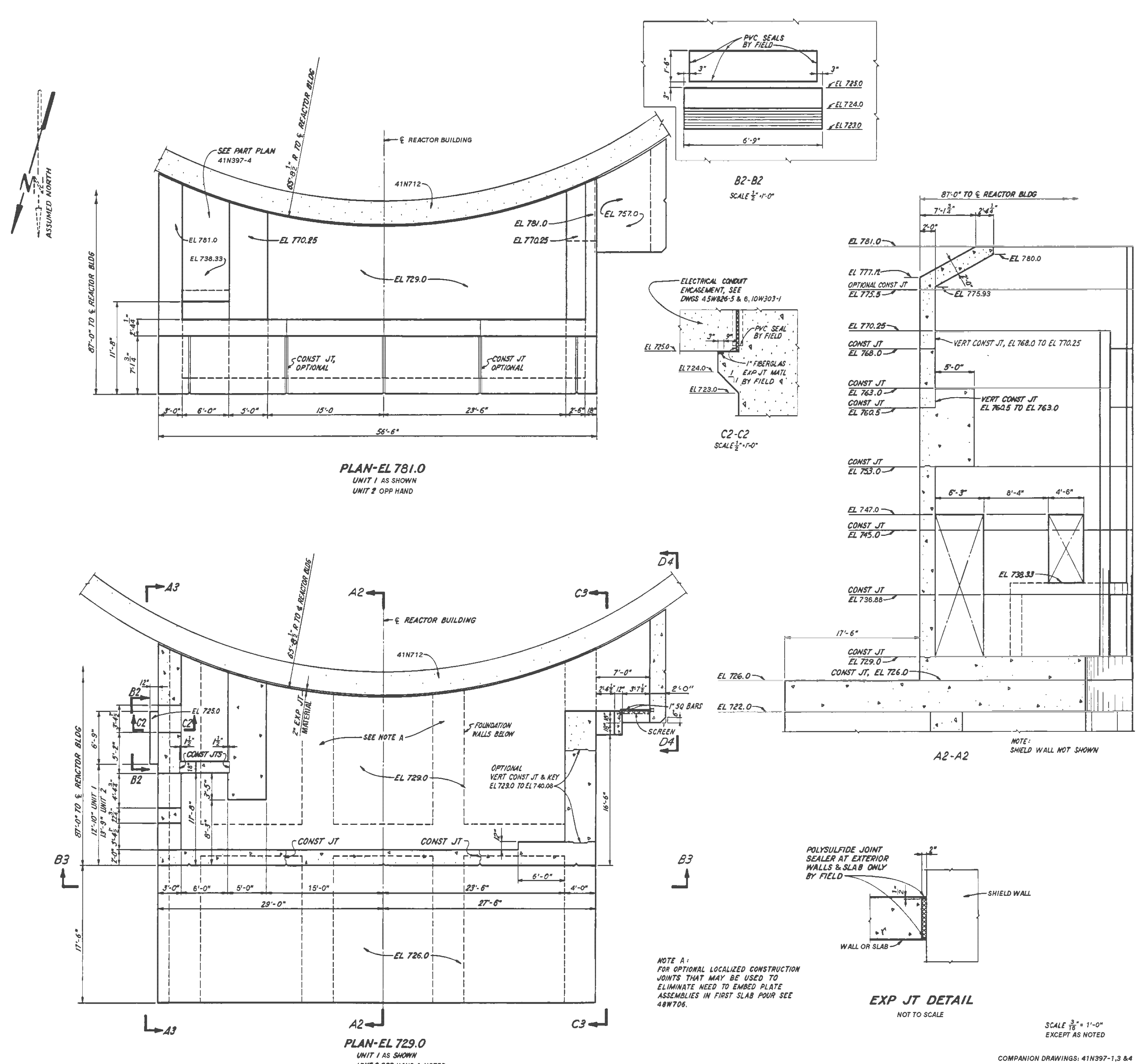


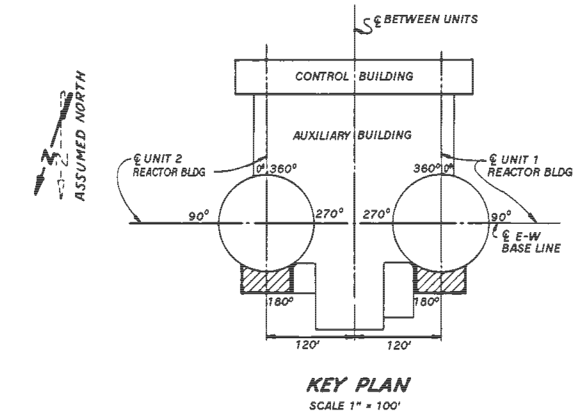
H5-H5
NTS

FIRST ISSUED FOR ECN-1395

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
ERCW STRUCTURES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W318-5 RE
FIGURE 3.8.4-46A





△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

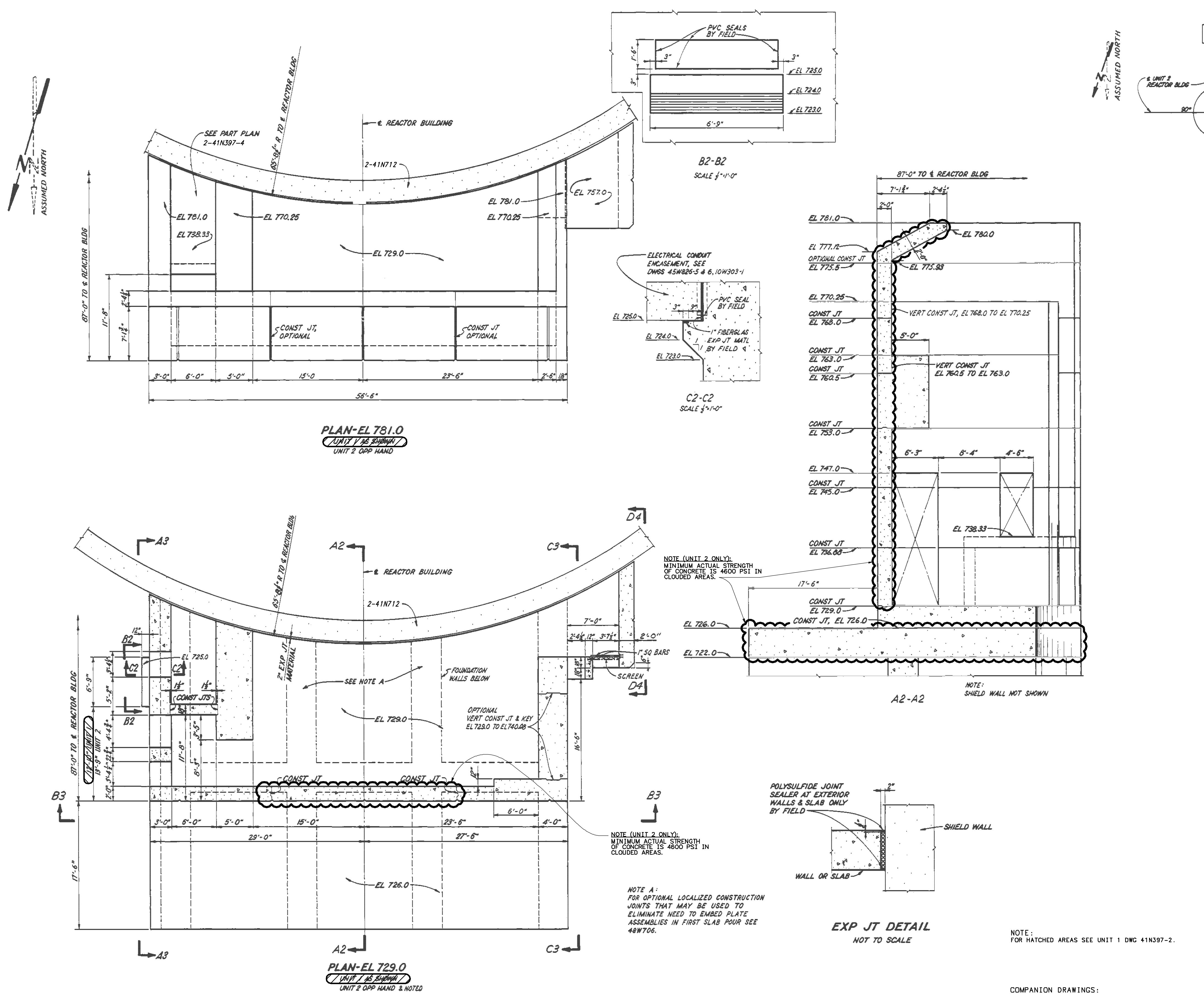
NOTES:

1. THIS BUILDING IS A CLASS 1 STRUCTURE AND QUALITY ASSURANCE IS REQUIRED.
2. FOR GENERAL NOTES, SEE 41N716-1.
3. △ CONC BET EL 722.0 & EL 729.0 SHALL BE CLASS 501.5A FW. CONCRETE ABOVE EL 729.0 SHALL BE CLASS 500.75A FW. IN ADDITION TO NORMAL REQUIREMENTS, COMPRESSIVE STRENGTH TESTS ARE REQUIRED AT 90 AND 180 DAYS TO VERIFY ULTIMATE DESIGN STRENGTH IN EXCESS OF 6750 PSI.
4. FOR EMBEDDED PARTS, PIPES, AND CONDUITS SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.
5. FLOOR TO HAVE STEEL TROWELED MONOLITHIC FINISH, SEE CONST SPEC G-22.
6. CONCRETE IN OPT TEMP OPENINGS NOT TO BE POURED UNTIL MAIN STEAM AND FEEDWATER SLEEVES ARE INSTALLED.
7. POURING RATE OF CONCRETE AGAINST EXPANSION JOINT MATERIAL SHALL NOT EXCEED 1½ FT PER HOUR.

REFERENCE DRAWING:
41BM397.... BILL OF MATERIAL

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
CONCRETE
NORTH STEAM VALVE ROOM
OUTLINE
TVA DWG NO. 41N397-2 RC
FIGURE 3.8.4-47



UNIT 2
ACTUAL CONCRETE STRENGTH
SHOWN ON THIS DRAWING
DERIVED FROM REPORT CEB-86-19-C.

NOTES:

1. THIS BUILDING IS A CLASS I STRUCTURE AND QUALITY ASSURANCE IS REQUIRED.
2. FOR GENERAL NOTES, SEE 2-41W716-1.
3. CONC BET EL 722.0 & EL 729.0 SHALL BE CLASS 501.5A FW. CONCRETE ABOVE EL 729.0 SHALL BE CLASS 5007.5A FW. IN ADDITION TO NORMAL REQUIREMENTS, COMPRESSIVE STRENGTH TESTS ARE REQUIRED AT 90 AND 180 DAYS TO VERIFY ULTIMATE DESIGN STRENGTH IN EXCESS OF 6750 PSI.
4. FOR EMBEDDED PARTS, PIPES, AND CONDUITS SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.
5. FLOOR TO HAVE STEEL TROWELED MONOLITHIC FINISH, SEE CONST SPEC G-22.
6. CONCRETE IN OPT TEMP OPENINGS NOT TO BE POURED UNTIL MAIN STEAM AND FEEDWATER SLEEVES ARE INSTALLED.
7. POURING RATE OF CONCRETE AGAINST EXPANSION JOINT MATERIAL SHALL NOT EXCEED 1½ FT PER HOUR.

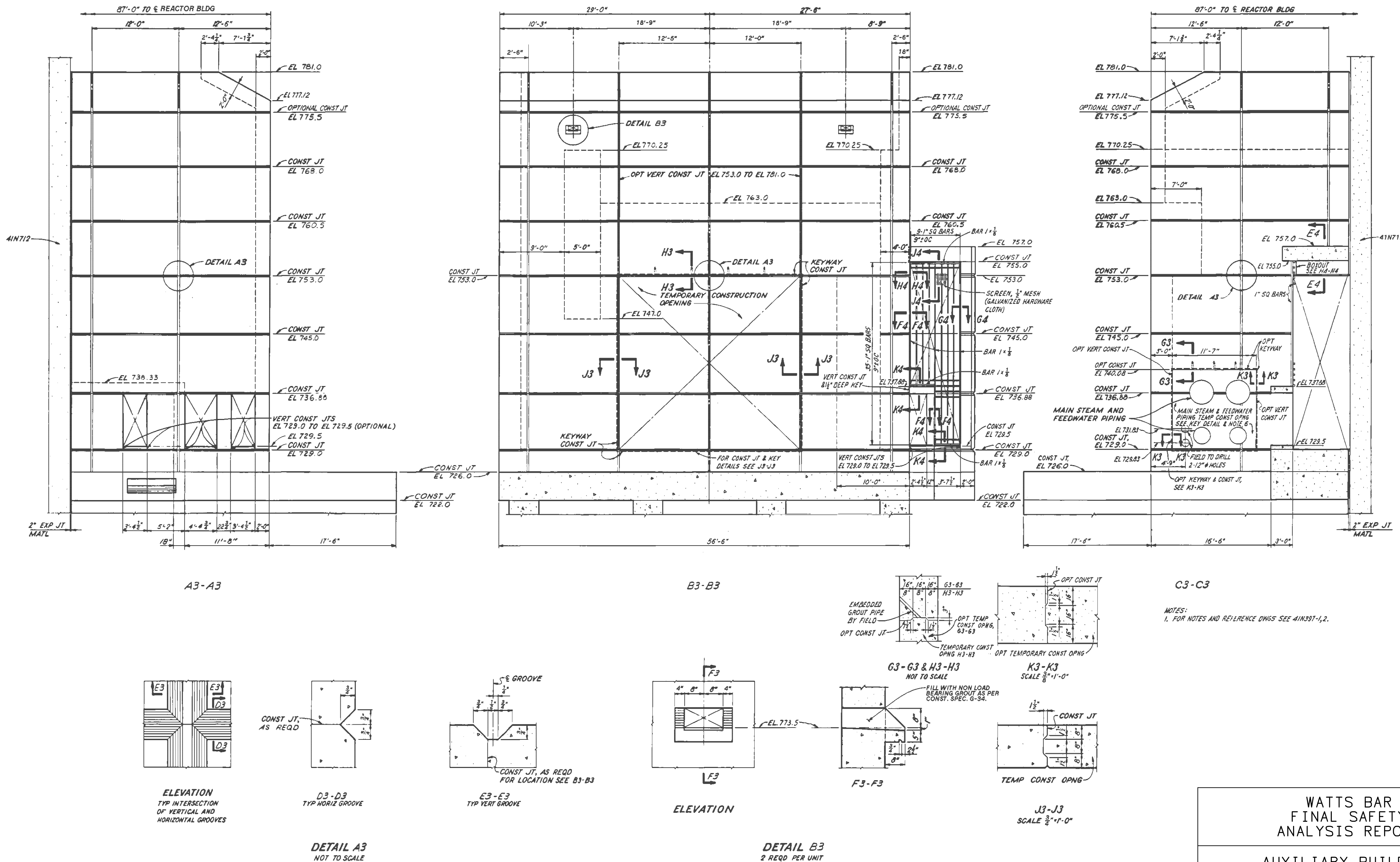
REFERENCE DRAWING:
41BM397---- BILL OF MATERIAL
41N10058-5B--CONCRETE POUR DRAWINGS

NOTE:
FOR HATCHED AREAS SEE UNIT 1 DWG 41N397-2.

COMPANION DRAWINGS:
2-41N397-1,2-41N397-4,2-41N397-3

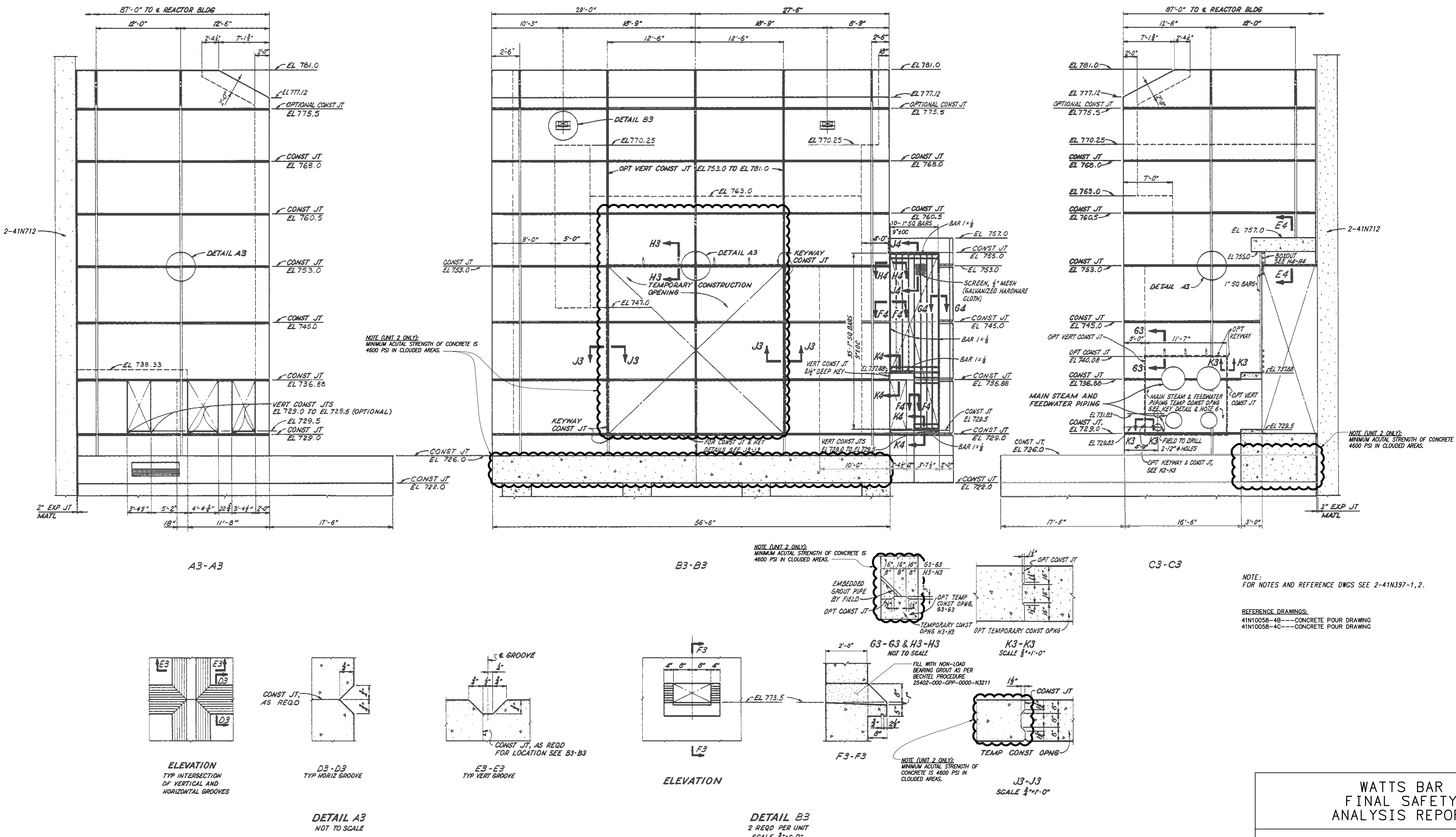
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
CONCRETE
NORTH STEAM VALVE ROOM
OUTLINE
TVA DWG NO. 2-41N397-2 RO
FIGURE 3.8.4-47(U2)



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
CONCRETE
NORTH STEAM VALVE ROOM
OUTLINE
TVA DWG NO. 41N397-3 RC
FIGURE 3.8.4-48



NOTE:
FOR NOTES AND REFERENCE DWGS SEE 2-41N397-1,2.

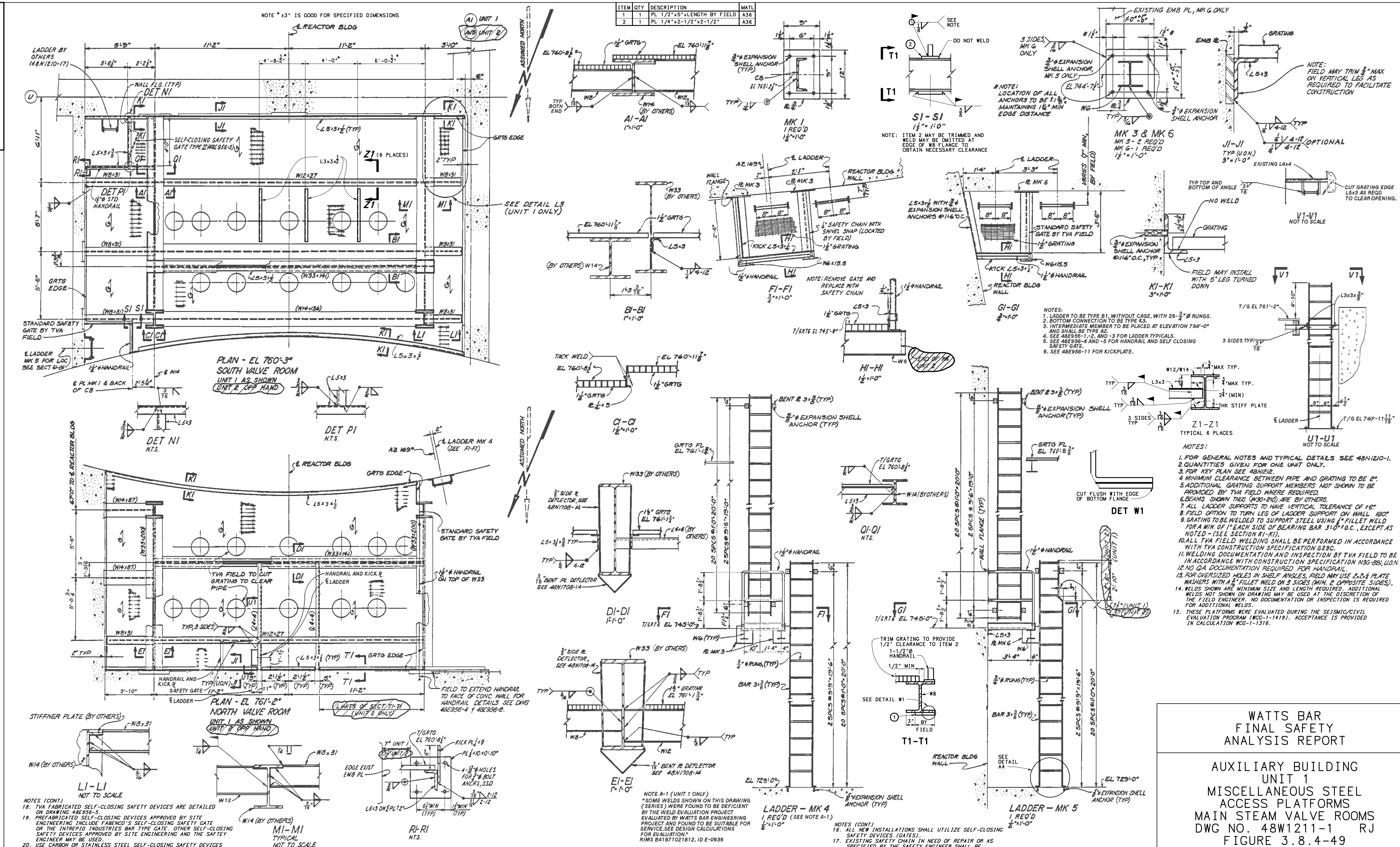
REFERENCE DRAWINGS:
41N10058-4B---CONCRETE POUR DRAWING
41N10058-4C---CONCRETE POUR DRAWING

NOTE:
FOR HATCHED AREAS SEE UNIT 1 DWG 41N397-3.

COMPANION DWGS:
2-41W397-1, 2, & 4

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

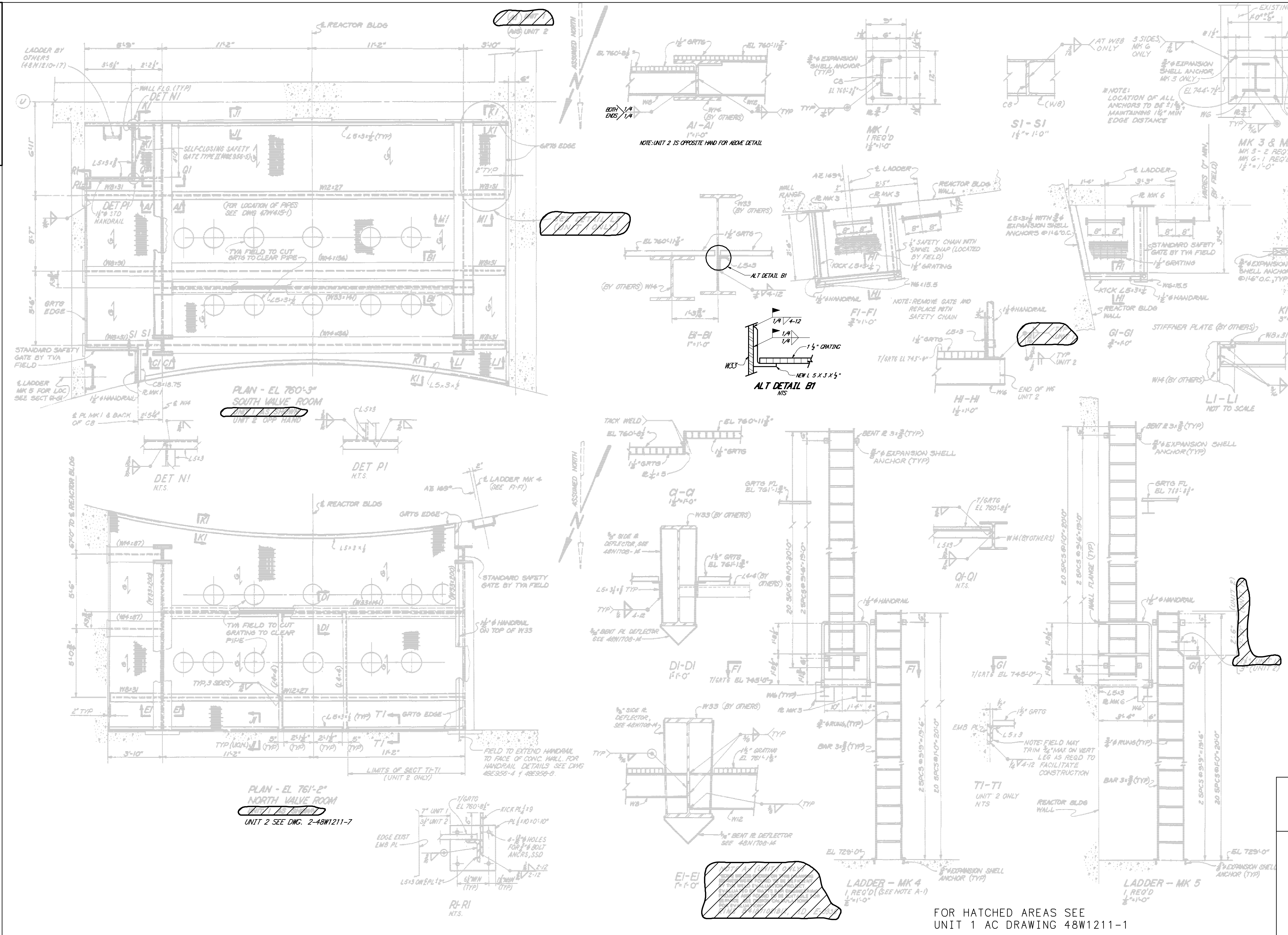
AUXILIARY BUILDING
UNIT 2
CONCRETE
NORTH STEAM VALVE ROOM
OUTLINE
TVA DWG NO. 2-41N397-3 R2
FIGURE 3.8.4-48(U2)

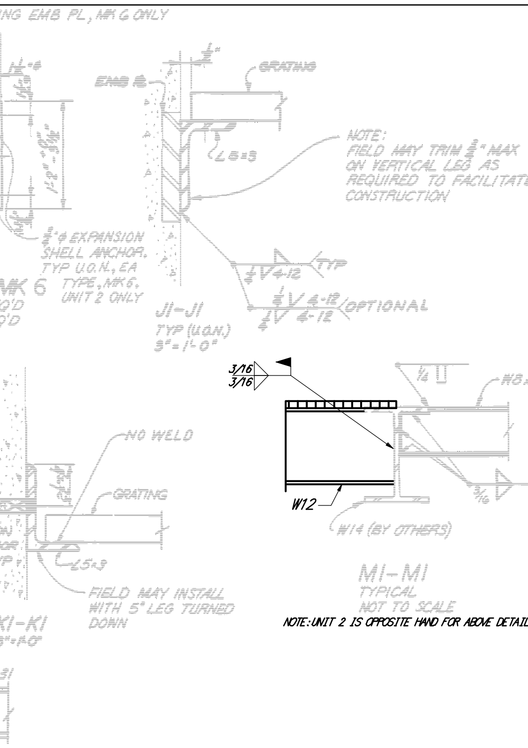


WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
MISCELLANEOUS STEEL
ACCESS PLATFORMS
MAIN STEAM VALVE ROOMS
DWG NO. 48W1211-1 RJ
FIGURE 3.8.4-49

CAD MAINTAINED DRAWING





NOTES:

1. FOR GENERAL NOTES AND TYPICAL DETAILS SEE 48N1210-1.
2. QUANTITIES GIVEN FOR ONE UNIT ONLY.
3. FOR KEY PLAN SEE 48N1212.
4. MINIMUM CLEARANCE BETWEEN PIPE AND GRATING TO BE 2".
5. ADDITIONAL GRATING SUPPORT MEMBERS NOT SHOWN TO BE PROVIDED BY TVA FIELD WHERE REQUIRED.
6. BEAMS SHOWN THUS (W30×210) ARE BY OTHERS.
7. ALL LADDER SUPPORTS TO HAVE VERTICAL TOLERANCE OF ±6".
8. FIELD OPTION TO TURN LEG OF LADDER SUPPORT ON WALL 180°.
9. GRATING TO BE WELDED TO SUPPORT STEEL USING 1/8" FILLET WELD FOR A MIN. OF 1" EACH SIDE OF BEARING BAR 3'-0" ±O.C., EXCEPT AS NOTED - (SEE SECTION K1-K1).
10. ALL TVA FIELD WELDING SHALL BE PERFORMED IN ACCORDANCE WITH TVA CONSTRUCTION SPECIFICATION G29C.
11. WELDING DOCUMENTATION AND INSPECTION BY TVA FIELD TO BE IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION N3G-881, U.O.N.
12. NO QA DOCUMENTATION REQUIRED FOR HANDRAIL.
13. FOR OVERSIZED HOLES IN SHELF ANGLES, FIELD MAY USE 2×2×1/4 PLATE WASHERS WITH A 3/16" FILLET WELD ON 3 SIDES (MIN. 2 OPPOSITE SIDES).
14. WELDS SHOWN ARE MINIMUM SIZE AND LENGTH REQUIRED. ADDITIONAL WELDS NOT SHOWN ON DRAWING MAY BE USED AT THE DISCRETION OF THE FIELD ENGINEER. NO DOCUMENTATION OR INSPECTION IS REQUIRED FOR ADDITIONAL WELDS.
15. HANDRAILS, SELF CLOSING GATES, GRATING, KICK PLATES, LADDERS, AND ASSOCIATED HARDWARE MAY BE PROCURED AND FABRICATED CONFORMING TO SEISMIC II/I CLASSIFICATION (TVA CATEGORY I(L).

MODIFICATION NOTES:

1. ALL STEEL (A36) UNLESS OTHERWISE NOTED.
2. STRUCTURAL STEEL ERECTION SHALL BE PERFORMED IN ACCORDANCE WITH BECHTEL PROCEDURE 25402-000-GPP-0000-N3221.
3. ALL WELDING SHALL BE PERFORMED IN ACCORDANCE WITH BECHTEL SPM 25402-000-GMX-GCE-00001.
4. PAINTING AND COATING SHALL BE IN ACCORDANCE WITH BECHTEL PROCEDURE 25402-000-GPP-0000-N3222.

FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 48W1211-1

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
MISCELLANEOUS STEEL
ACCESS PLATFORMS
MAIN STEAM VALVE ROOMS
DWG NO. 2-48W1211-1 R3
FIGURE 3.8.4-49(U2)

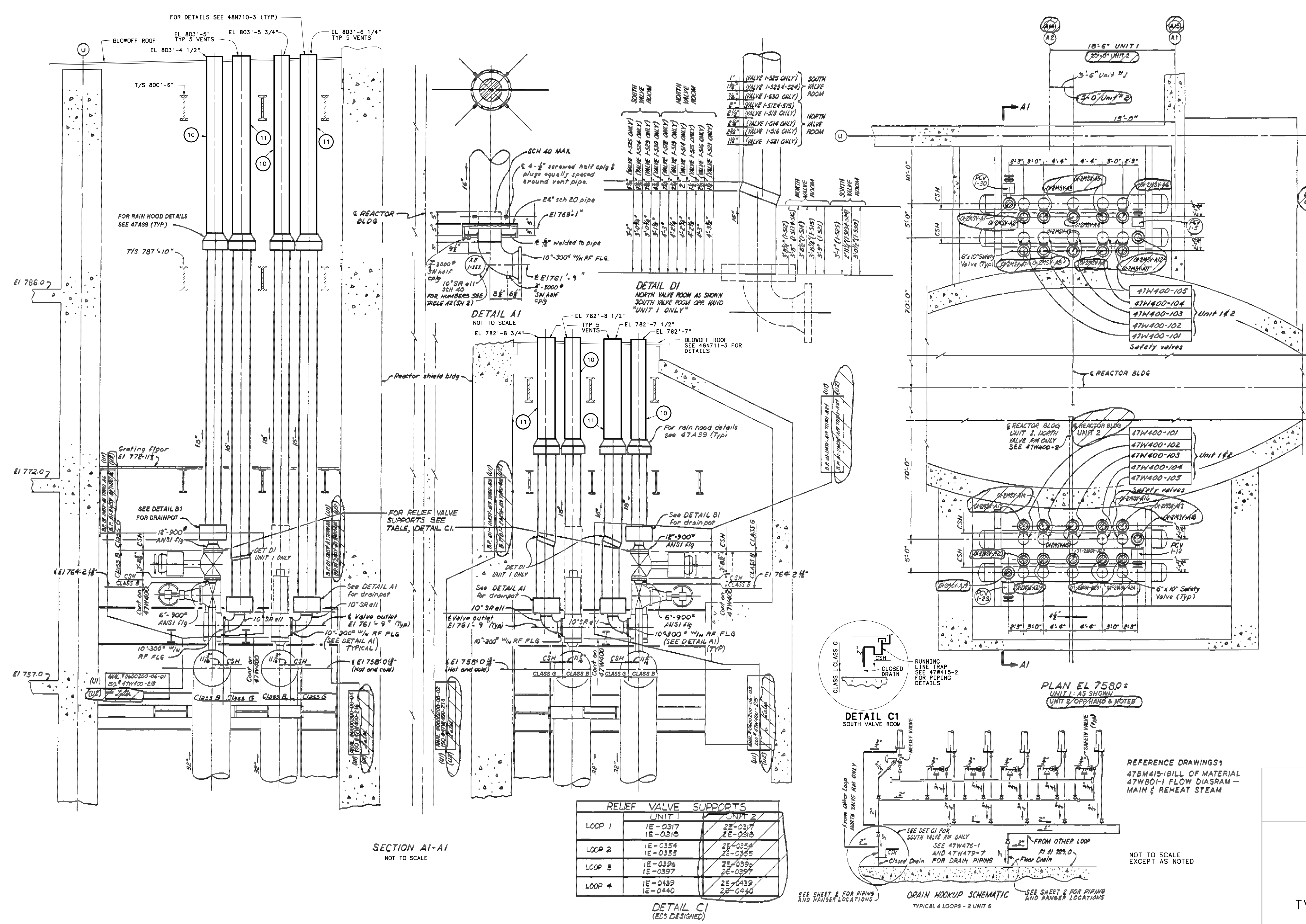
Figure 3.8.4-49A

Deleted

FIGURE 3.8.4-49B

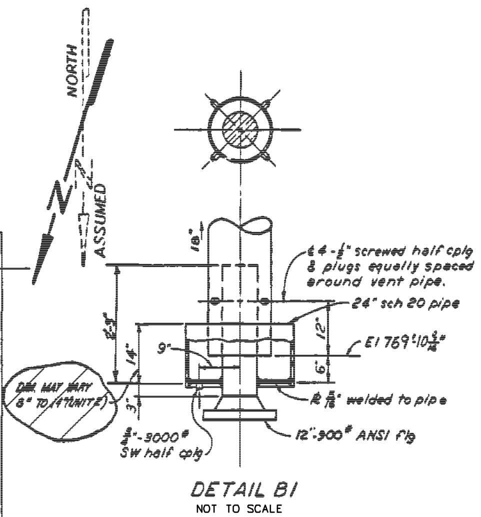
DELETED

CAD MAINTAINED DRAWING



RELIEF VALVE SUPPORTS	UNIT 1	UNIT 2
LOOP 1	1E-0317 1E-0318	2E-0317 2E-0318
LOOP 2	1E-0354 1E-0355	2E-0354 2E-0355
LOOP 3	1E-0396 1E-0397	2E-0396 2E-0397
LOOP 4	1E-0439 1E-0440	2E-0439 2E-0440

DETAIL C1
(EDS DESIGNED)



NOTES:

1. FOR GENERAL SPECIFICATIONS SEE TVA SPEC 1521, CONTR 74C38-83015.
2. FOR DETAILED SPECIFICATIONS SEE TVA BILL OF MATERIAL 47BM415-1.
3. CSH DENOTES "CONTRACT STARTS HERE."
4. UNIT 1 AS SHOWN; UNIT 2 OPPOSITE HAND, EXCEPT AS NOTED.
5. ALL WELDING RINGS, COMPANION FLANGES, BOLTS, NUTS, AND GASKETS REQUIRED AT CONTRACT STARTING AND STOPPING POINTS SHALL BE FURNISHED BY PIPING CONTRACTOR UNLESS OTHERWISE NOTED. (SEE NOTE 13.)
6. ALL PIPING 4" AND LARGER TO BE SUPPORTED USING BERGEN-PATERSON INDIVIDUAL HANGER DWG 01A-SERIES.
7. FOR DETAILS OF PIPE JOINTS FOR BUTT WELDING SEE 47B18 AND 47B19, A AND B.
8. FIELD SHALL DETERMINE EXACT LOCATION OF PIPING 3" AND SMALLER, AND SHALL SUPPORT USING TVA 47A058 SERIES TYPICALS(≤ 2 1/2") OR 47A059 SERIES TYPICALS(> 2 1/2").
9. 01-IMSV-XXX INDICATES A UNIT 1 HANGER LOCATION.UNIT 2 LOCATIONS ARE SIMILAR EXCEPT NUMBERED 01-2MSV-XXX. (BERGEN-PATERSON)
10. FOR AUXILIARY CONNECTIONS, DRAINS, AND VENT CONNECTIONS SEE 47A7.
11. ALL PIPING AND FITTING TO BE TVA CLASS G, ANSI B31.1, UNLESS OTHERWISE NOTED.
12. ALL PIPING SHALL BE FIELD CONSTRUCTED IN ACCORDANCE WITH CONST SPEC N3M-868, CLASS G, AS NOTED ON DRAWINGS.
13. STUDS AND NUTS FURNISHED FOR SAFETY VALVES BY PIPING CONTRACTOR
14. CLEANING DURING FABRICATION AND ERECTION SHALL BE IN ACCORDANCE WITH CONST SPEC NO, G-39, CLEANING DURING FABRICATION OF FLUID HANDLING COMPONENTS, ACCEPTANCE CRITERIA CLASS B CLEANLINESS LEVEL USING GRADE C WATER.
15. ALL DRAIN PIPING IS TVA CLASS G.

FOR MANUFACTURER'S DETAILS OF:

VALVE	MANUFACTURER	DWG	CONTR
RELIEF	COPES-VULCAN	E-174269	76K51-83081
SAFETY	CONSOLIDATED	SNC-040	76K51-83082

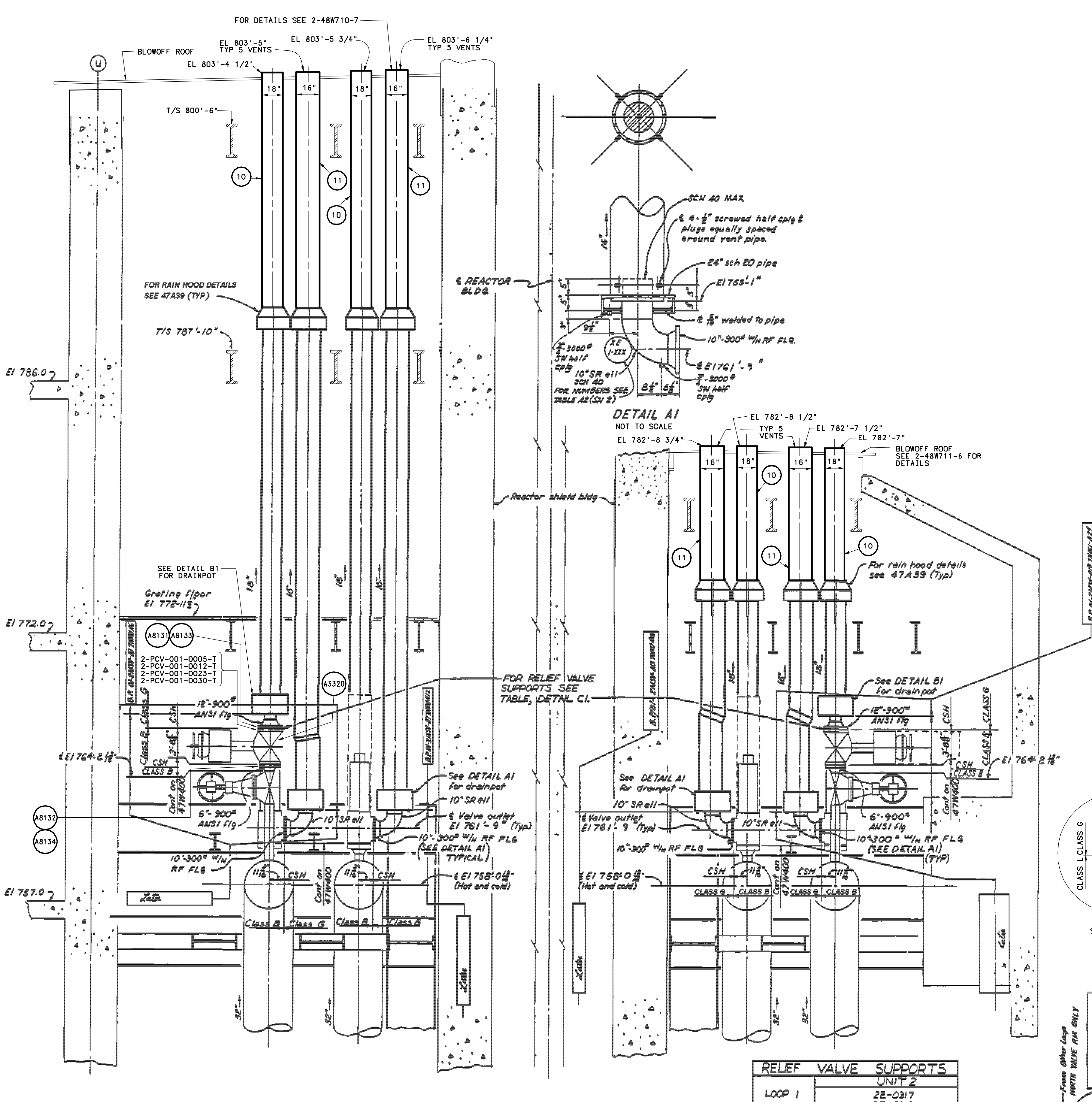
16. "SOME WELDS SHOWN ON THIS DRAWING (SERIES) WERE FOUND TO BE DEFICIENT BY THE WELD EVALUATION PROJECT, EVALUATED BY WATTS BAR ENGINEERING PROJECT AND FOUND TO BE SUITABLE FOR SERVICE. SEE DESIGN CALCULATIONS FOR EVALUATION". REFER TO RIMS NO. B26890901319 AND ALSO PIR WBN MEB8669.
17. ALL MAIN STEAM SAFETY VALVE MARK NUMBERS ARE INTERCHANGEABLE AND MAY BE INSTALLED IN THE SYSTEM AS REQUIRED FOLLOWING REFURBISHMENT.

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
MECHANICAL
MAIN STEAM RELIEF &
SAFETY VALVE VENTS
TVA DWG NO. 47W415-1 RK
FIGURE 3.8.4-49C

CAD MAINTAINED DRAWING

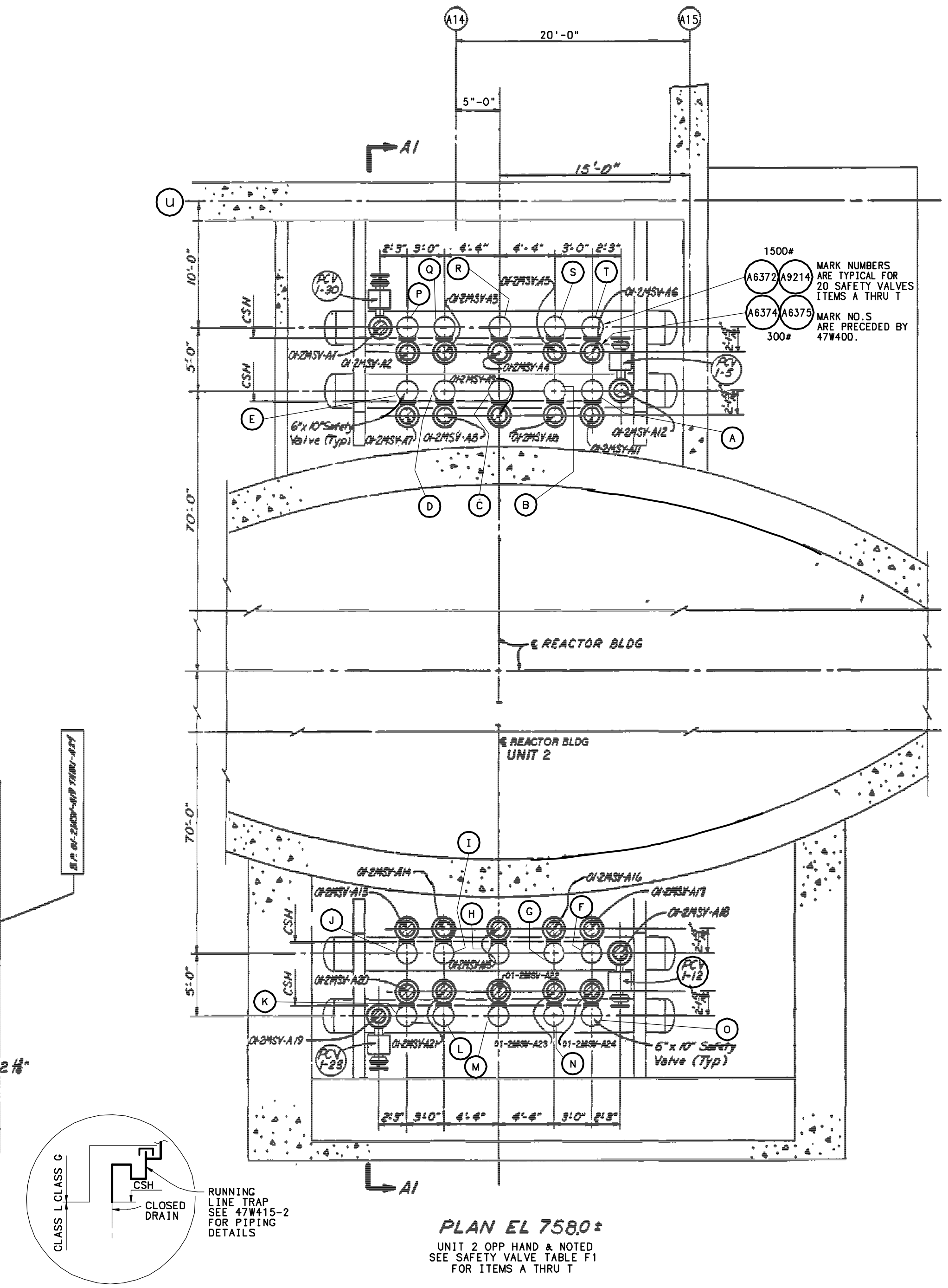


SECTION A1-A1
NOT TO SCALE

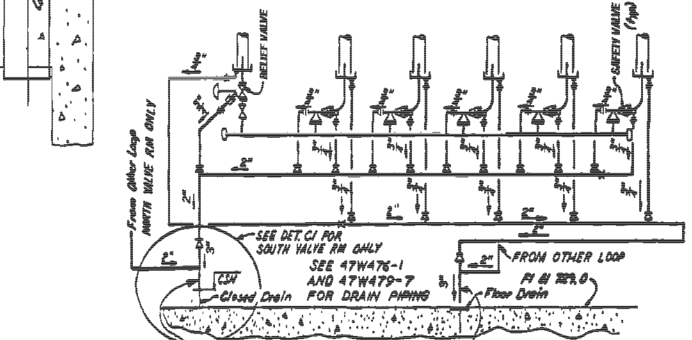
DETAIL A1
NOT TO SCALE

RELIEF VALVE SUPPORTS	
	UNIT 2
LOOP 1	2E-0317 2E-0318
LOOP 2	2E-0354 2E-0355
LOOP 3	2E-0396 2E-0397
LOOP 4	2E-0439 2E-0440

PLAN EL 758.0±
UNIT 2 OPP HAND & NOTED
SEE SAFETY VALVE TABLE F1
FOR ITEMS A THRU T



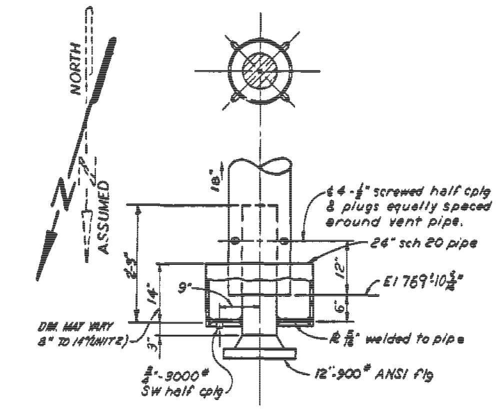
DETAIL C1
SOUTH VALVE ROOM



DRAIN HOOKUP SCHEMATIC
TYPICAL 4 LOOPS -

TABLE F1 (SEE NOTE 16)
SAFETY VALVE

ITEM	SAFETY VALVE UNID	SET PRESSURE (PSIG)
A	2-SFV-001-0522	1224
B	2-SFV-001-0523	1215
C	2-SFV-001-0524	1205
D	2-SFV-001-0525	1195
E	2-SFV-001-0526	1185
F	2-SFV-001-0517	1224
G	2-SFV-001-0518	1215
H	2-SFV-001-0519	1205
I	2-SFV-001-0520	1195
J	2-SFV-001-0521	1185
K	2-SFV-001-0512	1224
L	2-SFV-001-0513	1215
M	2-SFV-001-0514	1205
N	2-SFV-001-0515	1195
O	2-SFV-001-0516	1185
P	2-SFV-001-0527	1224
Q	2-SFV-001-0528	1215
R	2-SFV-001-0529	1205
S	2-SFV-001-0530	1195
T	2-SFV-001-0531	1185



DETAIL B1
NOT TO SCALE

NOTES:
1. FOR GENERAL SPECIFICATIONS SEE TVA SPEC 1521, CONTR 74C38-83015.
2. FOR DETAILED SPECIFICATIONS SEE TVA BILL OF MATERIAL 47BM415-1.
3. CSH DENOTES "CONTRACT STARTS HERE."
4. UNIT 1 AS SHOWN; UNIT 2 OPPOSITE HAND, EXCEPT AS NOTED.
5. ALL WELDING RINGS, COMPANION FLANGES, BOLTS, NUTS, AND GASKETS REQUIRED AT CONTRACT STARTING AND STOPPING POINTS SHALL BE FURNISHED BY PIPING CONTRACTOR UNLESS OTHERWISE NOTED. (SEE NOTE 13.)
6. ALL PIPING 4" AND LARGER TO BE SUPPORTED USING BERGEN-PATERSON INDIVIDUAL HANGER DWG 01A-SERIES.
7. FOR DETAILS OF PIPE JOINTS FOR BUTT WELDING SEE 47B18 AND 47B19, A AND B.
8. FIELD SHALL DETERMINE EXACT LOCATION OF PIPING 3" AND SMALLER, AND SHALL SUPPORT USING TVA 47A058 SERIES TYPICALS(≤ 2 1/2") OR 47A059 SERIES TYPICALS(> 2 1/2").
9. 01-1MSV-XXX INDICATES A UNIT 1 HANGER LOCATION.UNIT 2 LOCATIONS ARE SIMILAR EXCEPT NUMBERED 01-2MSV-XXX. (BERGEN-PATERSON)
10. FOR AUXILIARY CONNECTIONS, DRAINS, AND VENT CONNECTIONS SEE 47A7.
11. ALL PIPING AND FITTING TO BE TVA CLASS G, ANSI B31.1, UNLESS OTHERWISE NOTED.
12. ALL PIPING SHALL BE FIELD CONSTRUCTED IN ACCORDANCE WITH CONST SPEC N3M-868, CLASS G, AS NOTED ON DRAWINGS.
13. STUDS AND NUTS FURNISHED FOR SAFETY VALVES BY PIPING CONTRACTOR
14. CLEANING DURING FABRICATION AND ERECTION SHALL BE IN ACCORDANCE WITH CONST SPEC NO. G-39, CLEANING DURING FABRICATION OF FLUID HANDLING COMPONENTS, ACCEPTANCE CRITERIA CLASS B CLEANLINESS LEVEL USING GRADE C WATER.
15. ALL DRAIN PIPING IS TVA CLASS G.
16. ALL MAIN STEAM SAFETY VALVE MARK NUMBERS ARE INTERCHANGEABLE AND MAY BE INSTALLED IN THE SYSTEM AS REQUIRED FOLLOWING REFURBISHMENT.

FOR MANUFACTURER'S DETAILS OF:

VALVE	MANUFACTURER	DWG	CONTR
RELIEF	COPES-VULCAN	E-174269	76K51-83081
SAFETY	CONSOLIDATED	3NC-040	76K51-83082
SAFETY	CONSOLIDATED	3NC2034	P09096

REFERENCE DRAWINGS:

47BM415-1 BILL OF MATERIAL
47W801-1 FLOW DIAGRAM — MAIN & REHEAT STEAM

NOT TO SCALE
EXCEPT AS NOTED

AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

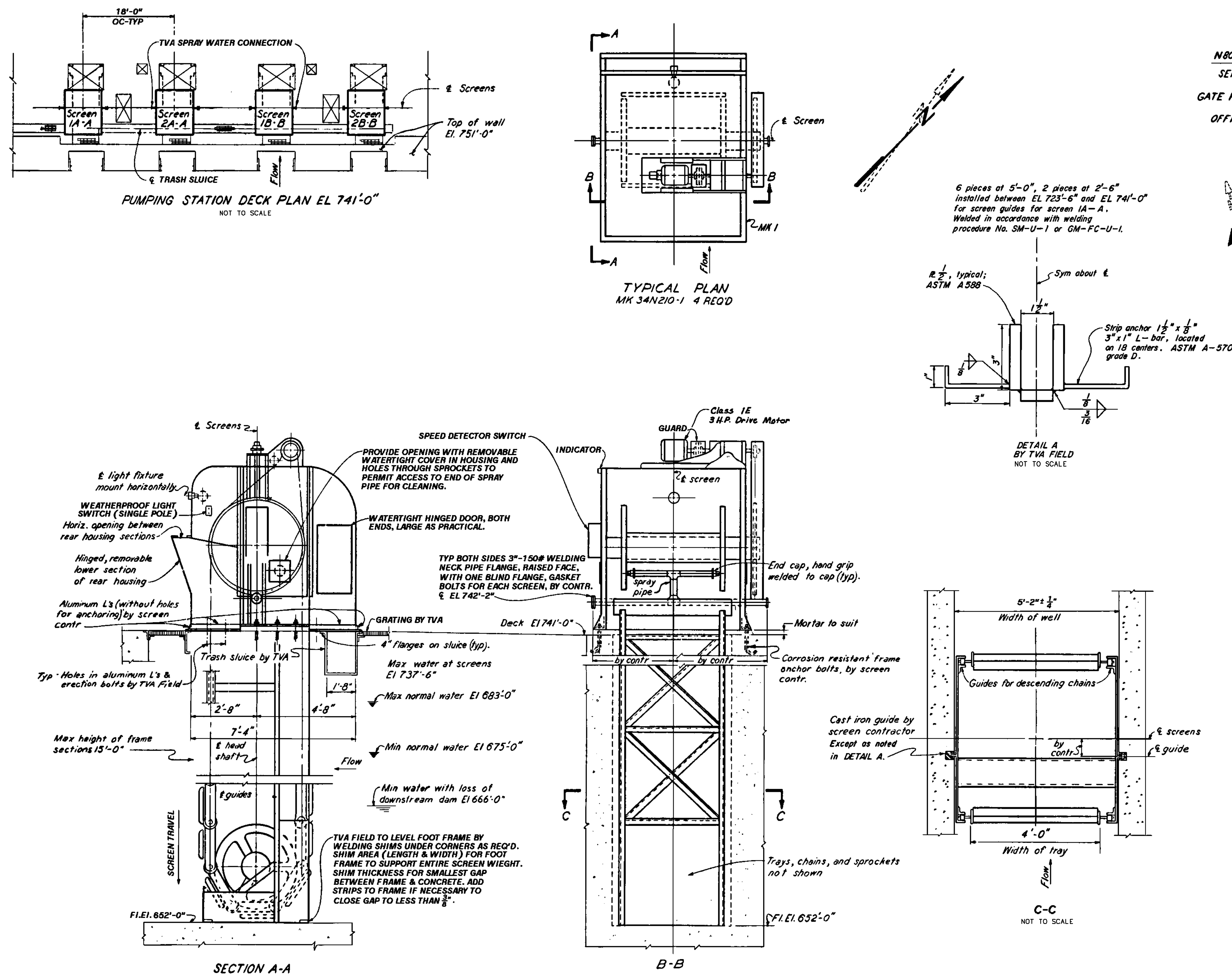
REACTOR BUILDING
UNIT 2
MECHANICAL
MAIN STEAM RELIEF &
SAFETY VALVE VENTS
TVA DWG NO. 2-47W415-1 R1
FIGURE 3.8.4-49C(U2)

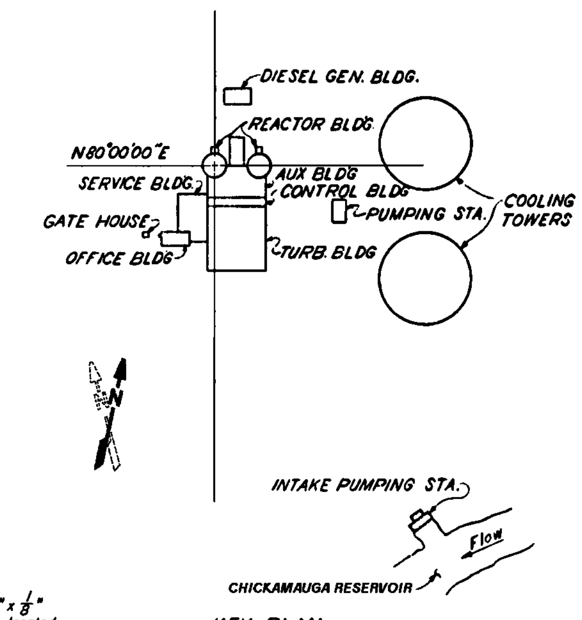
SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-50

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.8.4-51





KEY PLAN
NOT TO SCALE

For Manufacturer's details see Envirex Inc. File, TVA contract 74C38-85143.

NOTES:
Four screens required.
Capacity of each screen-25,000 gpm of clear water for clean screen with minimum headwater El 666'0" and max water velocity of 2.0 fps.
Available spray water per screen to be 120 gpm at 90 psi at the nozzles.
Control station furnished by TVA.

QUALITY ASSURANCE

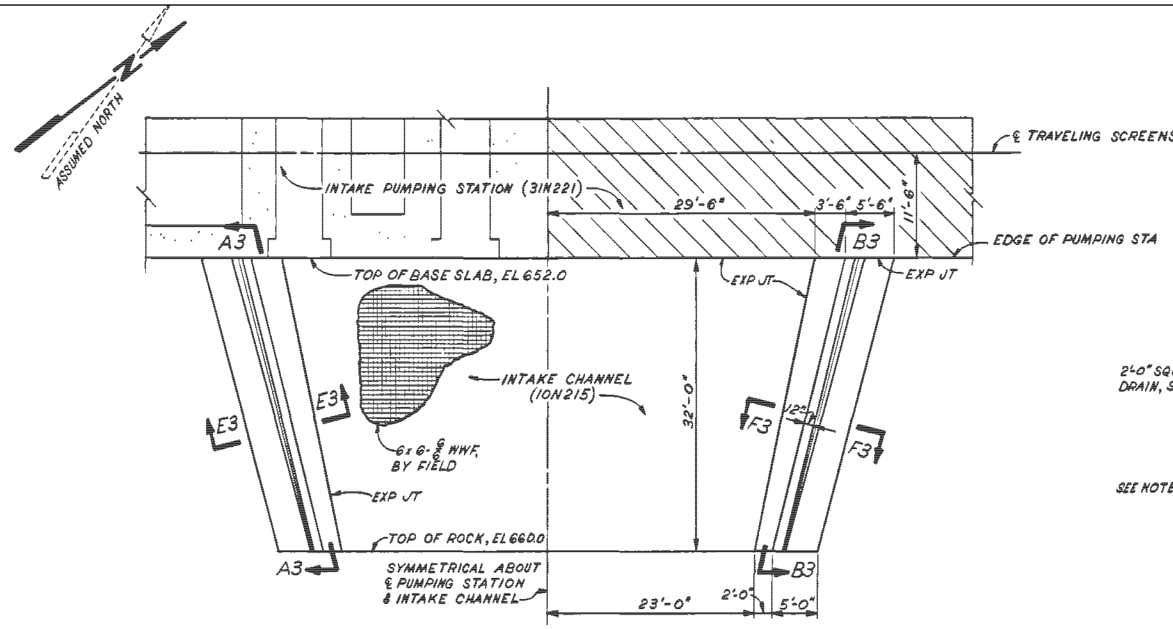
1. SCREENS ARE CATEGORY 1 EQUIPMENT.
2. CONTRACTOR REQUIREMENTS IN ACCORDANCE WITH TVA SPECIFICATION 2023
3. FIELD REQUIREMENTS:
   A. FIELD TO INSTALL SCREENS IN ACCORDANCE WITH ACCEPTED SCREEN ERECTION PRACTICES.
   B. FIELD TOUCHUP OF PAINT IN ACCORDANCE WITH CHIEF ARCHITECT'S INSTRUCTIONS.
4. ALL MATERAILS BY TVA FIELD AND FIELD WORK SHALL BE INACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

SCALE: NOT TO SCALE
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

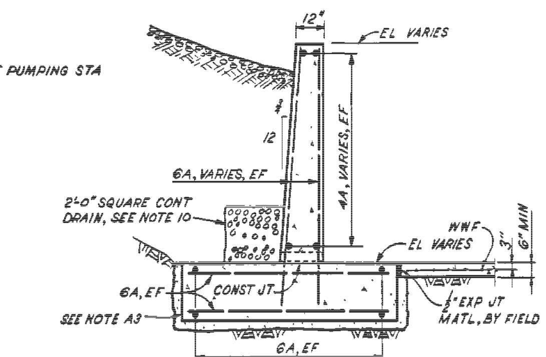
WATER SUPPLY
UNITS 1 & 2
INTAKE PUMPING STATION
TRAVELING SCREENS
ARRANGEMENT
TVA DWG 34N210 RB
FIGURE 3.8.4-52

CAD MAINTAINED DRAWING



PLAN - EL 663.5±

NTS



E3-E3 & F3-F3

NOTE A3:
THE 4 INCH SUBPOUR MAY BE ELIMINATED ON THE BOTTOM OF THE FOOTING IF THE STRUCTURAL CONCRETE CAN BE PLACED WITHIN 48 HOURS AFTER ROCK EXPOSURE.

METHOD OF EXCAVATION:

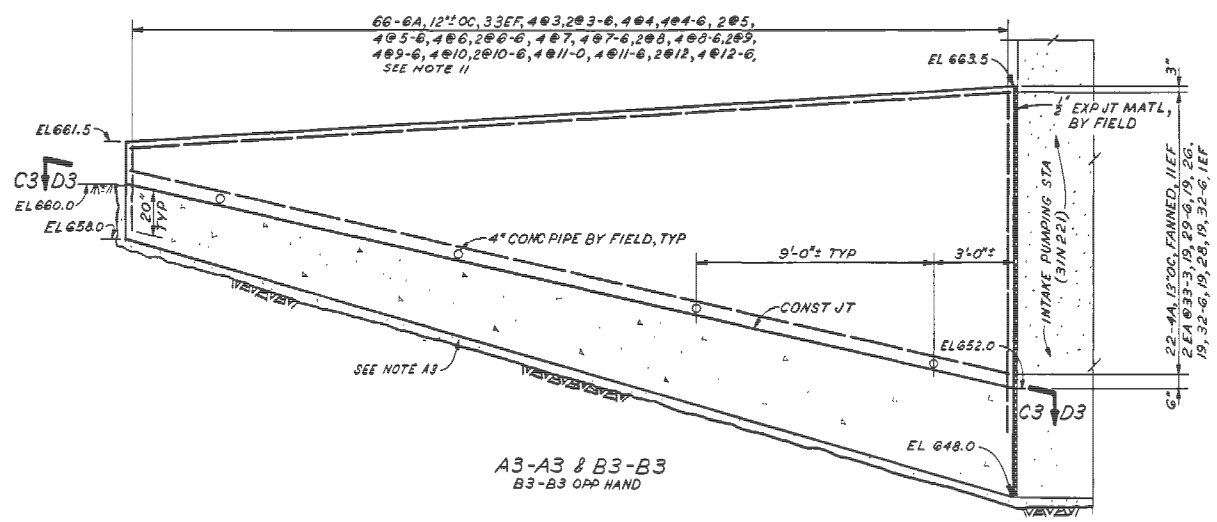
△ I EXCAVATE ROCK IN A SAW-TOOTH MANNER TO A FINAL LOCATION, A MINIMUM OF 4 INCHES BEYOND THE OUTLINE SHOWN FOR THE STRUCTURAL CONCRETE. PLACE A CONCRETE SUBPOUR TO ELEVATIONS AND DIMENSIONS SHOWN. CONCRETE FOR SUBPOURS SHALL BE CLASS 201.5 BFW OR 201.5 AFW IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-2.

△ II FOR THE AREA BETWEEN THE RETAINING WALLS EXCAVATE ROCK TO FINAL LOCATION A MINIMUM OF 6 INCHES BELOW THE ELEVATIONS SHOWN ON THE INTAKE CHANNEL DRAWING 10N215. PLACE CONCRETE REINFORCED WITH 6x6 WWF TO FINAL GRADE. CONCRETE SHALL BE CLASS 201.5 BFW OR CLASS 201.5 AFW IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-2.

III IT IS ANTICIPATED THAT EXCAVATIONS OF ROCK CAN BE ACCOMPLISHED WITH POWER EQUIPMENT AND BLASTING SHALL NOT BE PERMITTED.

IV SURFACES OF ROCK THAT ARE TO HAVE CONCRETE PLACED AGAINST THEM SHALL BE CLEANED OF ALL LOOSE AND SCALY MATERIAL BY USE OF ALL OR COMBINATIONS OF THE FOLLOWING: HAND SHOVELS, STIFF BROOMS AND AIR JET.

V THE SUBPOUR SHALL BE PLACED WITHIN 48 HOURS AFTER ROCK EXPOSURE.



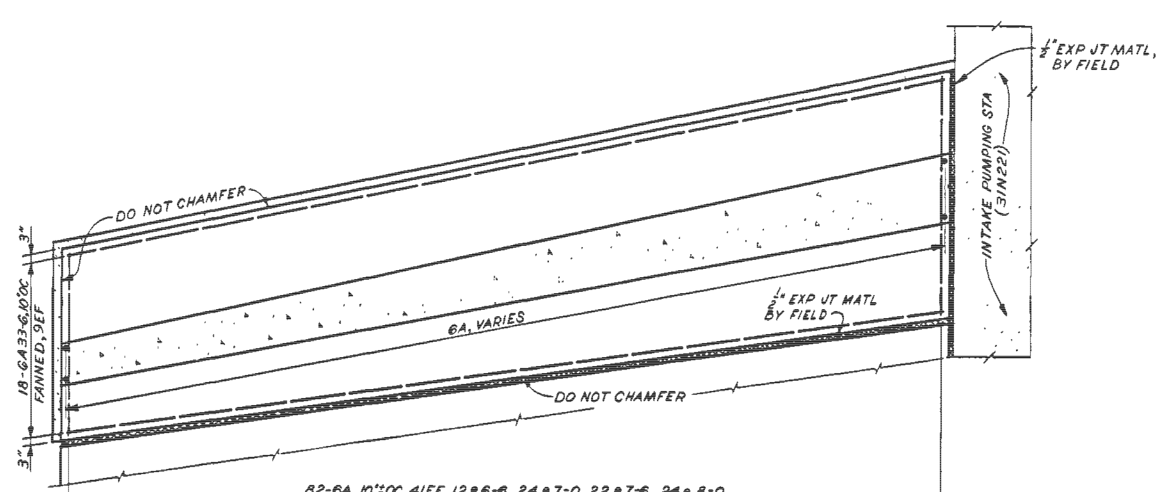
A3-A3 & B3-B3

B3-B3 OPP HAND

△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

NOTES:

1. THE CONCRETE RETAINING WALLS ARE CATEGORY I STRUCTURES AND QUALITY ASSURANCE IS REQUIRED.
△ 2. CONCRETE SHALL BE CLASS 301.5A FW AND PLACED IN ACCORDANCE WITH CONSTRUCTION SPEC G-2.
3. CHAMFER ALL EXPOSED EDGES 3/4" UNLESS OTHERWISE NOTED.
4. ALL REINFORCEMENT SHALL CONFORM TO ASTM SPECIFICATION A615, GRADE 60, WITH ADDITIONAL TESTING IN ACCORDANCE WITH CONSTRUCTION SPEC G-2.
5. WELDING OF OR TO REINFORCING BARS IS PROHIBITED WITHOUT PRIOR APPROVAL OF THE CIVIL ENGINEERING & DESIGN BRANCH CHIEF.
6. THE CLEAR COVER BETWEEN THE FACE OF CONCRETE AND NEAREST REINFORCING BAR IS 3" FOR BOTTOM FACE REINFORCEMENT AND 2" FOR ALL OTHER REINFORCEMENT UNLESS OTHERWISE NOTED.
7. THE FIRST AND LAST BAR IN A REINFORCING BILLING, UNLESS NOTED OTHERWISE, ARE POSITIONED ONE HALF OF THE DESIGNATED BAR SPACING OR 6 INCHES, WHICHEVER IS LESS, FROM THE NEAREST OUTLINE FEATURE DEFINING THE AREA BEING REINFORCED.
8. THE ELEVATIONS AND DIMENSIONS SHOWN INDICATE THE MINIMUM EXCAVATION REQUIRED.
9. EXPANSION JOINT MATERIAL CONFORMING TO ASTM SPECIFICATION D1752-67, TYPE II TO BE PROVIDED BY FIELD.
10. FILTER MATERIAL SHALL BE IN ACCORDANCE WITH SPECIFICATION TI, SECTION 1075.
11. WHERE GROUPS OF VARYING BARS ARE BILLED FOR TWO FACES, ONE-HALF OF EACH LENGTHS BILLING IS PLACED IN EACH FACE.
△ 12. TO FACILITATE CONSTRUCTION CLASS 401.5A FW CONCRETE MAY BE SUBSTITUTED FOR CLASS 301.5A FW CONCRETE.



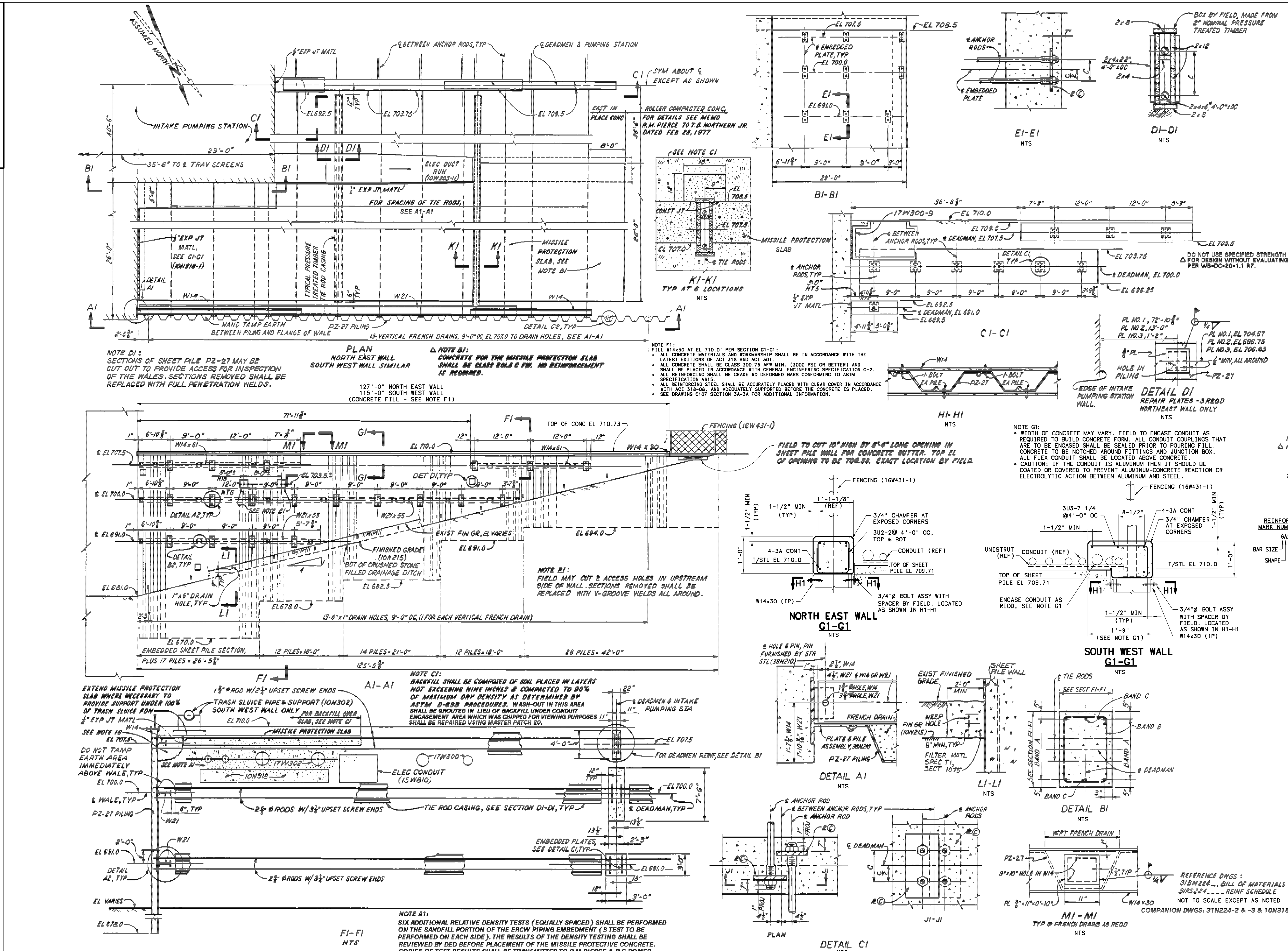
C3-C3 & D3-D3

D3-D3 OPP HAND

NOT TO SCALE
EXCEPT AS NOTED
COMPANION DWGS: 31N224-1 & 2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

WATER SUPPLY
UNITS 1 & 2
CONCRETE
RETAINING WALL AT INTAKE
PUMPING STA - OUTLINE & REINF.
TVA DWG NO. 31N224-3 RE
FIGURE 3.8.4-53



NOTES:

1. THE SHEET PILE RETAINING WALLS ARE CATEGORY I STRUCTURES AND QUALITY ASSURANCE IS REQUIRED.
2. QUALITY ASSURANCE FOR ALL MATERIAL FURNISHED BY FIELD, QUALITY LEVEL II IN ACCORDANCE WITH CONST SPEC N3G-881.
3. FOR GRADING PLAN SEE DRAWING 10N215.
4. BRACING OF PZ-27 SHEET PILING DURING CONSTRUCTION TO BE DETERMINED BY FIELD.
5. WALES MAY BE SPLICED AT QUARTER POINTS FROM ℄ OF TIE ROD IF DESIRED. ALL SPLICES SHALL BE DESIGNED TO DEVELOP THE FULL ALLOWABLE WEB SHEAR OF THE MEMBER.
6. 3½" AND 2¼" ϕ TIE RODS WITH ENDS NOT UPSET MAY BE RESPECTIVELY SUBSTITUTED FOR 2⅝" AND 1⅞" ϕ TIE RODS WITH UPSET ENDS. LARGER ROD WITH ENDS NOT UPSET MAY BE SPLICED TO SMALLER RODS IN ACCORDANCE WITH ROD SPLICE DETAIL.
7. TIE RODS SHALL NOT BE TIGHTENED UNTIL 7 DAYS AFTER CONCRETE DEADMAN HAS BEEN POURED.
8. A PROTECTIVE COATING CONSISTING OF COAL TAR EPOXY THAT MEETS ALL THE REQUIREMENTS OF CORPS OF ENGINEERS SPECIFICATION C-200 WILL BE PROVIDED FOR THE 1⅞" ϕ RODS ONLY.
9. ALL MATERIAL TO BE FURNISHED BY FIELD.
10. PLATES AND SHAPES SHALL CONFORM TO ASTM A36. BOLTS SHALL CONFORM TO ASTM A325.
11. FOR CONSTRUCTION AND BACKFILLING SEQUENCE SEE 31N224-2.
12. ALL WELDS TO BE MADE WITH AWS A5.1 E70 ELECTRODES.
13. ALL FIELD WELDING AND INSPECTIONS SHALL BE PERFORMED IN ACCORDANCE WITH TVA CONSTRUCTION SPECIFICATION G29.
14. ALL FIELD WELDS TO HAVE VISUAL INSPECTION.
15. FOR ADDITIONAL NOTES, SEE DRAWING 31N224-3.
16. FIELD TO PLACE FILL CONCRETE, CLASS 201.5 C FW, IN AREAS WHERE BACKFILL WAS WASHED OUT. FIELD TO EXTEND PRESSURE TREATED TIMBER BOX TO EDGE OF WALL & PLACE FILL CONC TO EL 708.5 ALONG ENTIRE LENGTH OF BOTH SHEET PILE WALLS.
17. LAP SPLICES FOR REINFORCING BARS USED IN W14 FILL SHALL BE A MINIMUM OF 24".

REINFORCING BAR MARK NUMBER LEGEND: 6A2-10 — BAR SIZE, SHAPE, TOTAL LENGTH IN FEET-INCHES

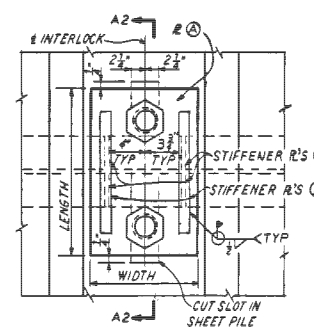
REINFORCEMENT SCHEDULE

BAR MARK NUMBER	NUMBER REQD	BENDING DIMENSIONS a	b	c
3U2-2	60	8"	9"	8"
3A	26	40'-0"		
3U3-7 1/4	30	9"	10 1/4"	2'-0*

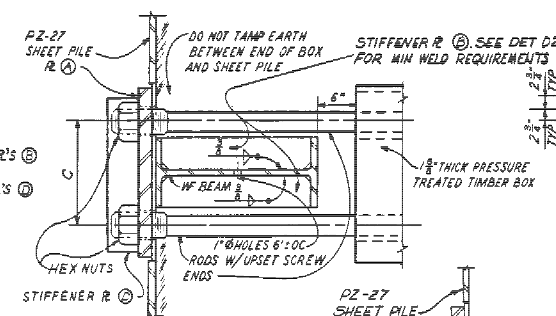
* FIELD CUT AS REQD

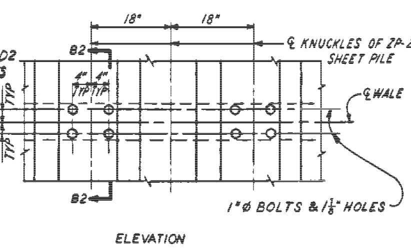
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

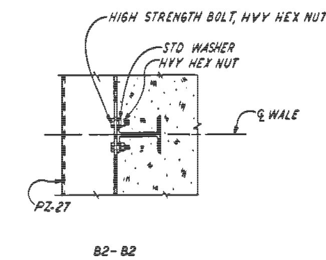
WATER SUPPLY
UNITS 1 & 2
CONCRETE
RETAINING WALL AT INTAKE PUMPING
STATION-OUTLINE & REINFORCEMENT
TVA DWG NO. 31N224-1 RJ
FIGURE 3.8.4-54



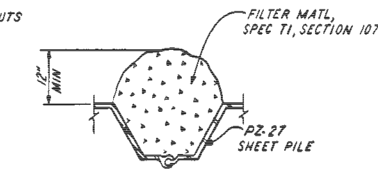
DETAIL A2
NOT TO SCALE



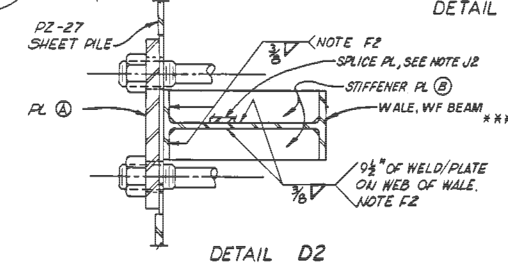




DETAIL B2



DETAIL C2
2G REQD
13 EA WALL

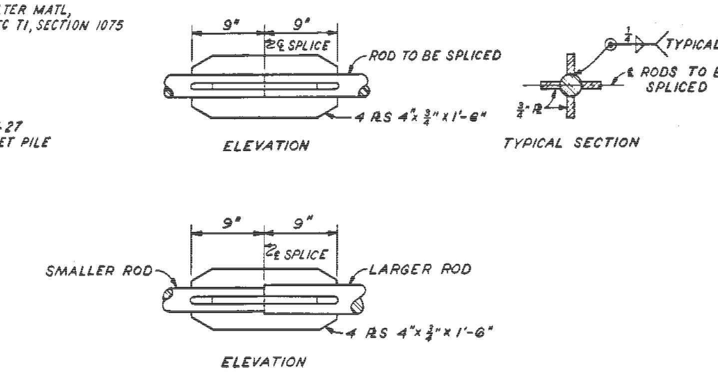


DETAIL D2
MIN WELDS REQD AT STIFFENER PLS Ⓑ ***
NTS

NOTE F2:
WELD MAY BE ON EITHER SIDE OF PL.

*** NOTE G2:
MINIMUM OR BETTER WELDS FOR STIFFENER PLATES "B" ARE REQUIRED AT ALL TIE ROD SUPPORT POINTS ON THE WALE @ EL 691.0 AND THE LAST SIX TIE ROD SUPPORT POINTS AWAY FROM THE PUMPING STATION AT THE EL 700.0 WALE. THERE ARE NO WELD REQUIREMENTS FOR STIFFENER PLS "B" AT THE REMAINING TIE ROD SUPPORT POINTS.

NOTE J2:
STIFFENER PLATE Ⓑ MAY BE NOTCHED TO CLEAR WIDE FLANGE WALE SPLICE PLATE.



ROD SPLICE DETAIL
NTS

PLATE SCHEDULE

ELEVATION	℞ A NO REQD	℞ A DIMENSIONS	℞ A HOLE DIA	℞ A DIM C	℞ A MK NO	℞ B # NO. REQD	℞ B # DIMENSIONS	℞ B # MK NO	℞ C NO. REQD	℞ C DIMENSIONS	℞ C HOLE DIA	℞ C DIM C	℞ C MK NO
707.5	12	11x⅞x 2'-0"	2½"	14½"	1	48	4¾ x ½" x 1'-0⅝"	3	12	9x⅞ x 1'-10"	2½"	14½"	5
700.0	18	11x 1¼ x 2'-6"	3½"	13½"	2	72	3⅞ x ½" x 1'-7¾"	4	18	12 x ⅞ x 2'-0"	3½"	13½"	6
691.0	8	11x 1¼ x 2'-6"	3½"	13½"	2	32	3⅞ x ½" x 1'-7¾"	4	8	12 x ⅞ x 2'-0"	3½"	13½"	6

\# NOTE C2:
STIFFENER ℞ B DIMENSION ARE EXACT. FIELD TO CHAMFER CORNERS TO FIT.

UPSET END TIE ROD SCHEDULE

ELEVATION	DIA	UPSET END DIA	NO. NUTS REQD	LENGTH OF THREADS AT EACH END OF ROD	SEE NOTE A2 LENGTH	SEE NOTE A2 NO. REQD	SEE NOTE B2 LENGTH	SEE NOTE B2 NO. REQD
707.5	1¾"	2¼"	48	6½"	28'-0¾"	12	66'-3¾"	12
700.0	2⅝"	3¼"	72	9½"	28'-3½"	12	66'-11"	24
691.0	2⅝"	3¼"	32	9½"	28'-3½"	12	67'-3½"	4

NOTE A2:
RODS ANCHORED TO PUMPING STATION WALL

NOTE B2:
RODS ANCHORED TO DEADMEN

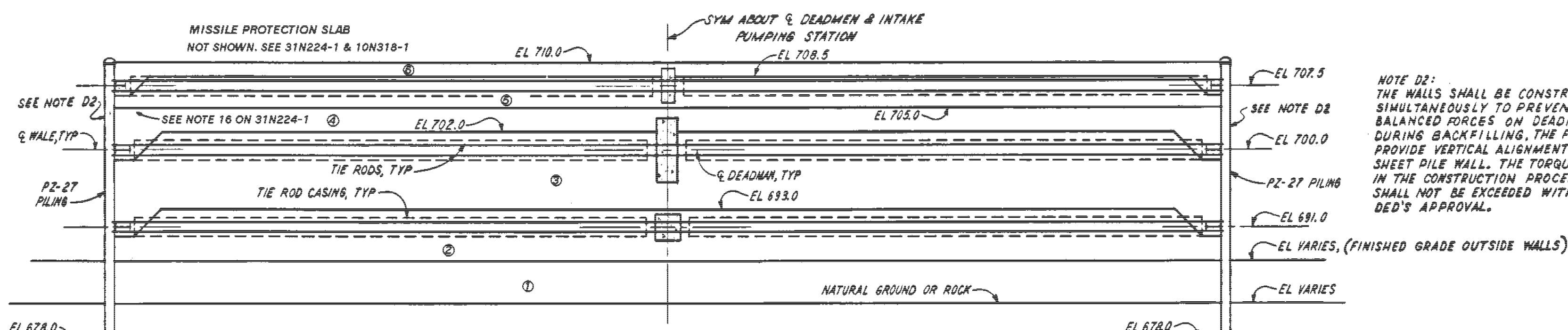
DEADMAN REINFORCEMENT

ELEVATION	BAND A	BAND B	BAND C
707.5	5-8A36-6,9"±OC	37-4T10-3,12"±OC	NONE REQD
700.0	7-8A53, 12"±OC	53-4T18-3,12"±OC	1-8A53
691.0	4-9A9-6,9"±OC	10-4T10-9,12"±OC	1-9A9-6

PLATE SCHEDULE, CONT.

ELEVATION	℞ D** NO. REQD	℞ D** DIMENSIONS	℞ D** MK NO.
707.5	24	¾ x 4¾ x 1'-10"	7
700.0	36	¾ x 4¾ x 2'-4"	8
691.0	16	¾ x 4¾ x 2'-4"	8

** NOTE E2:
FIELD MAY SUBSTITUTE A516-72 GRADE 70 STEEL FOR A36 STEEL FOR STIFFENER PL'S D.



NOTE D2:
THE WALLS SHALL BE CONSTRUCTED SIMULTANEOUSLY TO PREVENT UNBALANCED FORCES ON DEADMAN DURING BACKFILLING, THE FIELD SHALL PROVIDE VERTICAL ALIGNMENT OF THE SHEET PILE WALL. THE TORQUE VALUES IN THE CONSTRUCTION PROCEDURES SHALL NOT BE EXCEEDED WITHOUT DED'S APPROVAL.

NOTE:
FOR LIST OF NOTES AND REFERENCE DRAWINGS, SEE 31N224-1.

CONSTRUCTION PROCEDURE

1) BACKFILL AREA ①.
2) BACKFILL AREA ② DIG TRENCHES, INSTALL TIE RODS, BOXES AND POUR DEADMAN @ EL 691.0 AND TIGHTEN TIE RODS FINGER TIGHT.
3) BACKFILL AREA ③ TIGHTEN TIE RODS @ EL 691.0 TO 13 TONS IE 7300 IN.LBS TORQUE ON NUT. DIG TRENCHES INSTALL TIE RODS, BOXES AND POUR DEADMAN @ EL 700.0
4) BACKFILL AREA ④ TIGHTEN TIE RODS @ EL 700.0 TO 10 TONS IE 5600 IN.LBS TORQUE ON NUT. DIG TRENCH FOR ERCW SLAB, DRIVE PILING AND POUR SLAB.
5) BACKFILL ABOVE SLAB WITH SAND AND PLACE MECHANICAL PIPING.
6) BACKFILL AREA ⑤ DIG TRENCHES, INSTALL TIE RODS, BOXES AND POUR DEADMAN @ EL 707.5
7) BACKFILL AREA ⑥ TIGHTEN TIE RODS @ EL 707.5 TO 2 TONS IE 1000 IN LBS TORQUE ON NUT.

NOTE H2:
BACKFILL SHALL BE COMPOSED OF SOIL PLACED IN LAYERS NOT EXCEEDING SIX INCHES AND COMPACTED TO 95 PERCENT OF THE MAXIMUM DRY DENSITY AS DETERMINED BY ASTM D-698 PROCEDURES

NOTE K2:
IN LIEU OF MEETING THE REQUIREMENTS OF THE FIRST TWO SENTENCES OF THE FOURTH PARAGRAPH OF SECTION 11.2 OF GENERAL CONSTRUCTION SPECIFICATION G-9, THE FOLLOWING TESTING TO CONFIRM CORRELATION WITH THE LABORATORY COMPACTION CURVES SHALL BE MADE. AT LEAST ONE 3-POINT MOISTURE-DENSITY COMPACTION TEST SHALL BE PERFORMED FOR EACH 50 ROUTINE TESTS ON SOIL BEING USED FOR FILL, REGARDLESS OF SOIL TYPE.

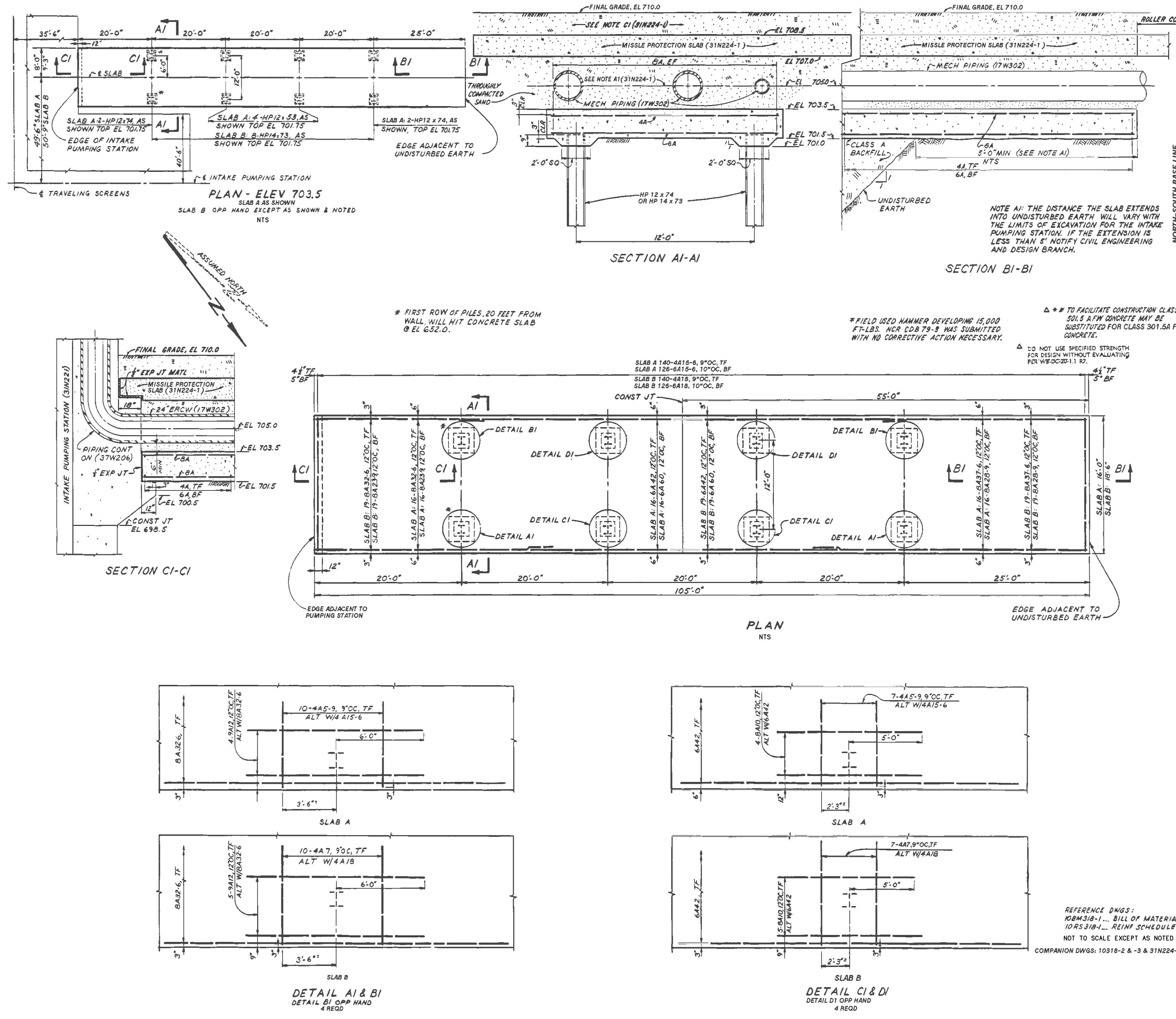
CONSTRUCTION & BACKFILLING PROCEDURE
NOT TO SCALE

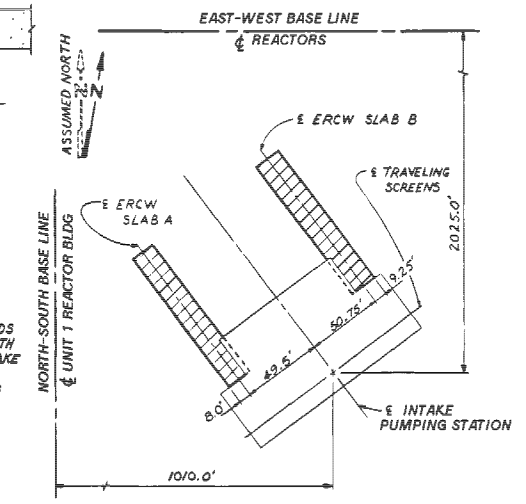
NOT TO SCALE
EXCEPT AS NOTED
COMPANION DWG: 31N224-1 & -3

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

WATER SUPPLY
UNITS 1 & 2
CONCRETE
RETAINING WALL AT INTAKE PUMPING
STATION-OUTLINE & REINFORCEMENT
TVA DWG NO. 31N224-2 RG
FIGURE 3.8.4-55

CAD MAINTAINED DRAWING



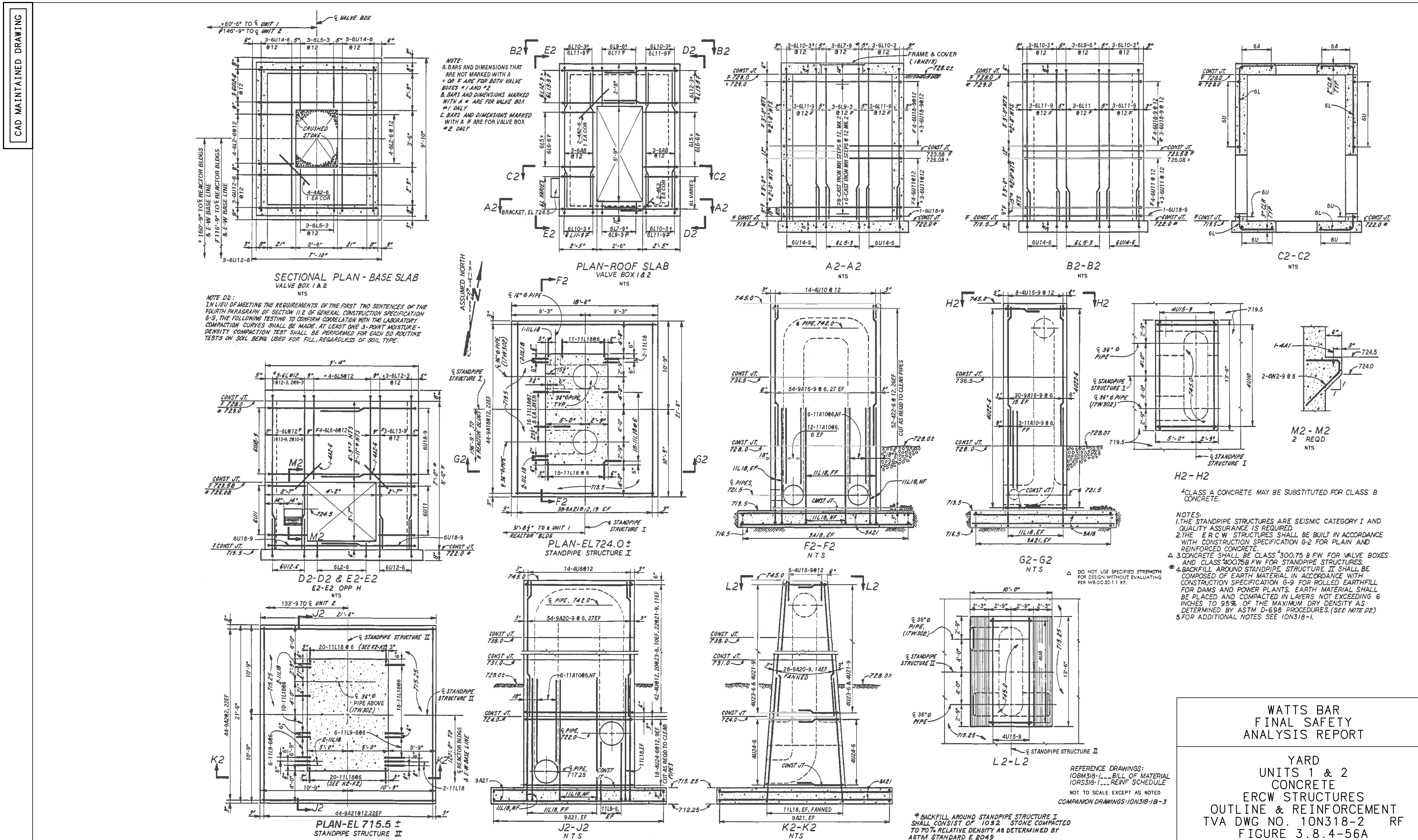


NOTES:

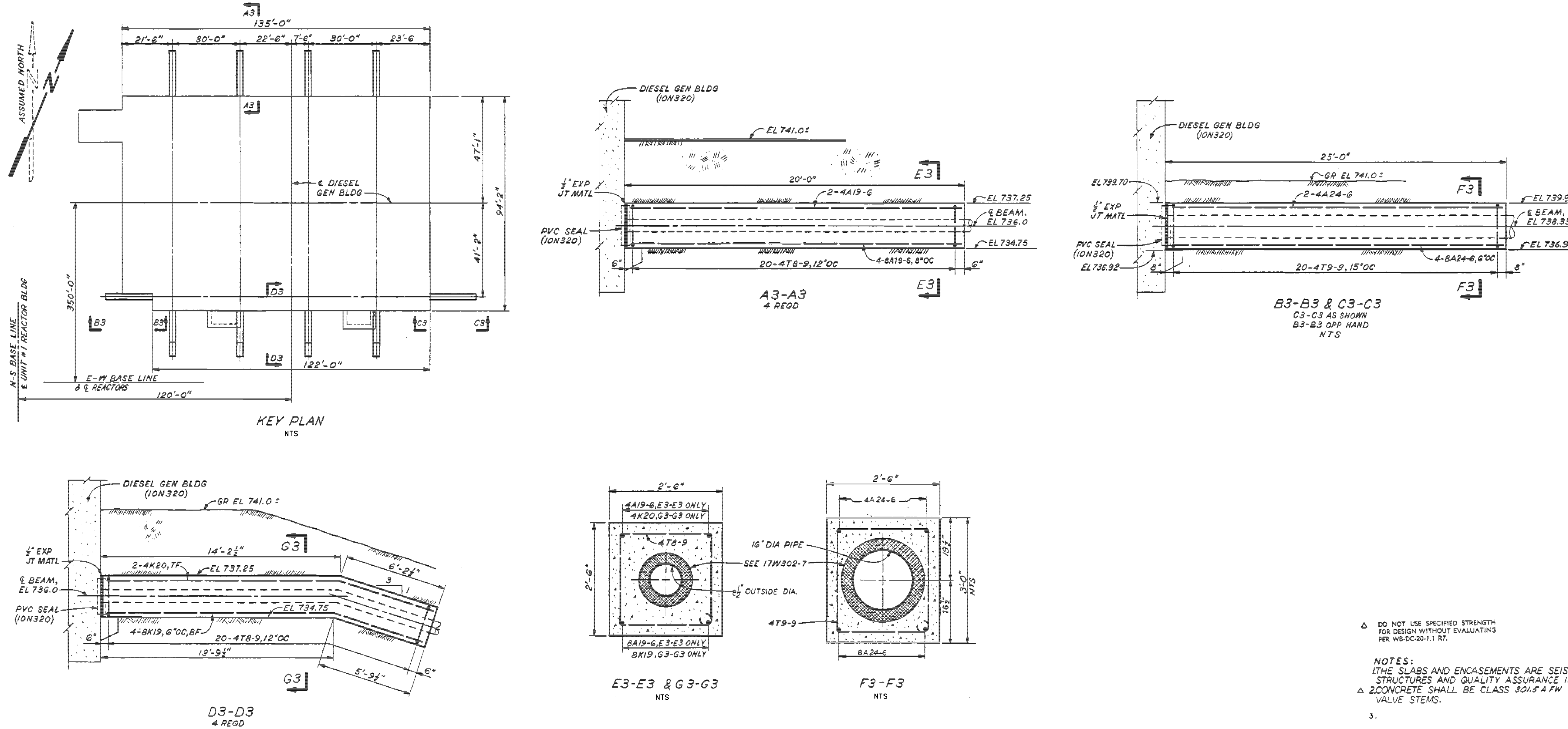
1. THE ERCW SLABS ARE CATEGORY I STRUCTURES AND QUALITY ASSURANCE IS REQUIRED.
2. QUALITY ASSURANCE FOR ALL MISCELLANEOUS STEEL: QUALITY LEVEL I IN ACCORDANCE WITH DEC-QCP-2.3
3. Δ CONCRETE SHALL BE CLASS *301.5A FW AND PLACED IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-2.
4. PILES SHALL BE DRIVEN BY AN APPROVED HAMMER DEVELOPING NOT LESS THAN *30,000 FOOT-POUNDS OF ENERGY PER BLOW.
5. PILES SHALL BE DRIVEN TO REFUSAL IN SOUND ROCK OR CONCRETE. PILES DRIVEN TO ROCK SHALL BE ASSUMED DRIVEN TO REFUSAL WHEN THE LAST SIX INCHES SHALL AVERAGE FORTY BLOWS PER INCH AND THE FINAL INCH OF PENETRATION SHALL BE A MINIMUM OF FORTY BLOWS. PILES DRIVEN TO CONCRETE SHALL BE ASSUMED DRIVEN TO REFUSAL WHEN THE FINAL 1/4 INCH OF PENETRATION REQUIRES A MINIMUM OF TEN BLOWS
6. THE DEVIATION TOLERANCE IN LOCATION OF ANY PILE SHALL NOT EXCEED 6 INCHES
7. WHERE SPLICES ARE REQUIRED THEY SHALL MAINTAIN THE TRUE ALIGNMENT AND POSITION OF THE PILE SECTION. SPLICES SHALL BE FULLY BUTT-WELDED WITH POSITIONING PLATES WELDED TO THE FLANGES AND WEB OF THE PILE.
8. PILES SHALL BE ORIENTED AS SHOWN IN PLAN.
9. ALL WELDS SHALL BE MADE WITH AWS A5.1, E70 SERIES ELECTRODES.
10. ALL FIELD WELDING AND INSPECTIONS SHALL BE PERFORMED IN ACCORDANCE WITH TVA CONSTRUCTION SPECIFICATION G-29.
11. ALL REINFORCEMENT SHALL CONFORM TO ASTM SPECIFICATION A615, GRADE 60, WITH ADDITIONAL TESTING IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-2.
12. WELDING OF OR TO REINFORCING BARS IS PROHIBITED (EXCEPT WHERE SPECIFIED ON THE DRAWINGS), WITHOUT PRIOR APPROVAL OF THE CHIEF, CIVIL ENGINEERING AND DESIGN BRANCH.
13. A COMPLETE PILE DRIVING RECORD SHALL BE MADE INCLUDING PILE NUMBER AND OR LOCATION, PILE SIZE, CUTOFF AND TIP ELEVATIONS, TYPE OF HAMMER, DATE DRIVEN, BLOWS PER FOOT OF PENETRATION, BLOWS AT REFUSAL AND DEVIATION(S) OF TOP OF PILE. A COPY OF THE RECORD SHALL BE SENT TO THE CHIEF, CIVIL ENGINEERING AND DESIGN BRANCH.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
ERCW STRUCTURES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10N318-1 RD
FIGURE 3.8.4-56


CAD MAINTAINED DRAWING
SECTIONAL PLAN - BASE SLAB
VALVE BOX 1 & 2
NTS
PLAN-ROOF SLAB
VALVE BOX 1 & 2
NTS
NOTE:
A. BARS AND DIMENSIONS THAT ARE NOT MARKED WITH A * OR ∓ ARE FOR BOTH VALVE BOXES #1 AND #2
B. BARS AND DIMENSIONS MARKED WITH A * ARE FOR VALVE BOX #1 ONLY
C. BARS AND DIMENSIONS MARKED WITH A ∓ ARE FOR VALVE BOX #2 ONLY
A2-A2
NTS
B2-B2
NTS
C2-C2
NTS
NOTE D2:
IN LIEU OF MEETING THE REQUIREMENTS OF THE FIRST TWO SENTENCES OF THE FOURTH PARAGRAPH OF SECTION 11.2 OF GENERAL CONSTRUCTION SPECIFICATION G-9, THE FOLLOWING TESTING TO CONFIRM CORRELATION WITH THE LABORATORY COMPACTION CURVES SHALL BE MADE. AT LEAST ONE 3-POINT MOISTURE-DENSITY COMPACTION TEST SHALL BE PERFORMED FOR EACH 50 ROUTINE TESTS ON SOIL BEING USED FOR FILL, REGARDLESS OF SOIL TYPE.
ASSUMED NORTH
D2-D2 & E2-E2
E2-E2 OPP H
NTS
PLAN-EL 724.0 ±
STANDPIPE STRUCTURE I
F2-F2
NTS
G2-G2
NTS
H2-H2
M2-M2
2 REQD
NTS
∆CLASS A CONCRETE MAY BE SUBSTITUTED FOR CLASS B CONCRETE.
NOTES:
1. THE STANDPIPE STRUCTURES ARE SEISMIC CATEGORY I AND QUALITY ASSURANCE IS REQUIRED.
2. THE ERCW STRUCTURES SHALL BE BUILT IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-2 FOR PLAIN AND REINFORCED CONCRETE.
∆ 3. CONCRETE SHALL BE CLASS ∆300.75 B FW FOR VALVE BOXES AND CLASS ∆400.75B FW FOR STANDPIPE STRUCTURES.
* 4. BACKFILL AROUND STANDPIPE STRUCTURE II SHALL BE COMPOSED OF EARTH MATERIAL IN ACCORDANCE WITH CONSTRUCTION SPECIFICATION G-9 FOR ROLLED EARTHFILL FOR DAMS AND POWER PLANTS. EARTH MATERIAL SHALL BE PLACED AND COMPACTED IN LAYERS NOT EXCEEDING 6 INCHES TO 95% OF THE MAXIMUM DRY DENSITY AS DETERMINED BY ASTM D-698 PROCEDURES. (SEE NOTE D2)
5. FOR ADDITIONAL NOTES SEE 10N318-1.
∆ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.
PLAN-EL 715.5 ±
STANDPIPE STRUCTURE II
J2-J2
NTS
K2-K2
NTS
L2-L2
REFERENCE DRAWINGS:
10BM318-1___BILL OF MATERIAL
10RS318-1___REINF SCHEDULE
NOT TO SCALE EXCEPT AS NOTED
COMPANION DRAWINGS: 10N318-1 & -3
* BACKFILL AROUND STANDPIPE STRUCTURE I SHALL CONSIST OF 10S2 STONE COMPACTED TO 70% RELATIVE DENSITY AS DETERMINED BY ASTM STANDARD E 2049
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT
YARD
UNITS 1 & 2
CONCRETE
ERCW STRUCTURES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10N318-2 RF
FIGURE 3.8.4-56A

CAD MAINTAINED DRAWING



△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

NOTES:
1. THE SLABS AND ENCASEMENTS ARE SEISMIC CATEGORY I STRUCTURES AND QUALITY ASSURANCE IS REQUIRED.
△ 2. CONCRETE SHALL BE CLASS 301.5 A FW FOR SLABS @ ERCW VALVE STEMS.
3.
4. FOR ADDITIONAL NOTES SEE 10N318-1.

REFERENCE DRAWINGS:
10BM318-3 ---- BILL OF MATERIALS
10RS318-3 ---- REINF SCHEDULE

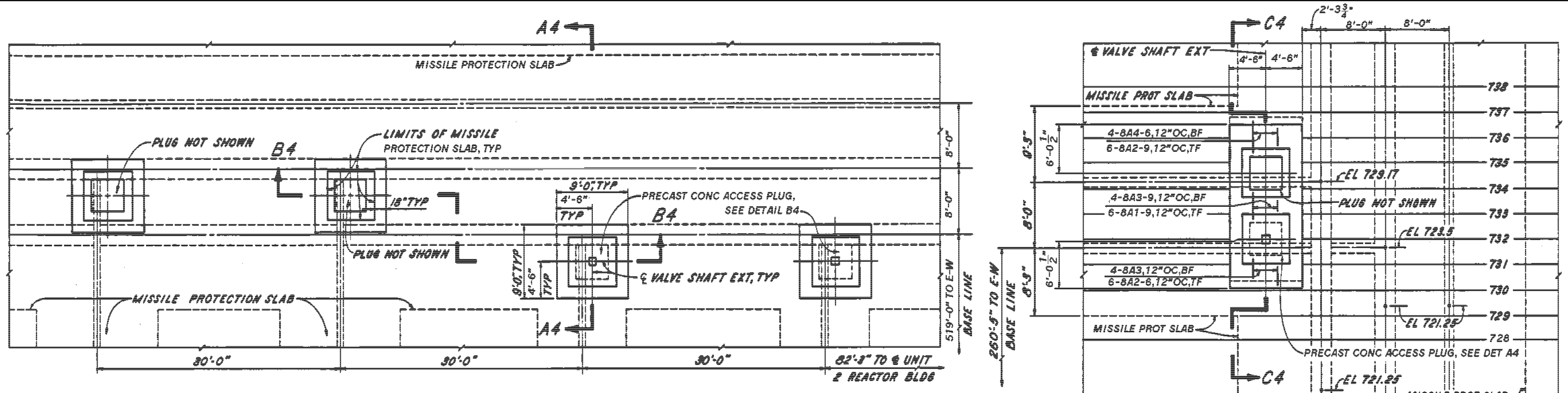
NOT TO SCALE
EXCEPT AS NOTED

COMP DWGS: 10N318-1 & 2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

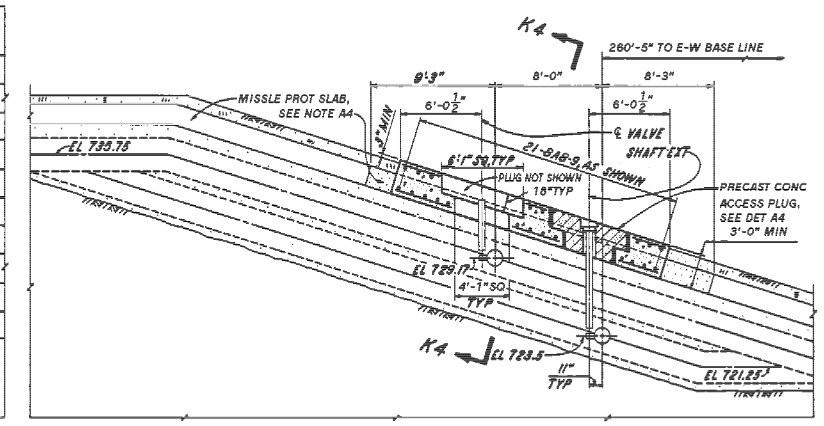
YARD
UNITS 1 & 2
CONCRETE
ERCW STRUCTURES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10N318-3 RD
FIGURE 3.8.4-56B

CAD MAINTAINED DRAWING

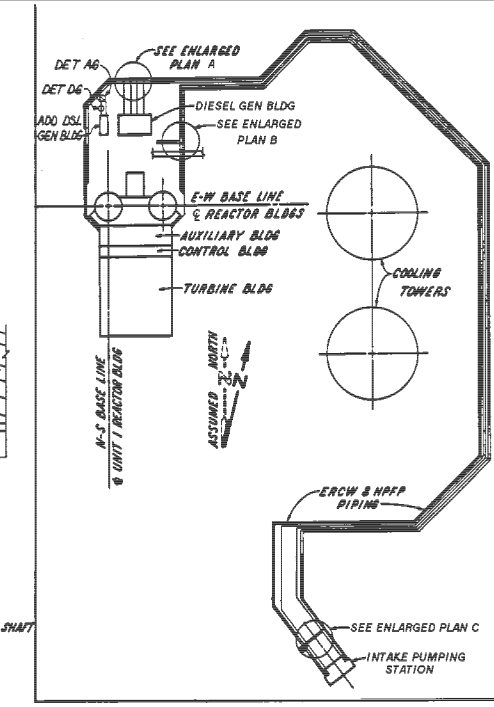


ENLARGED PLAN A

ENLARGED PLAN B



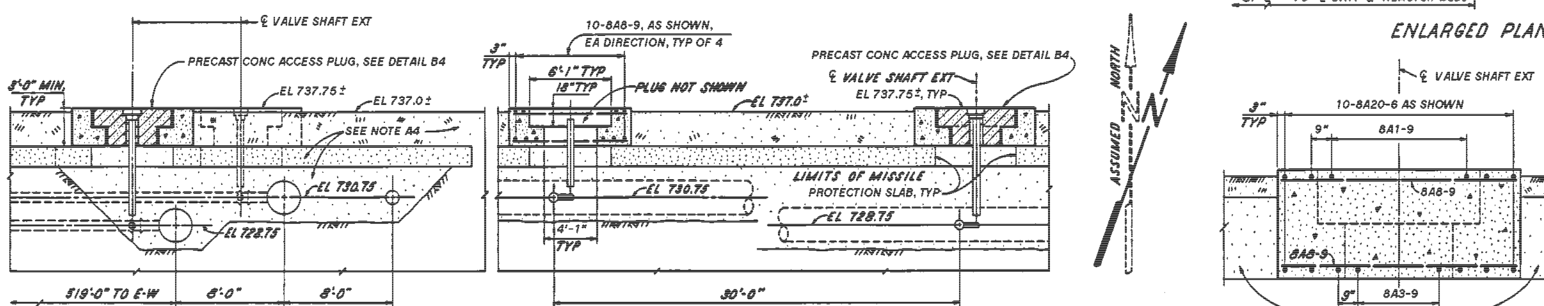
C4-C4
NTS



KEY PLAN
NTS

NOTES:
1. THE SLABS AND ENCASEMENTS ARE SEISMIC CATEGORY I STRUCTURES.. QUALITY LEVEL II IS REQUIRED IN ACCORDANCE WITH CONST SPEC N3G-881.
△ 2. CONCRETE SHALL BE CLASS 301.5 A FW.
3. FOR ADDITIONAL NOTES SEE 10N318-1.
4. FOR EXISTING VALVE PITS, CONCRETE ACCESS PLUGS SHALL BE SIZED TO FIT THE EXISTING OPENING. TOLERANCES FOR DIMENSIONS ON ACCESS PLUGS ARE AS FOLLOWS:

LENGTH AND WIDTH: -1", +6"
THICKNESS: -1", +4"
LOCATION OF VALVE SHAFT OPENING: 12" FROM ℄ OF PLUG
GAP BETWEEN PLUG AND OPENING: 1/2" ± 1/4"

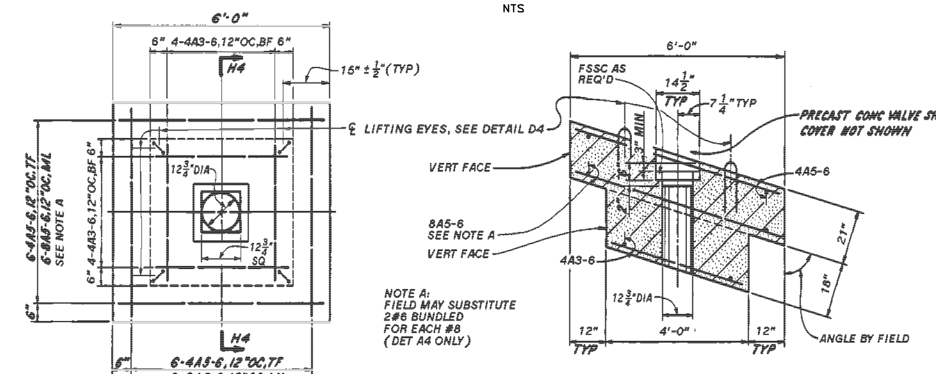


SECT A4-A4
NTS

NOTE A4:
FOR DETAILS OF MISSILE PROTECTION SLAB & BACKFILL, SEE 17W302.

B4-B4
NTS

K4-K4
NTS

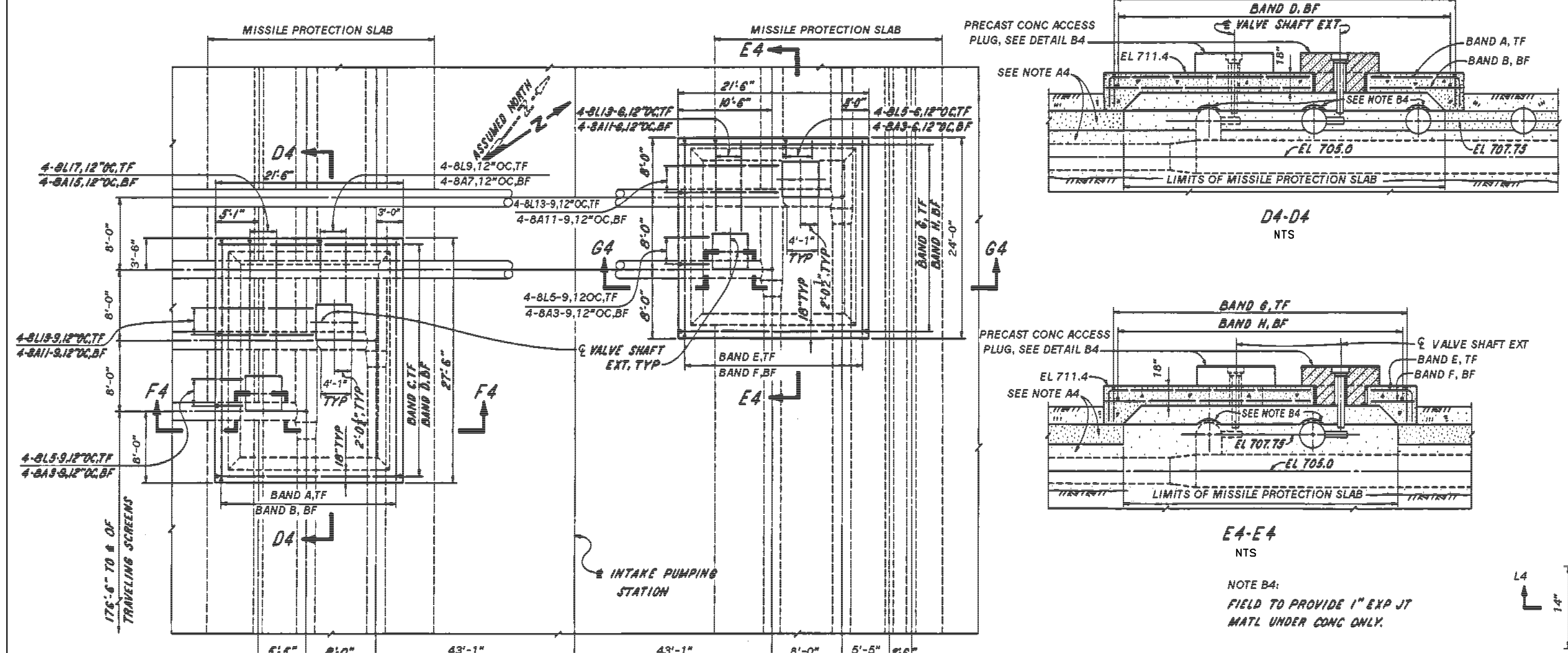


DET A4
PRECAST CONC ACCESS PLUG
2 REQD
NTS

H4-H4

△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

REFERENCE DWG:
10RS318-4 ________ REINF SCHEDULE

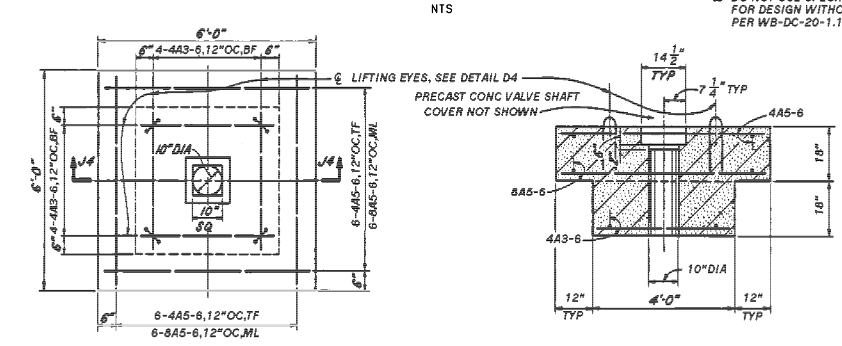


ENLARGED PLAN C

D4-D4
NTS

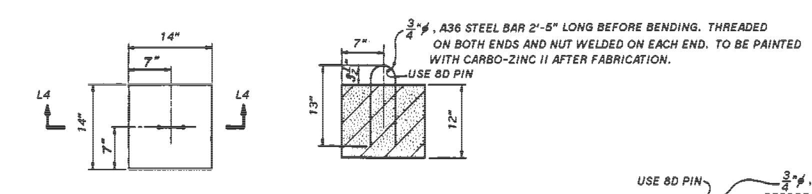
E4-E4
NTS

NOTE B4:
FIELD TO PROVIDE 1" EXP JT MATL UNDER CONC ONLY.

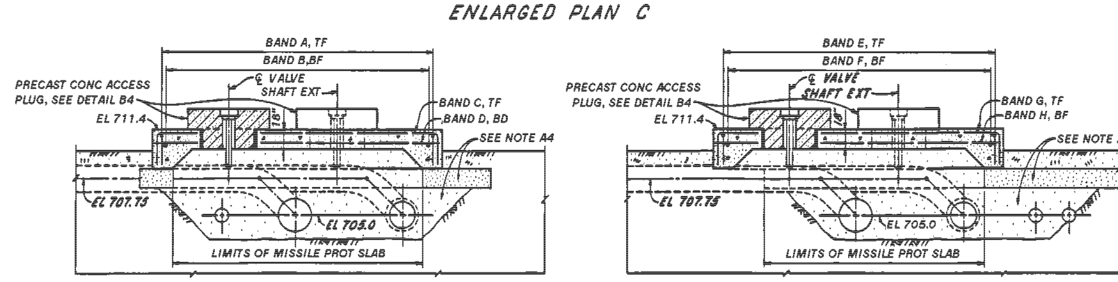


DET B4
PRECAST CONC ACCESS PLUG
8 REQD
NTS

J4-J4



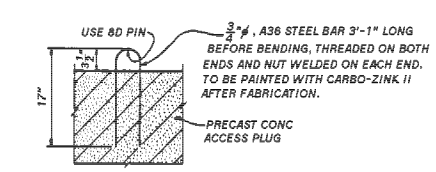
DET C4
PRECAST CONC VALVE SHAFT COVER
10 REQD
NTS



F4-F4
NTS

G4-G4
NTS

REINF BANDS FOR ENLARGED PLAN C	
BAND A	22 BARS, 12"OC, TF, 4-8U31-3, 4-8L9-9, 4-8U31-3, 4-8L17-9, 6-8U31-3
BAND B	22-8A, 12"OC, BF, 4@27, 4@7-9, 4@27, 4@15-9, 6@27
BAND C	28 BARS, 12"OC, TF, 8-8U25-3, 4-8L15, 4-8U25-3, 4-8L7, 8-8U25-3
BAND D	28-8A, 12"OC, BF, 8@21, 4@13, 4@21, 4@5, 8@21
BAND E	22 BARS, 12"OC, TF, 4-8U27-9, 4-8L9-9, 4-8U27-9, 4-8L17-9, 6-8U27-9
BAND F	22-8A, 12"OC, BF, 4@23-6, 4@7-9, 4@23-6, 4@15-9, 6@23-6
BAND G	24 BARS, 12"OC, TF, 8-8U25-3, 4-8L15, 4-8U25-3, 4-8L7, 4-8U25-3
BAND H	24-8A, 12"OC, BF, 8@21, 4@13, 4@21, 4@5, 4@21



DET D4
LIFTING EYES
40 REQD
NTS

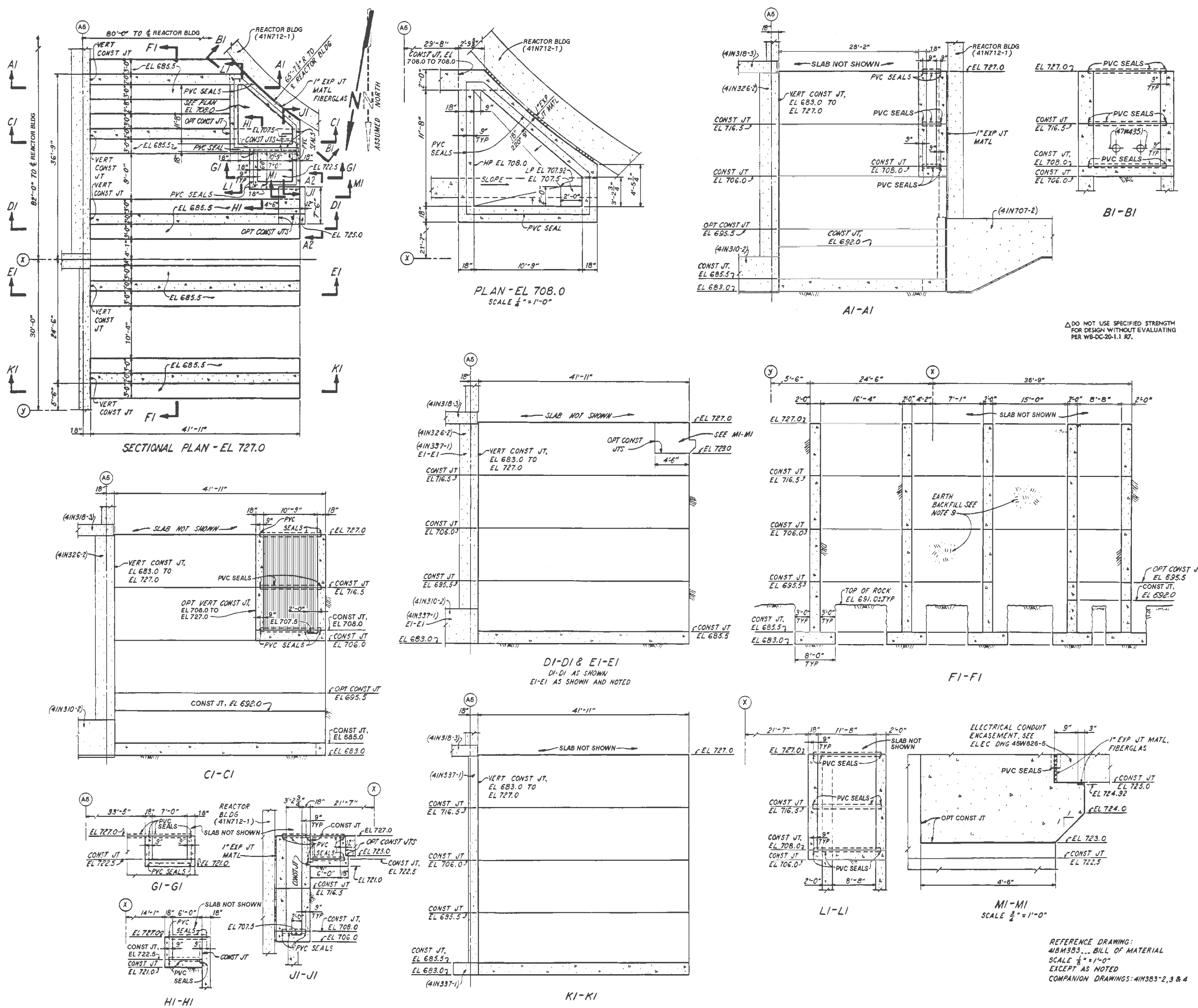
NOT TO SCALE EXCEPT AS NOTED

COMP DWGS: 10N318-1, -2, -3, -5 & -6

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

YARD
UNITS 1 & 2
CONCRETE
ERCW STRUCTURES
OUTLINE & REINFORCEMENT
TVA DWG NO. 10W318-4 RE
FIGURE 3.8.4-56C

CAD MAINTAINED DRAWING



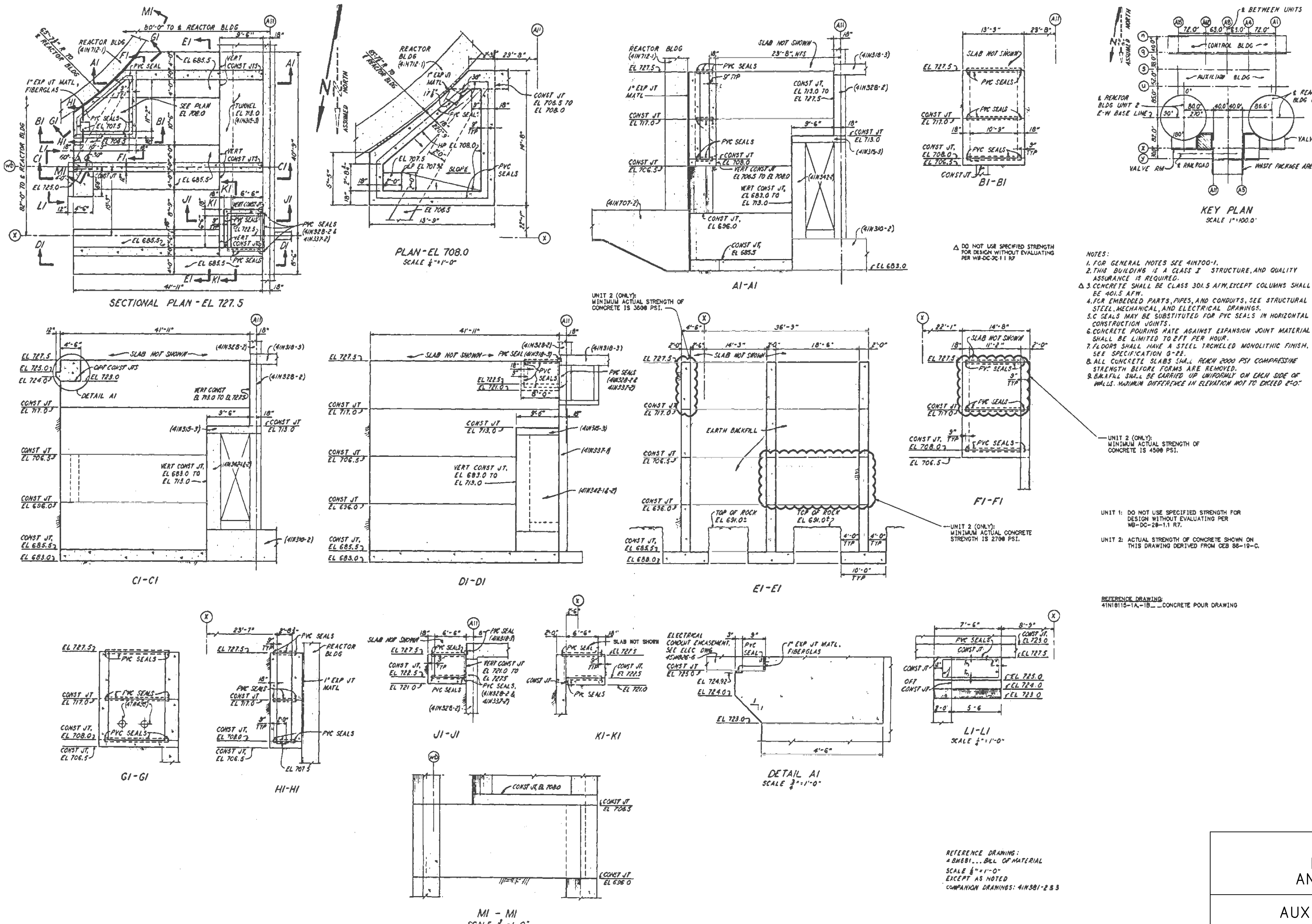
NOTES:
1. THIS BUILDING IS A CLASS I STRUCTURE AND QUALITY ASSURANCE IS REQUIRED.
2. FOR GENERAL NOTES, SEE 41N700-1.
3. ALL CONCRETE SHALL BE CLASS 301.5 AFW EXCEPT COLUMNS. COLUMNS SHALL BE CLASS 401.5 AFW.
4. POURING RATE OF CONCRETE AGAINST EXPANSION JOINT MATERIAL SHALL BE LIMITED TO 2 FT PER HOUR.
5. FOR EMBEDDED PARTS, PIPES AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL AND ELECTRICAL DRAWINGS.
6. FLOORS SHALL HAVE A STEEL TROWELED MONOLITHIC FINISH, SEE SPECIFICATION G22.
7. ALL CONCRETE SLABS SHALL REACH 2000 PSI COMPRESSIVE STRENGTH BEFORE FORMS ARE REMOVED. IN ADDITION, SHORING SHALL BE LEFT UNDER THE EL 763.5 SLAB UNTIL ICE BIN IS PLACED AND EL 775.25 SLAB HAS REACHED 2000 PSI.
8. C SEALS MAY BE SUBSTITUTED FOR PVC SEALS IN HORIZONTAL CONSTRUCTION JOINTS.
9. BACKFILL SHALL BE CARRIED UP UNIFORMLY ON EACH SIDE OF WALLS. MAXIMUM DIFFERENCE IN ELEVATION NOT TO EXCEED 2'. FOR TYPE OF BACKFILL SEE 10N210.
10. METHODOLOGY CALCULATION WC6-1-585 AND THE WORST CASE SELECTION CALCULATION WC6-1-744 WERE UTILIZED IN ENGINEERING EVALUATIONS OF THE CONCRETE FEATURES SHOWN IN THIS DRAWING SERIES.

△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

REFERENCE DRAWING:
41BM383... BILL OF MATERIAL
SCALE 1/4" = 1'-0"
EXCEPT AS NOTED
COMPANION DRAWINGS: 41N383-2, 3 & 4

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
CONCRETE
ADDITIONAL EQUIPMENT BUILDING
OUTLINE
TVA DWG NO. 41N383-1 RC
FIGURE 3.8.4-57



NOTES:

1. FOR GENERAL NOTES SEE 41N700-1.
2. THIS BUILDING IS A CLASS I STRUCTURE, AND QUALITY ASSURANCE IS REQUIRED.
3. CONCRETE SHALL BE CLASS 301.5 AFW, EXCEPT COLUMNS SHALL BE 401.5 AFW.
4. FOR EMBEDDED PARTS, PIPES, AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.
5. C SEALS MAY BE SUBSTITUTED FOR PVC SEALS IN HORIZONTAL CONSTRUCTION JOINTS.
6. CONCRETE POURING RATE AGAINST EXPANSION JOINT MATERIAL SHALL BE LIMITED TO 2 FT PER HOUR.
7. FLOORS SHALL HAVE A STEEL TROWELED MONOLITHIC FINISH. SEE SPECIFICATION G-22.
8. ALL CONCRETE SLABS SHALL REACH 2000 PSI COMPRESSIVE STRENGTH BEFORE FORMS ARE REMOVED.
9. BACKFILL SHALL BE CARRIED UP UNIFORMLY ON EACH SIDE OF WALLS. MAXIMUM DIFFERENCE IN ELEVATION NOT TO EXCEED 2'-0".

UNIT 1: DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

UNIT 2: ACTUAL STRENGTH OF CONCRETE SHOWN ON THIS DRAWING DERIVED FROM CEB 86-19-C.

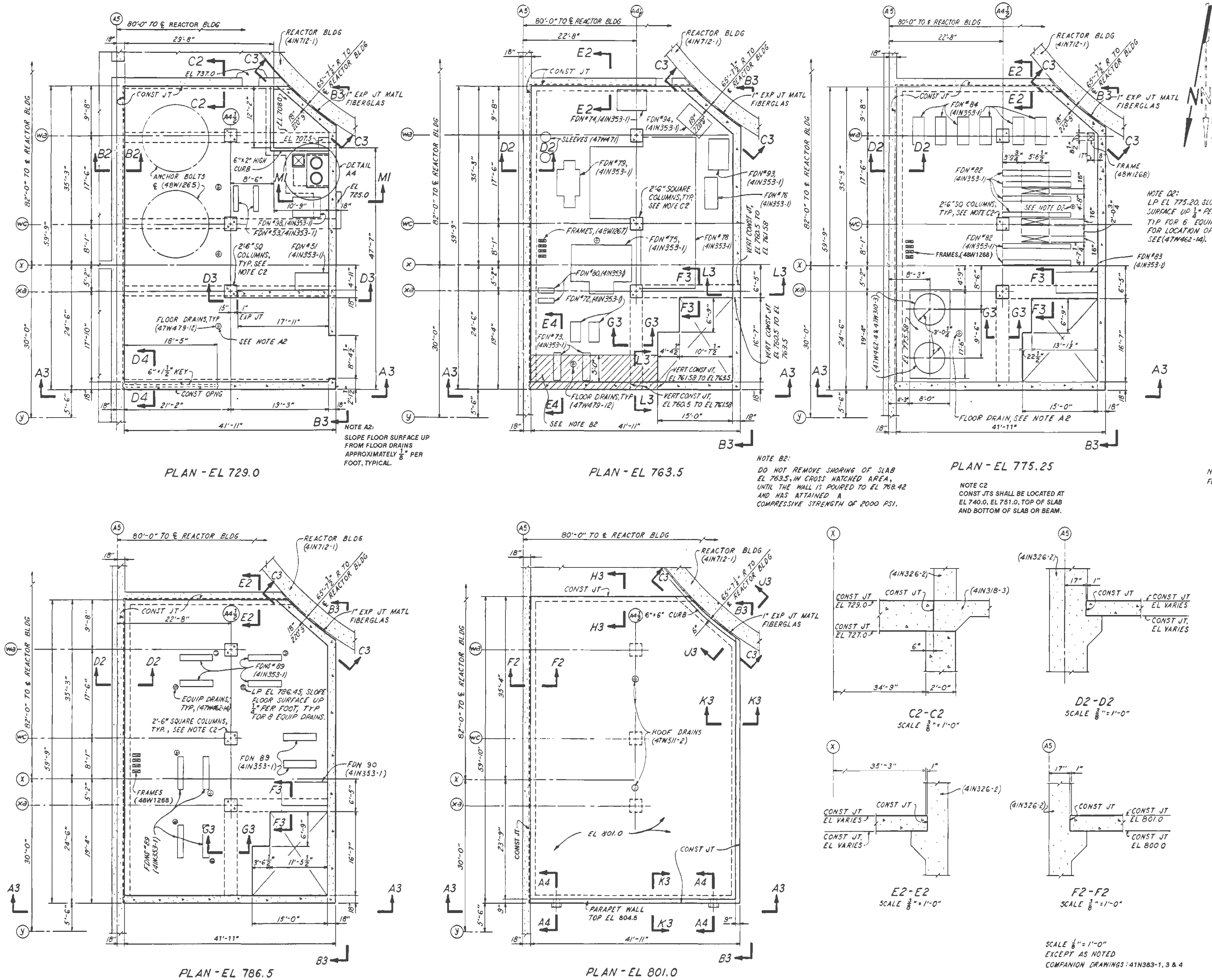
REFERENCE DRAWING:
41N18115-1A,-1B ... CONCRETE POUR DRAWING

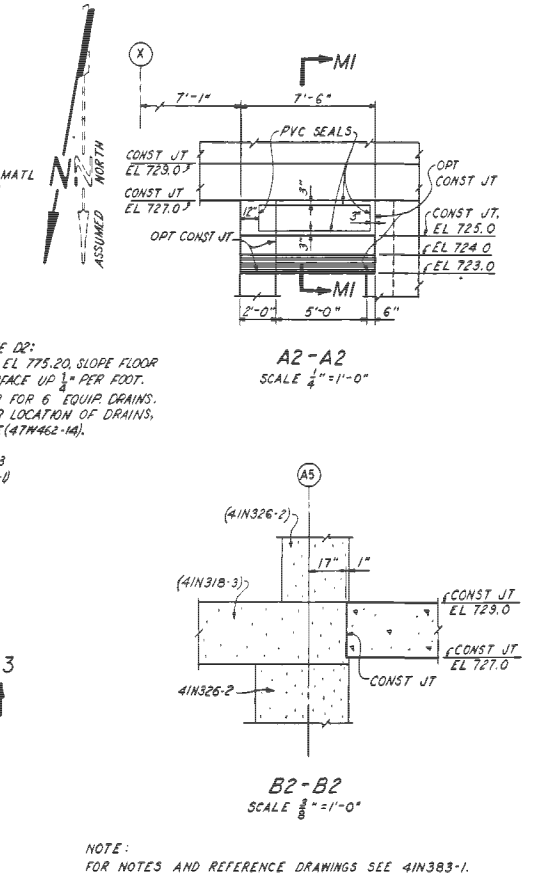
REFERENCE DRAWING:
4BH881 ... BILL OF MATERIAL
SCALE 1/4"=1'-0"
EXCEPT AS NOTED
COMPANION DRAWINGS: 41N381-2 & 3

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
CONCRETE
ADDITIONAL EQUIPMENT BUILDING
OUTLINE
TVA DWG NO. 41N381-1 R3
FIGURE 3.8.4-57(U2)

CAD MAINTAINED DRAWING

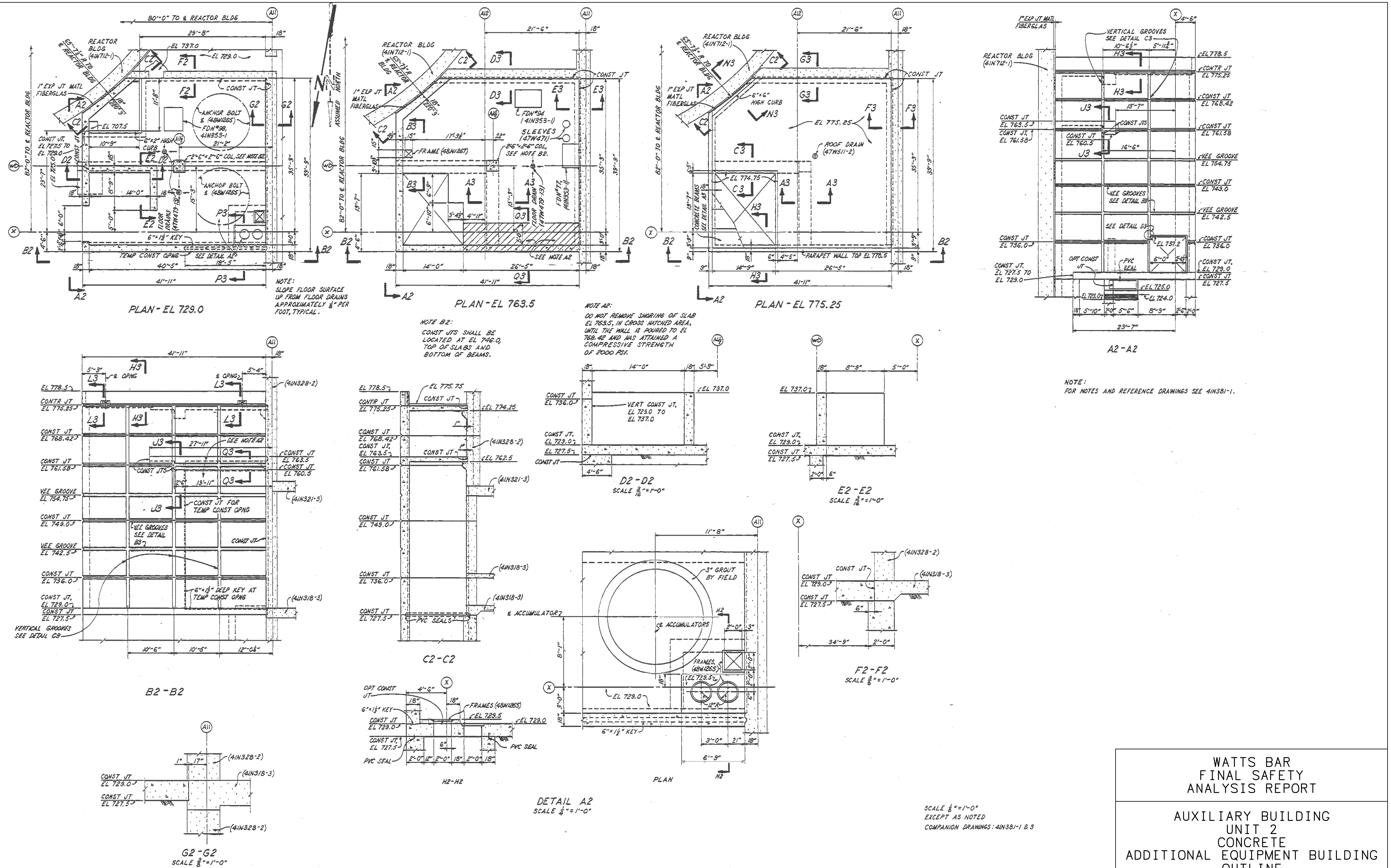




WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
CONCRETE
ADDITIONAL EQUIPMENT BUILDING
OUTLINE
TVA DWG NO. 41N383-2 RB
FIGURE 3.8.4-58

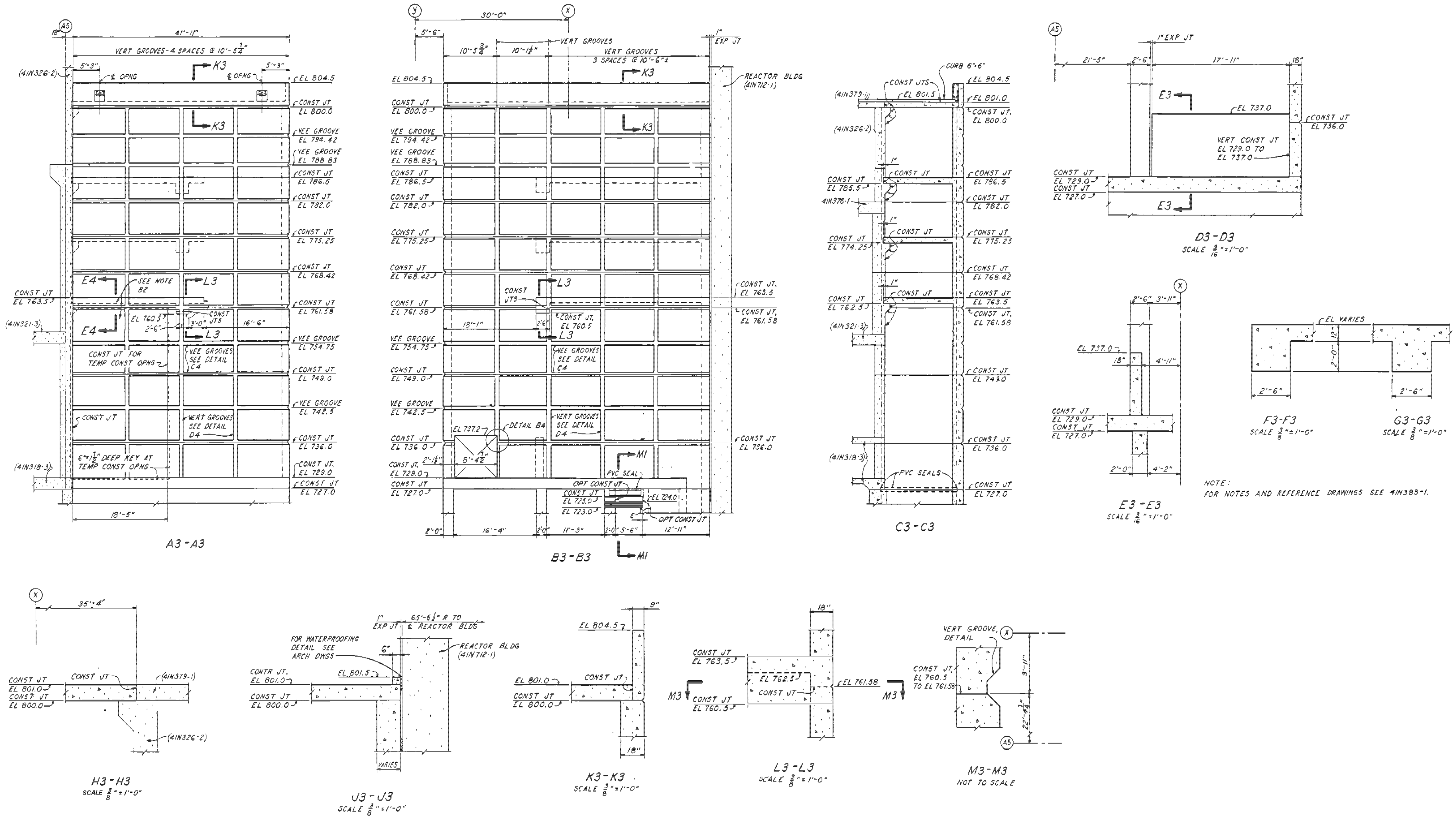
CAD MAINTAINED DRAWING



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
CONCRETE
ADDITIONAL EQUIPMENT BUILDING
OUTLINE
TVA DWG NO. 41N381-2 R2
FIGURE 3.8.4-58(U2)

CAD MAINTAINED DRAWING

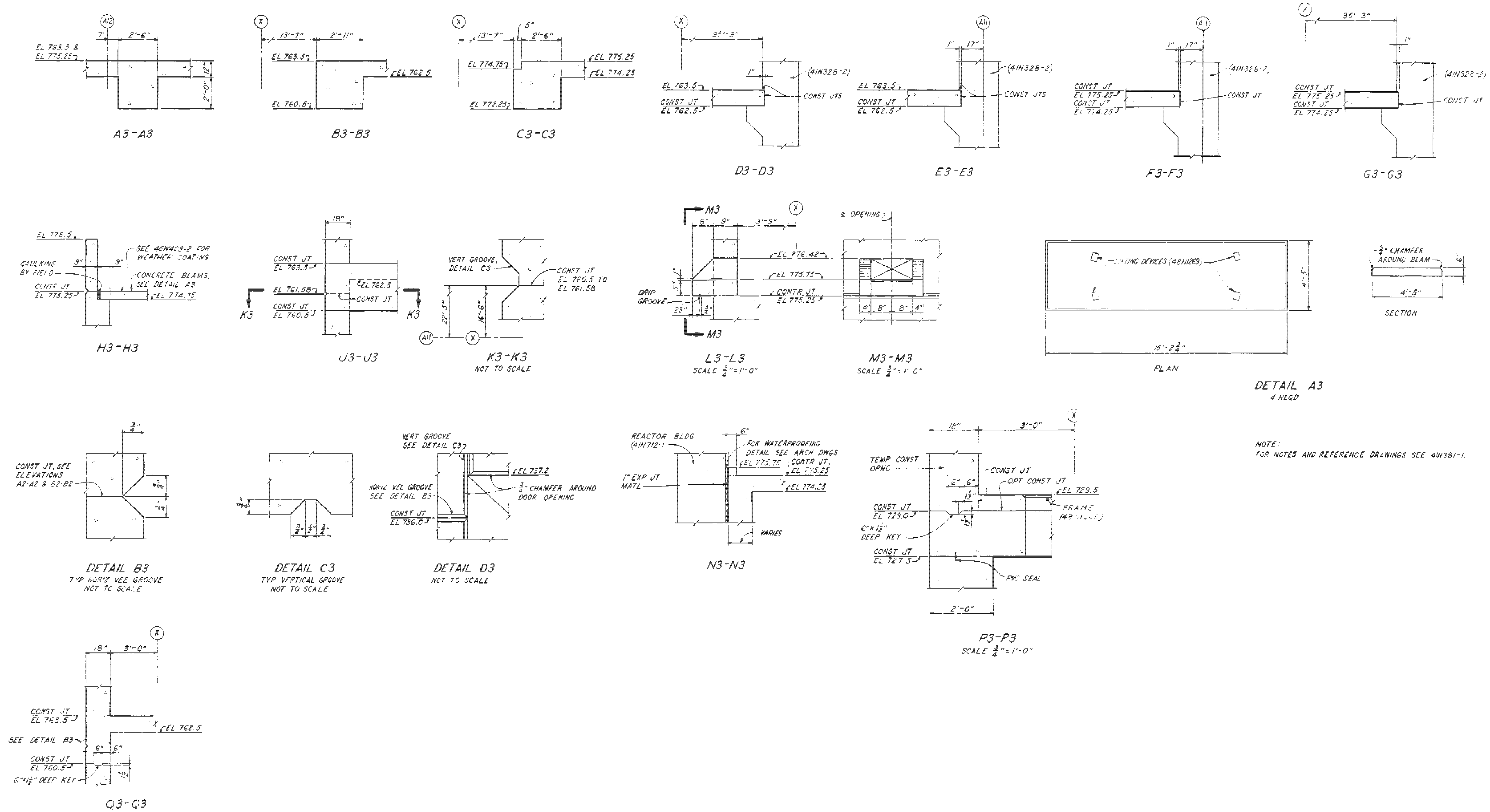


SCALE 1/8"=1'-0"
EXCEPT AS NOTED
COMPANION DRAWINGS: 41N383-1,2 & 4

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 1
CONCRETE
ADDITIONAL EQUIPMENT BUILDING
OUTLINE
TVA DWG NO. 41N383-3 RB
FIGURE 3.8.4-59

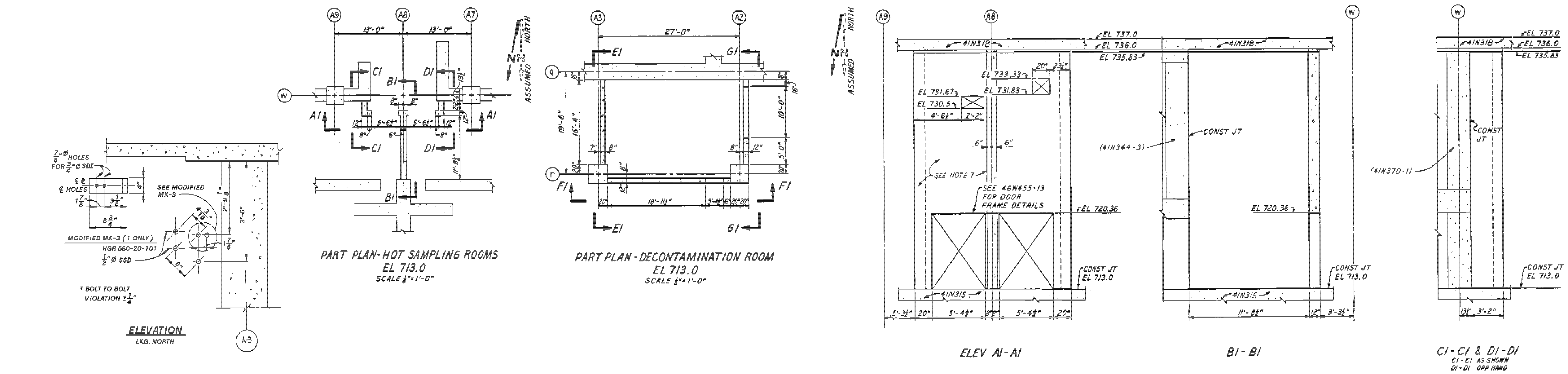
CAD MAINTAINED DRAWING

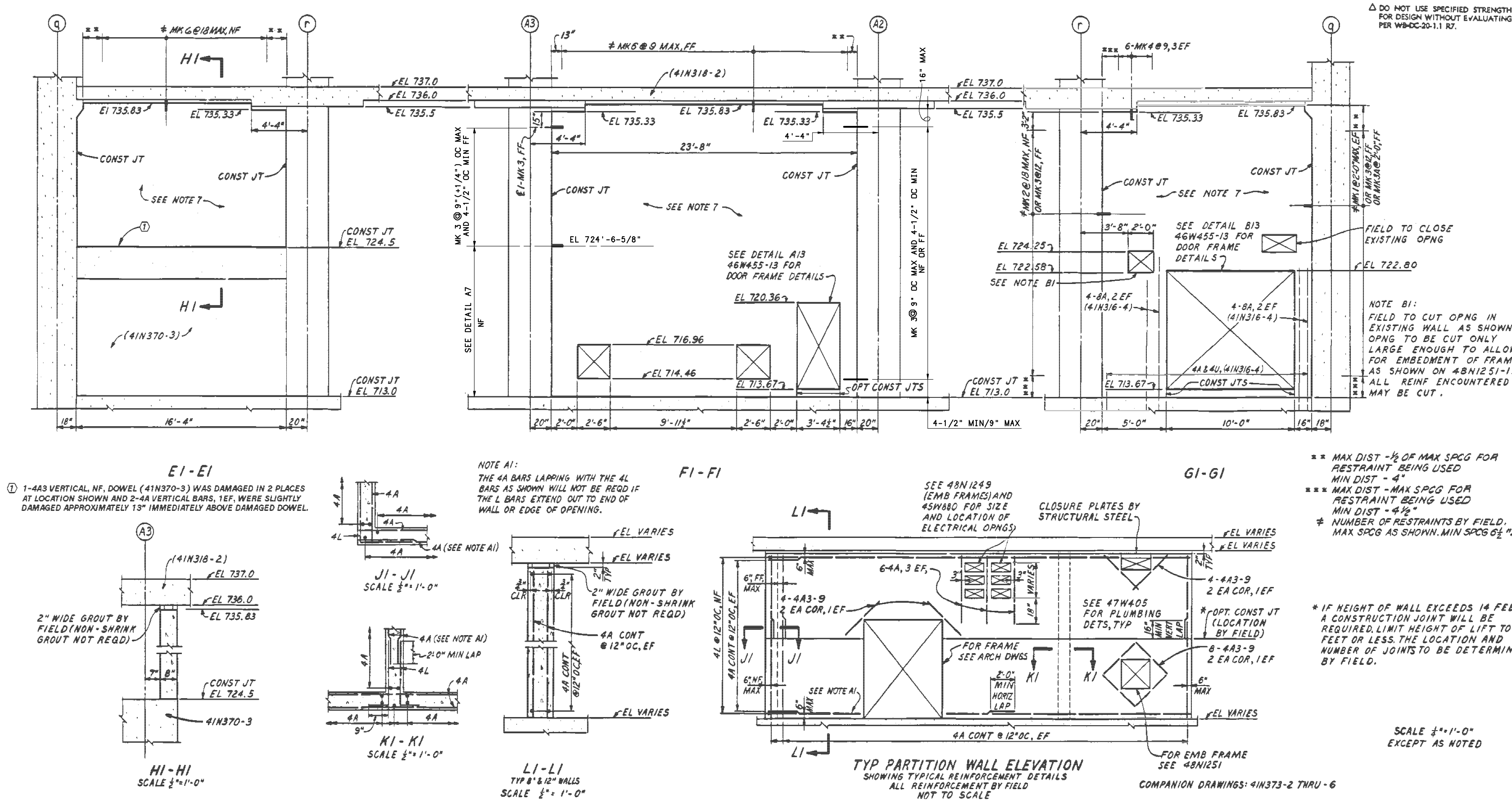


WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
CONCRETE
ADDITIONAL EQUIPMENT BUILDING
OUTLINE
TVA DWG NO. 41N381-3 R1
FIGURE 3.8.4-59(U2)

CAD MAINTAINED DRAWING





△ DO NOT USE SPECIFIED STRENGTH FOR DESIGN WITHOUT EVALUATING PER WB-DC-20-1.1 R7.

NOTES:

1. FOR GENERAL NOTES, SEE 41N700-1.
2. △ CONCRETE SHALL BE CLASS 300.75 BFW EXCEPT AS NOTED IN NOTE B (41N373-5).
3. ALL REINFORCEMENT SHALL CONFORM TO ASTM SPECIFICATION A615, GRADE 60.
4. DO NOT WELD TO REINFORCEMENT.
5. FOR EMBEDDED PARTS, PIPES, AND CONDUITS, SEE STRUCTURAL STEEL, MECHANICAL, AND ELECTRICAL DRAWINGS.
6. CUT OR SPACE REINFORCEMENT AS REQUIRED TO CLEAR SLEEVES AND OPENINGS.
7. FOR REINFORCEMENT AND OTHER INFORMATION SEE TYPICAL PARTITION WALL ELEVATION SHOWN ON 41N373-1.
8. FOR THE RECORD OF CUT REINFORCEMENT NOT SHOWN ON DRAWINGS IN THIS SERIES, REFER TO WATTS BAR CALCULATION BOOK "REBAR CUTS, BOOK I" (MEDS ACCESSION NO. WBP 830923027).
9. FOR STRUCTURAL STEEL WALL RESTRAINT NOTES, SEE 41N373-4. FOR MK DETAILS, SEE 41N373-4.
10. PERMANENT SIGNS SHALL BE PLACED ON THE PARTITION WALLS SHOWN ON THIS SERIES OF DRAWINGS TO AVERT UNAUTHORIZED PLACEMENT OF ATTACHMENTS. FOR SIGN DETAILS, SEE DRAWING 46W405-5.
11. THE REINFORCEMENT SHOWN ON THIS SERIES OF AS-CONSTRUCTED DRAWINGS MAY HAVE HAD ADDITIONAL REBAR CUTS MADE SINCE JUNE 1982. FOR ALL REBAR CUTS MADE AFTER JUNE 1982, SEE INDIVIDUAL AS-DESIGNED DRAWING IN THE SERIES.
12. METHODOLOGY CALCULATION WCG-1-585 AND THE WORST CASE SELECTION CALCULATION WCG-1-742 WERE UTILIZED IN THE ENGINEERING EVALUATIONS OF THE CONCRETE FEATURES SHOWN IN THIS DRAWING SERIES.

** MAX DIST - 1/2 OF MAX SPCG FOR RESTRAINT BEING USED
MIN DIST - 4"

*** MAX DIST - MAX SPCG FOR RESTRAINT BEING USED
MIN DIST - 4 1/2"

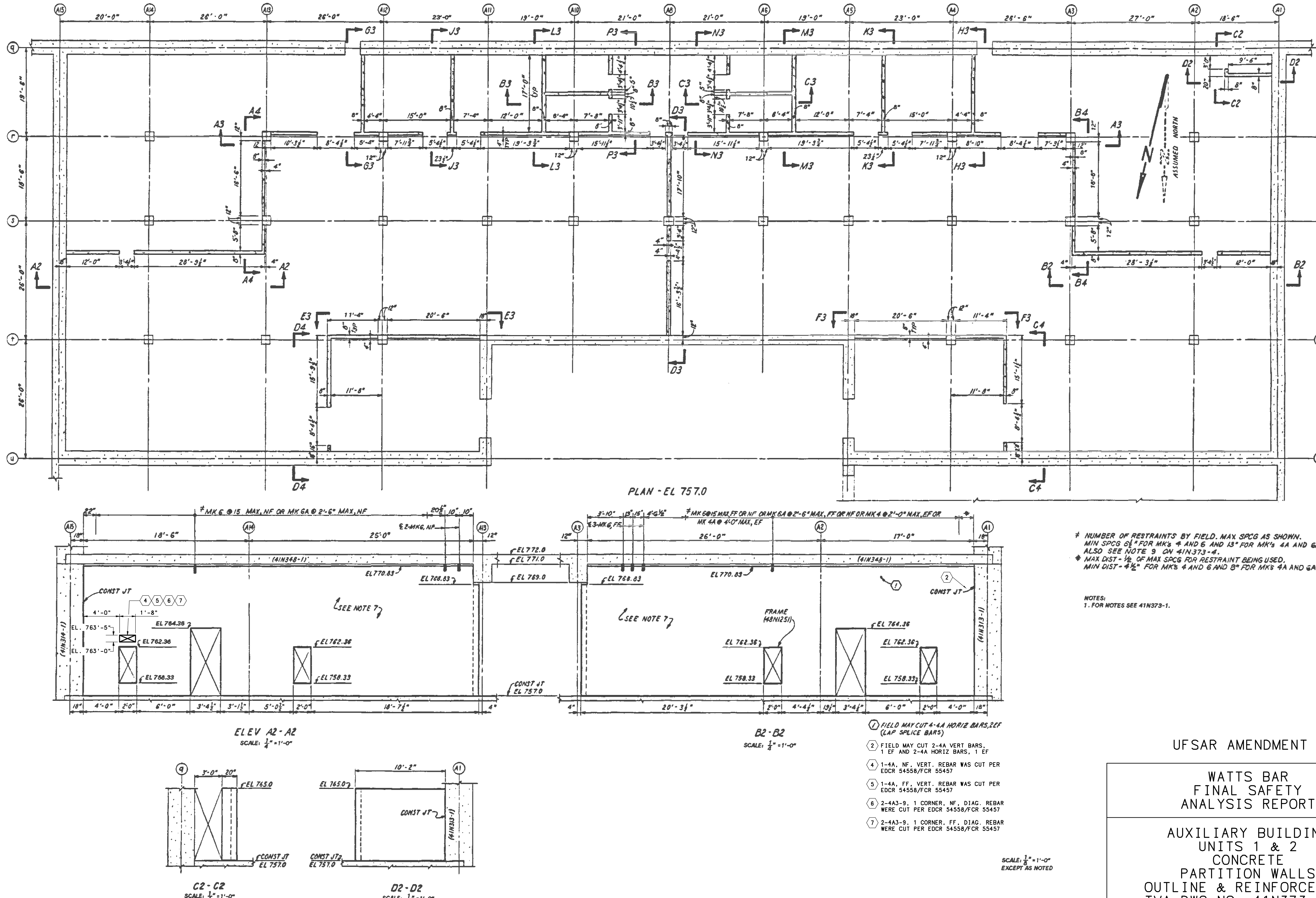
≢ NUMBER OF RESTRAINTS BY FIELD. MAX SPCG AS SHOWN. MIN SPCG 6 1/2".

* IF HEIGHT OF WALL EXCEEDS 14 FEET, A CONSTRUCTION JOINT WILL BE REQUIRED. LIMIT HEIGHT OF LIFT TO 14 FEET OR LESS. THE LOCATION AND NUMBER OF JOINTS TO BE DETERMINED BY FIELD.

SCALE 1/4"=1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
CONCRETE
PARTITION WALLS
OUTLINE & REINFORCEMENT
TVA DWG NO. 41N373-1 RH
FIGURE 3.8.4-60



\# NUMBER OF RESTRAINTS BY FIELD. MAX SPCG AS SHOWN. MIN SPCG 6½" FOR MK's 4 AND 6 AND 13" FOR MK's 4A AND 6A. ALSO SEE NOTE 9 ON 41N373-4.

* MAX DIST = ½ OF MAX SPCG FOR RESTRAINT BEING USED. MIN DIST = 4½" FOR MK's 4 AND 6 AND 8" FOR MK's 4A AND 6A.

NOTES:
1. FOR NOTES SEE 41N373-1.

(1) FIELD MAY CUT 4-4A HORIZ BARS, 2EF (LAP SPLICE BARS)

(2) FIELD MAY CUT 2-4A VERT BARS, 1 EF AND 2-4A HORIZ BARS, 1 EF

(4) 1-4A, NF, VERT. REBAR WAS CUT PER EDCR 54558/FCR 55457

(5) 1-4A, FF, VERT. REBAR WAS CUT PER EDCR 54558/FCR 55457

(6) 2-4A3-9, 1 CORNER, NF, DIAG. REBAR WERE CUT PER EDCR 54558/FCR 55457

(7) 2-4A3-9, 1 CORNER, FF, DIAG. REBAR WERE CUT PER EDCR 54558/FCR 55457

SCALE: ⅛" = 1'-0" EXCEPT AS NOTED

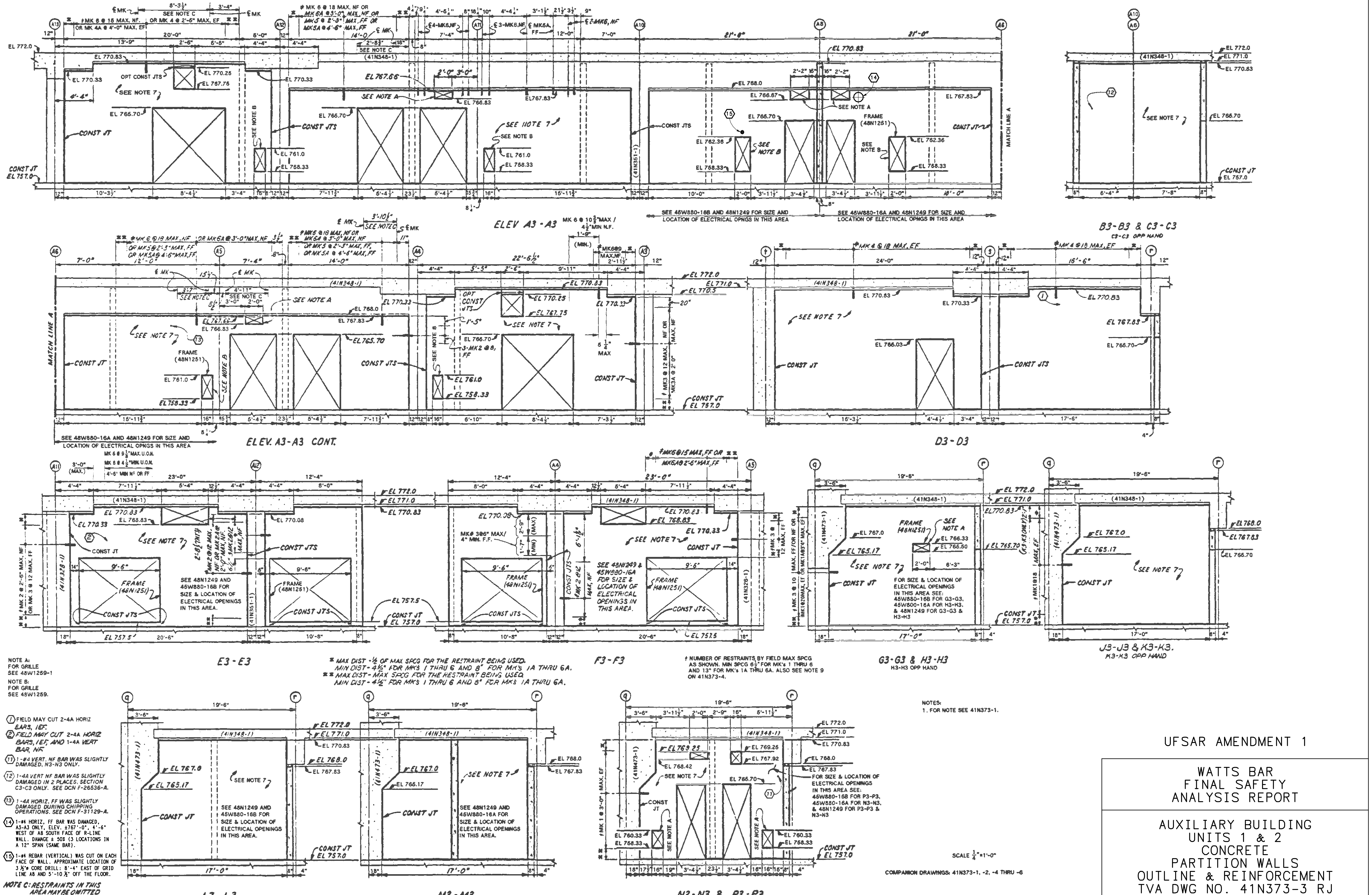
COMPANION DRAWINGS: 41N373-1, -3 THRU -6

UFSAR AMENDMENT 1

WATTS BAR FINAL SAFETY ANALYSIS REPORT

AUXILIARY BUILDING UNITS 1 & 2 CONCRETE PARTITION WALLS OUTLINE & REINFORCEMENT TVA DWG NO. 41N373-2 RJ

FIGURE 3.8.4-61



NOTE A:
FOR GRILLE
SEE 48W1259-1

NOTE B:
FOR GRILLE
SEE 48W1259.

* MAX DIST - ½ OF MAX SPCG FOR THE RESTRAINT BEING USED.
MIN DIST - 4½" FOR MK'S 1 THRU 6 AND 8" FOR MK'S 1A THRU 6A.
** MAX DIST - MAX SPCG FOR THE RESTRAINT BEING USED.
MIN DIST - 4½" FOR MK'S 1 THRU 6 AND 8" FOR MK'S 1A THRU 6A.

† NUMBER OF RESTRAINTS BY FIELD MAX SPCG AS SHOWN. MIN SPCG 6½" FOR MK's 1 THRU 6 AND 13" FOR MK's 1A THRU 6A. ALSO SEE NOTE 9 ON 41N373-4.

(1) FIELD MAY CUT 2-4A HORIZ BARS, IEF.

(2) FIELD MAY CUT 2-4A HORIZ BARS, IEF, AND 1-4A VERT BAR, NF.

(11) 1-#4 VERT. NF BAR WAS SLIGHTLY DAMAGED, N3-N3 ONLY.

(12) 1-4A VERT NF BAR WAS SLIGHTLY DAMAGED IN 2 PLACES. SECTION C3-C3 ONLY. SEE DCN F-26536-A.

(13) 1-4A HORIZ. FF WAS SLIGHTLY DAMAGED DURING CHIPPING OPERATIONS. SEE DCN F-31129-A.

(14) 1-#4 HORIZ, FF BAR WAS DAMAGED, A3-A3 ONLY, ELEV. ±767'-0", 4'-6" WEST OF A8 SOUTH FACE OF R-LINE WALL. DAMAGE ± 50% (3 LOCATIONS IN A 12" SPAN (SAME BAR).

(15) 1-#4 REBAR (VERTICAL) WAS CUT ON EACH FACE OF WALL. APPROXIMATE LOCATION OF 3 ½"ø CORE DRILL: 8'-4" EAST OF GRID LINE A8 AND 5'-10 ½" OFF THE FLOOR.

NOTE C: RESTRAINTS IN THIS AREA MAY BE OMITTED DUE TO INTERFERENCES.

NOTES:
1. FOR NOTE SEE 41N373-1.

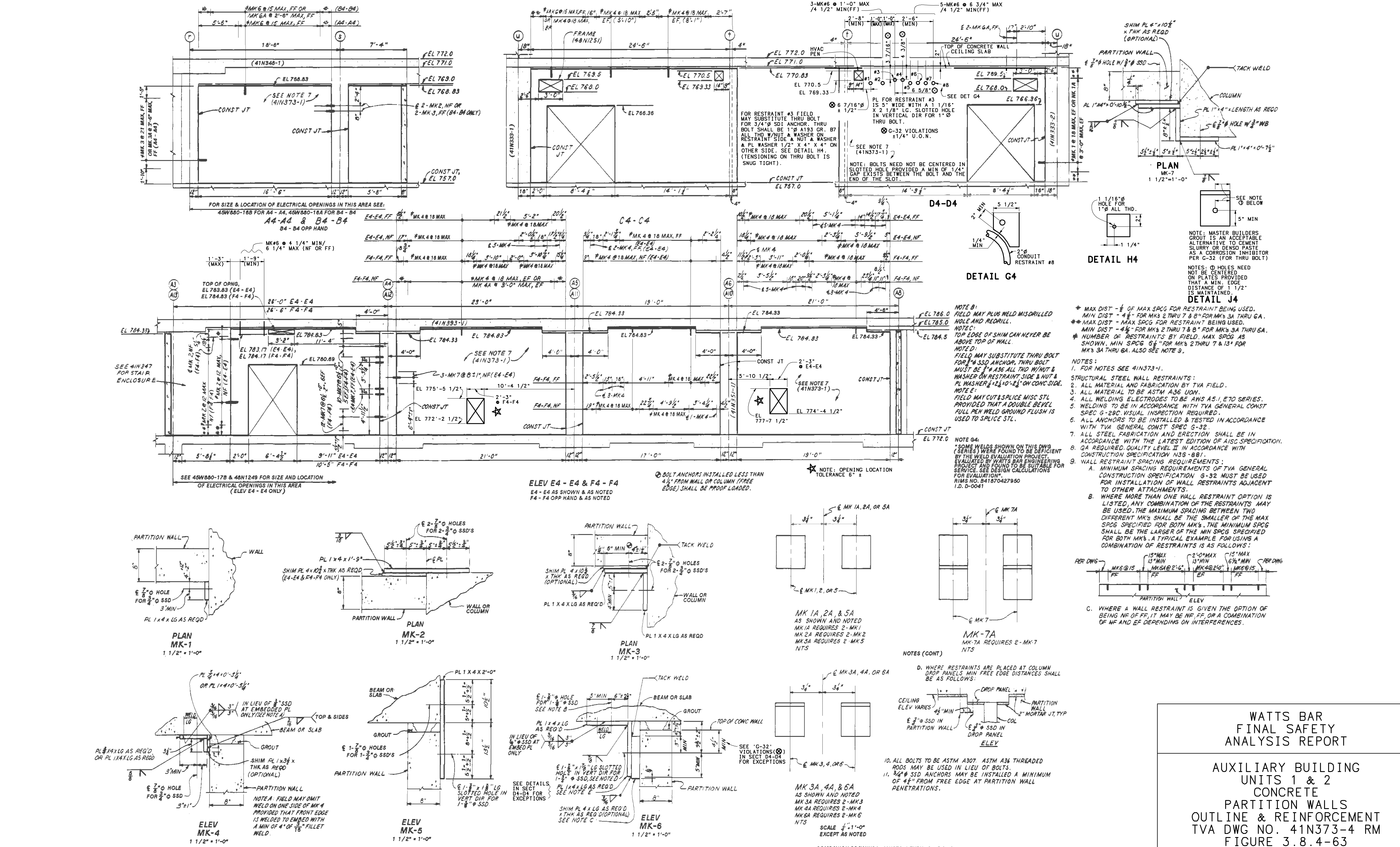
SCALE ¼"=1'-0"

COMPANION DRAWINGS: 41N373-1, -2, -4 THRU -6

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

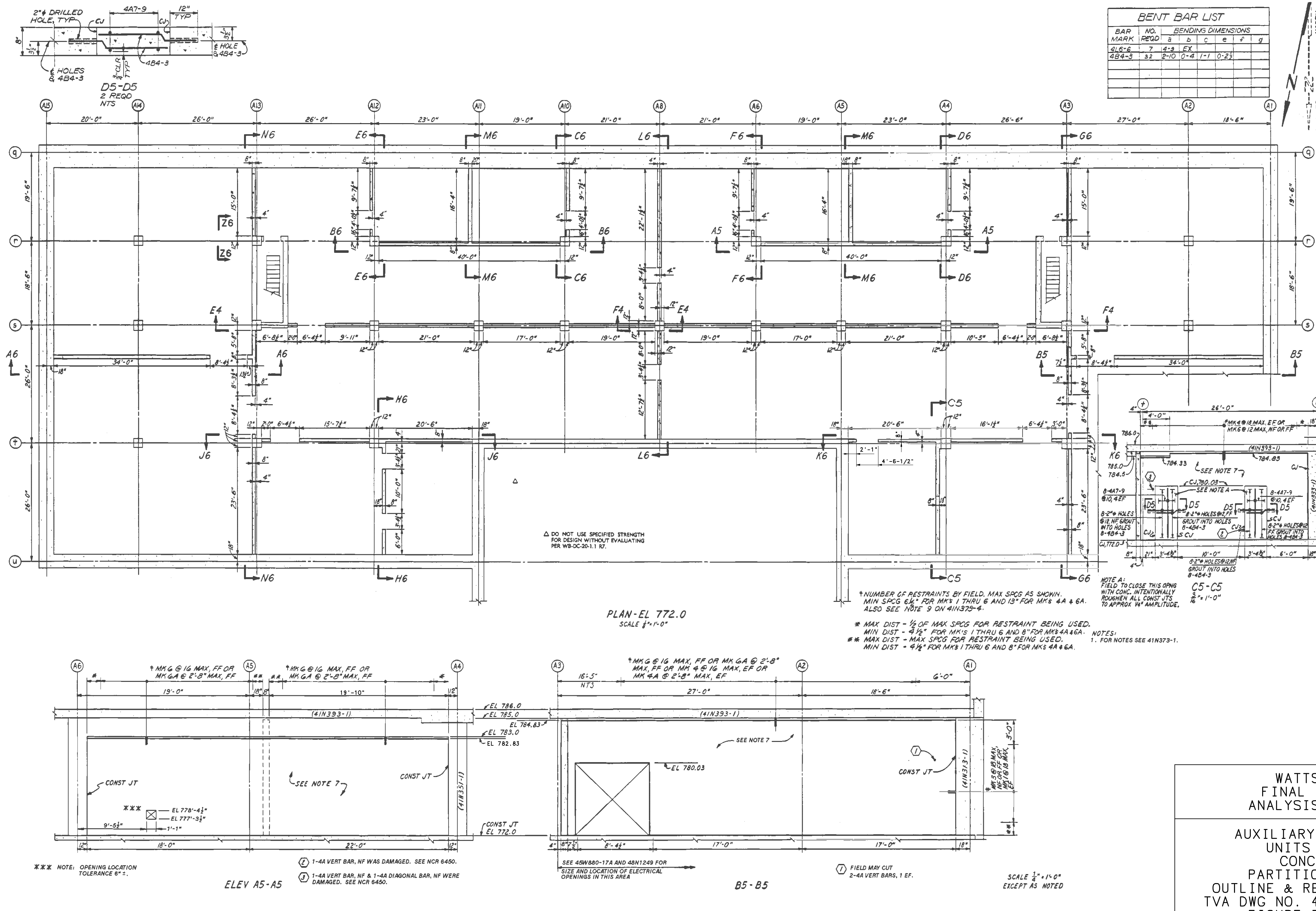
AUXILIARY BUILDING
UNITS 1 & 2
CONCRETE
PARTITION WALLS
OUTLINE & REINFORCEMENT
TVA DWG NO. 41N373-3 RJ
FIGURE 3.8.4-62



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
CONCRETE
PARTITION WALLS
OUTLINE & REINFORCEMENT
TVA DWG NO. 41N373-4 RM
FIGURE 3.8.4-63

CAD MAINTAINED DRAWING



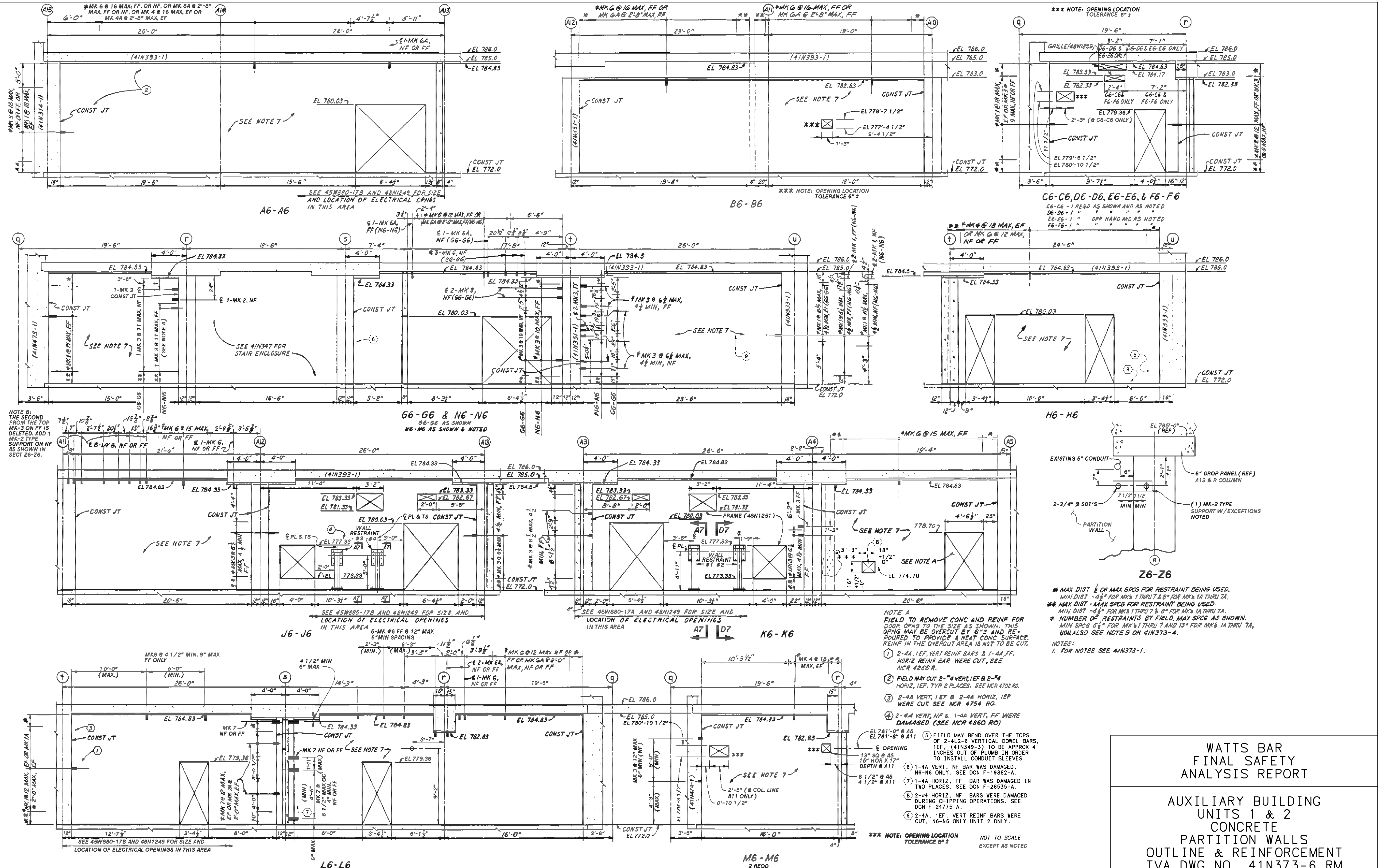
BENT BAR LIST

BAR MARK	NO. REQD	BENDING DIMENSIONS a	b	c	e	f	g
4L6-6	7	4-9	EX				
4B4-3	32	2-10	0-4	1-1	0-2½		

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
CONCRETE
PARTITION WALLS
OUTLINE & REINFORCEMENT
TVA DWG NO. 41N373-5 RL
FIGURE 3.8.4-64

CAD MAINTAINED DRAWING



NOTE A
FIELD TO REMOVE CONC AND REINF FOR DOOR OPNG TO THE SIZE AS SHOWN. THIS OPNG MAY BE OVERCUT BY 6"± AND RE-POURED TO PROVIDE A NEAT CONC SURFACE. REINF IN THE OVERCUT AREA IS NOT TO BE CUT.

1. 2-4A, 1EF, VERT REINF BARS & 1-4A, FF, HORIZ REINF BAR WERE CUT, SEE NCR 4266R.
2. FIELD MAY CUT 2-#4 VERT, 1EF & 2-#4 HORIZ, 1EF. TYP 2 PLACES. SEE NCR 4702 R0.
3. 2-4A VERT, 1EF & 2-4A HORIZ, 1EF WERE CUT. SEE NCR 4754 R0.
4. 2-4A VERT, NF & 1-4A VERT, FF WERE DAMAGED. (SEE NCR 4860 R0)
5. FIELD MAY BEND OVER THE TOPS OF 2-4L2-6 VERTICAL DOWEL BARS, 1EF, (41N349-3) TO BE APPROX 4 INCHES OUT OF PLUMB IN ORDER TO INSTALL CONDUIT SLEEVES.
6. 1-4A VERT, NF BAR WAS DAMAGED, N6-N6 ONLY. SEE DCN F-19882-A.
7. 1-4A HORIZ, FF, BAR WAS DAMAGED IN TWO PLACES. SEE DCN F-26535-A.
8. 2-#4 HORIZ, NF, BARS WERE DAMAGED DURING CHIPPING OPERATIONS. SEE DCN F-24775-A.
9. 2-4A, 1EF, VERT REINF BARS WERE CUT, N6-N6 ONLY UNIT 2 ONLY.

* MAX DIST ½ OF MAX SPCG FOR RESTRAINT BEING USED. MIN DIST - 4½" FOR MK's 1 THRU 7 & 8" FOR MK's 1A THRU 7A.
** MAX DIST - MAX SPCG FOR RESTRAINT BEING USED. MIN DIST - 4½" FOR MK's 1 THRU 7 & 8" FOR MK's 1A THRU 7A.
# NUMBER OF RESTRAINTS BY FIELD. MAX SPCG AS SHOWN. MIN SPCG 6½" FOR MK's 1 THRU 7 AND 13" FOR MK's 1A THRU 7A. USE ALSO SEE NOTE 9 ON 41N373-4.

NOTES:
1. FOR NOTES SEE 41N373-1.

NOTE B: THE SECOND FROM THE TOP MK-3 ON FF IS DELETED. ADD 1 MK-2 TYPE SUPPORT ON NF AS SHOWN IN SECT Z6-Z6.

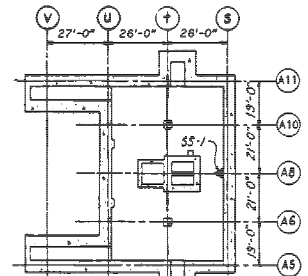
xxx NOTE: OPENING LOCATION TOLERANCE 6" ±

NOT TO SCALE EXCEPT AS NOTED

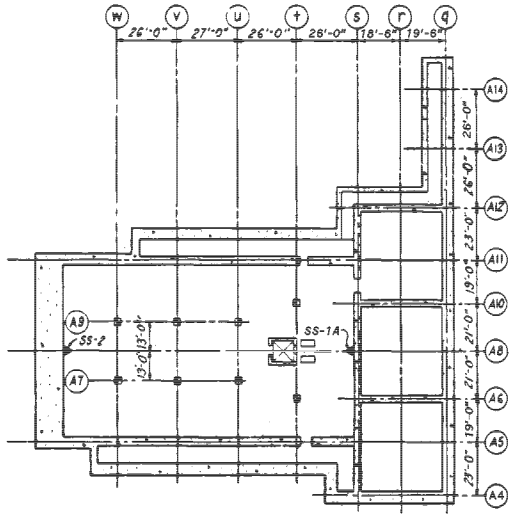
COMPANION DRAWINGS: 41N373-1 THRU -5

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
CONCRETE
PARTITION WALLS
OUTLINE & REINFORCEMENT
TVA DWG NO. 41N373-6 RM
FIGURE 3.8.4-65



PLAN - A1
EL 666.0±

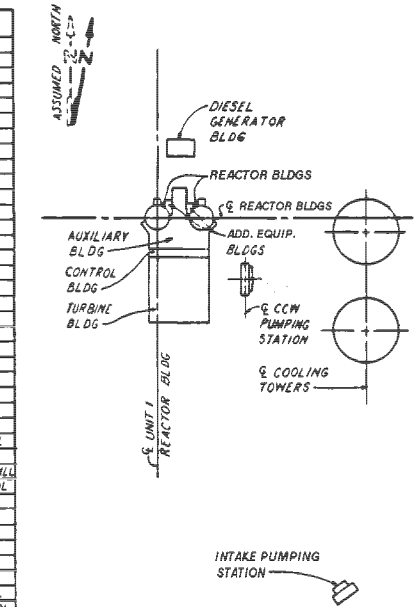


PLAN - B1
EL VARIES

SETTLEMENT STATION	ELEVATION	LOCATION	REMARKS
SS-1	EL 666.58	AB, 2'-0" NORTH OF S LINE	
* SS-2	EL 672.0	AB, 23'-0" NORTH OF W LINE	
SS-1A	EL 676.0	AB, 2'-0" NORTH OF S LINE	
* SS-3	EL 690.0	S, 2'-0" WEST OF A15 LINE	
SS-4	EL 729.0	18'-8" NORTH OF U LINE, 5'-0" EAST OF A14 LINE	
* SS-5	EL 696.62	0° ℄ REACTOR BLDG, 62'-0" RADIUS	
* SS-6	EL 696.62	90° ℄ REACTOR BLDG, 62'-0" RADIUS	
* SS-7	EL 696.62	229° ℄ REACTOR BLDG, 62'-0" RADIUS	
SS-8		35'-3" SOUTH OF X LINE, 31'-2" EAST OF A4 LINE	
SS-9	EL 729.0	2'-6" NORTH OF X LINE, 41'-6" EAST OF A11 LINE	
* SS-10	EL 690.0	AB, 15" SOUTH OF X LINE	
SS-11	EL 729.0	7'-6" SOUTH OF Y LINE, 41'-6" WEST OF AB LINE	
SS-12	EL 729.0	35'-9" SOUTH OF X LINE, 31'-2" WEST OF AB LINE	
* SS-13	EL 696.62	231° ℄ REACTOR BLDG, 62'-0" RADIUS	
* SS-14	EL 696.62	90° ℄ REACTOR BLDG, 62'-0" RADIUS	
* SS-15	EL 696.62	359° ℄ REACTOR BLDG, 62'-0" RADIUS	
SS-16	EL 729.0	18'-8" NORTH OF U LINE, 5'-0" WEST OF A2 LINE	
* SS-17	EL 690.0	S, 2'-0" EAST OF A1 LINE	
SS-2A	EL 692.0	6" WEST OF AB LINE, 23'-0" NORTH OF W LINE	
SS-1B	EL 692.0	AB, 2'-0" NORTH OF S LINE	
* SS-18	EL 683.5	2'-0" NORTH OF n LINE, 2'-0" EAST OF C1 LINE	
* SS-19	EL 689.5	C7, 2'-0" NORTH OF n LINE	
* SS-20	EL 683.5	2'-0" NORTH OF n LINE, 2'-0" WEST OF C13 LINE	
SS-21	EL 710.0	m, WEST FACE OF T16 LINE WALL	
SS-22	EL 710.0	m, T8	SOUTH FACE OF COL
SS-23	EL 710.0	m, EAST FACE OF T1 LINE WALL	
SS-24	EL 687.0	12'-6" EAST OF T1 LINE, 19'-7" SOUTH OF J LINE	SOUTH FACE OF WALL
SS-25	EL 710.0	f, T7	WEST FACE OF COL
SS-26	EL 687.0	5'-0" NORTH OF h LINE, 3'-0" EAST OF T3 LINE	UNIT 1
SS-26	EL 687.0	5'-0" NORTH OF h LINE, 3'-0" WEST OF T14 LINE	UNIT 2
SS-27	EL 687.0	19'-0" NORTH OF e LINE, 3'-0" WEST OF T6 LINE	UNIT 1
SS-27	EL 687.0	19'-0" NORTH OF e LINE, 3'-0" EAST OF T11 LINE	UNIT 2
SS-28	EL 682.0	17'-6" NORTH OF C LINE, 3'-0" EAST OF T3 LINE	UNIT 1
SS-28	EL 682.0	17'-6" NORTH OF C LINE, 3'-0" WEST OF T14 LINE	UNIT 2
SS-29	EL 687.0	12'-0" SOUTH OF C LINE, 5'-0" WEST OF T4 LINE	EAST FACE OF COL
SS-30	EL 726.0	b, T6	SOUTH FACE OF COL
SS-31	EL 726.0	b, T7	EAST FACE OF COL
SS-32	EL 726.0	b, T8	SOUTH FACE OF COL
SS-33	EL 710.0	f, T9	EAST FACE OF COL

* SETTLEMENT STATION SHALL BE TRANSFERRED BY THE DIVISION OF CONSTRUCTION TO THE WALL PRIOR TO PLACEMENT OF FILL SLAB.

TABLE CONT'D ON SH 2



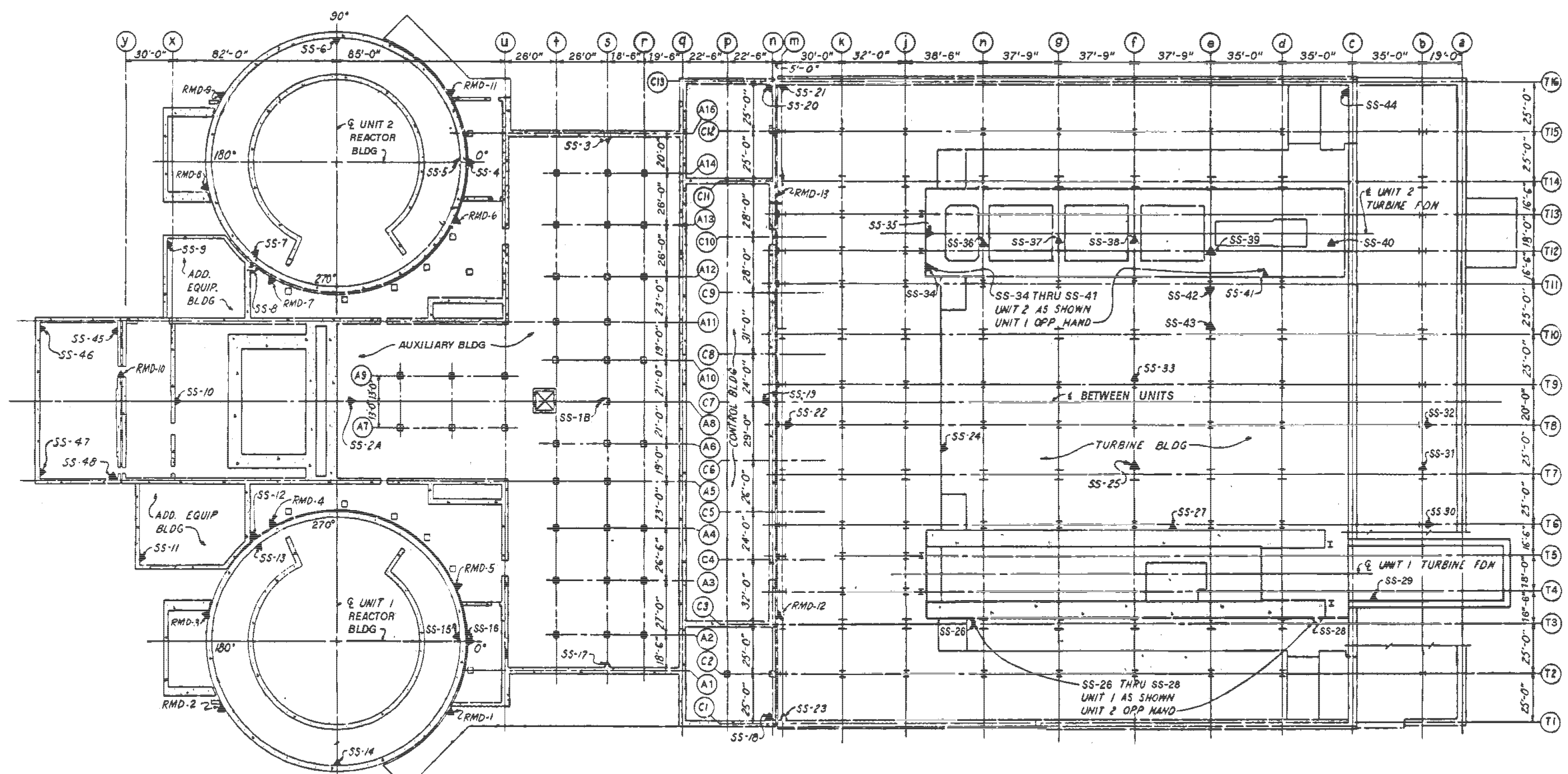
KEY PLAN
NOT TO SCALE

GENERAL NOTES:

1. FOR CONCRETE GENERAL FEATURES AND CONCRETE DETAILS SEE CONCRETE GENERAL FEATURE DRAWINGS AND CONCRETE OUTLINE DRAWINGS.
2. FOR LOCATION OF SETTLEMENT STATIONS NOT SHOWN ON PLAN SEE TABLE.
3. THE FIELD SHALL BE RESPONSIBLE FOR SETTING THE SETTLEMENT STATIONS, AND THE DATA SERVICES BRANCH OF WATER MANAGEMENT DIV SHALL BE RESPONSIBLE FOR PROVIDING THE SETTLEMENT STATION MARKERS AND READING THE ELEVATIONS.
4. THE LOCATION OF THE SETTLEMENT STATIONS MAY BE ADJUSTED TO SUIT FIELD CONDITIONS.
5. A PERMANENT CONTROL BENCH MARK SHALL BE LOCATED A MINIMUM OF 1000 FT WEST OF UNIT 1 CENTER LINE. THE EXACT LOCATION SHALL BE SET BY FIELD.
6. FOR NOTES CONCERNING RELATIVE MOVEMENT DETECTORS (RMDS), SEE SH 3.

SPECIFIC NOTES:

SETTLEMENT STATIONS 1 THROUGH 33 AND 42 THROUGH 48.
ELEVATIONS OF SETTLEMENT STATIONS SHALL BE READ MONTHLY THROUGH SEPTEMBER 1978 AND QUARTERLY THEREAFTER UNTIL THE END OF JUNE 1980 AT WHICH TIME THE READINGS WILL BE DISCONTINUED, EXCEPT FOR SETTLEMENT STATIONS 1 THROUGH 4 (DIESEL GENERATOR BLDG ONLY) AND 45 THROUGH 48 WHICH SHALL BE READ AND SUBMITTED TO EN DES SEMIANNUALLY FROM JULY 1980 UNTIL THE END OF CONSTRUCTION.
THE INITIAL READINGS SHALL BE TAKEN AS SOON AS PRACTICAL AFTER THE CONCRETE IS PLACED OR THE STEEL IS ERECTED. READINGS SHALL BE MADE BY SECOND ORDER METHODS AND PUBLISHED TO 0.001 FT.
THE SETTLEMENT STATION READINGS AND THE TOTAL VOLUME OF CONCRETE PLACED IN EACH BUILDING AND TURBINE FOUNDATION AT THE TIME OF THE READING SHALL BE SUBMITTED TO EN DES MONTHLY UNTIL SEPTEMBER 1978 AND QUARTERLY UNTIL JULY 1980. READINGS OF A SETTLEMENT STATION LOCATED ON WALLS, SLABS OR COLUMNS DIRECTLY UNDER ANOTHER SETTLEMENT STATION MAY BE DISCONTINUED AFTER THE ELEVATION OF THE SETTLEMENT STATION DIRECTLY ABOVE HAS BEEN ESTABLISHED. IN THE EVENT A SETTLEMENT STATION IS OBSTRUCTED BY EQUIPMENT OR FILL SLABS, THE SETTLEMENT STATION SHALL BE TRANSFERRED IN CLOSE PROXIMITY TO A WALL OR COLUMN THAT HAS DIRECT BEARING ON ROCK.
NOTES CONT'D ON SH 2

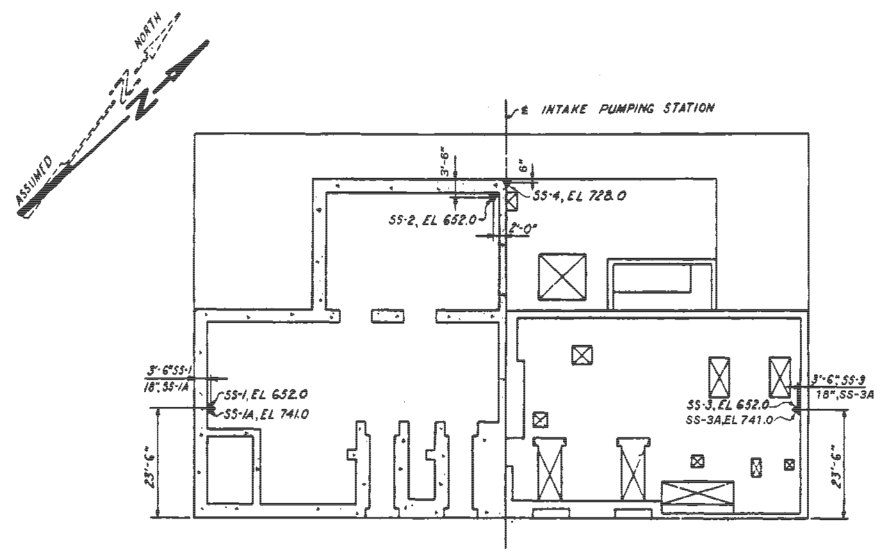


PLAN - C1
EL VARIES

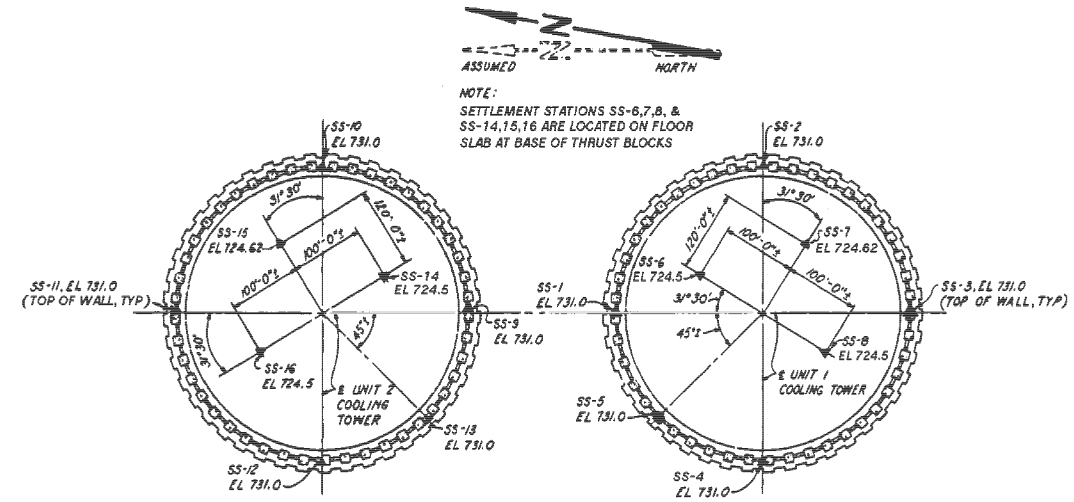
NOT TO SCALE
EXCEPT AS NOTED
COMPANION DRAWINGS: 10N203-2 & -3

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

GENERAL
LOCATION
OF
SETTLEMENT STATIONS
TVA DWG NO. 10N203-1 RD
FIGURE 3.8.4-66



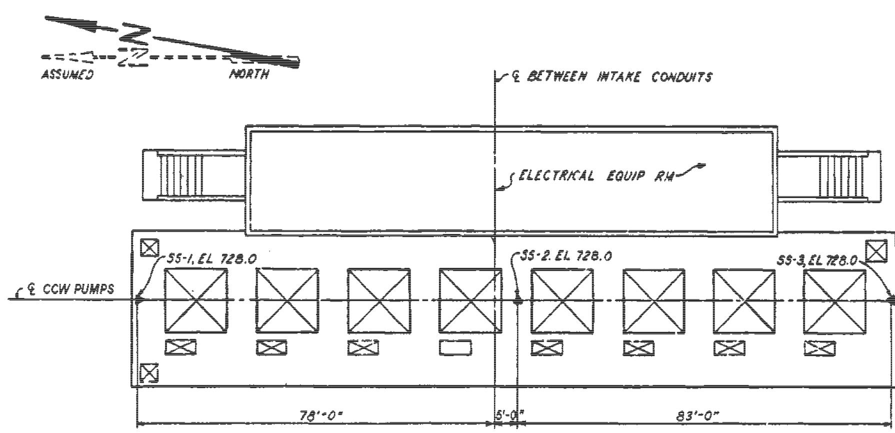
INTAKE PUMPING STATION



COOLING TOWERS
NOT TO SCALE

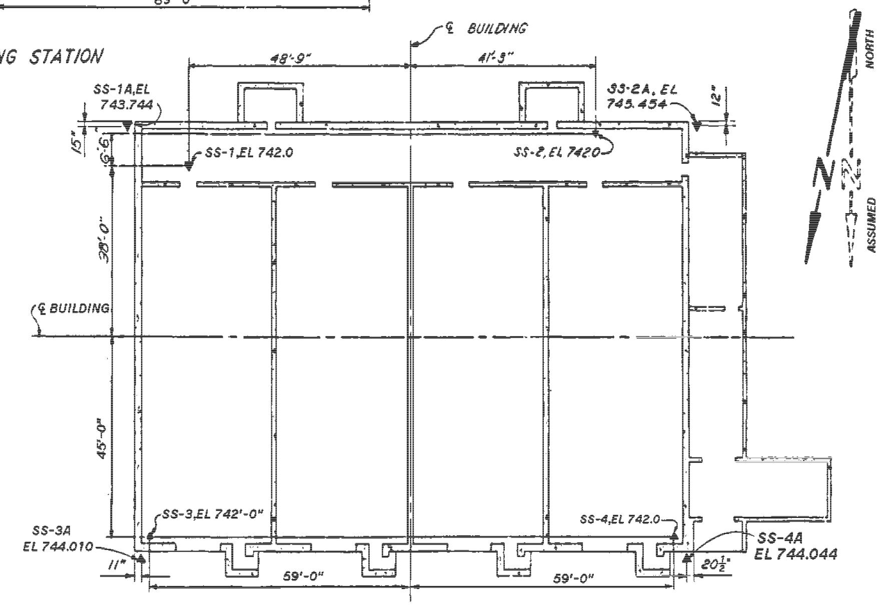
NOTE :

FOR ADDITION NOTES AND REFERENCE DWGS, SEE 10N203-1.
SETTLEMENT STATIONS 34 THROUGH 41
SETTLEMENT STATIONS SS-34 THROUGH SS-41 SHALL BE SET BEFORE INSTALLATION OF TURBINE GENERATOR PARTS. EXTREME CARE SHALL BE TAKEN NOT TO DISTURB THEM DURING TURBINE AND GENERATOR ERECTION AND STARTUP. ELEVATIONS SHALL BE READ (1) IMMEDIATELY BEFORE INSTALLATION OF TURBINE-GENERATOR PARTS (2) IMMEDIATELY BEFORE INITIAL OPERATION OF THE UNIT, AND (3) AFTER 1 YEAR OF OPERATION. THEREAFTER ELEVATIONS OF THESE STATIONS SHALL BE READ WHEN REQUESTED BY THE DIVISION OF POWER PRODUCTION. THESE READINGS SHALL BE MADE BY FIRST ORDER METHODS AND PUBLISHED TO 0.001 FT
SETTLEMENT READINGS FOR THE TURBINE GENERATORS SHALL BE DISCONTINUED AFTER APRIL 83. THIS SUPERSEDES THE ABOVE INSTRUCTIONS.



CCW PUMPING STATION

SETTLEMENT STATION	ELEVATION	LOCATION	REMARKS
SS-34	EL 755.0	4'-6" WEST OF T6 LINE, 11'-10" SOUTH OF J LINE	UNIT 1
SS-34	EL 755.0	4'-6" EAST OF T12 LINE, 11'-10" SOUTH OF J LINE	UNIT 2
SS-35	EL 755.0	9'-0" EAST OF T6 LINE, 11'-10" SOUTH OF J LINE	UNIT 1
SS-35	EL 755.0	9'-0" EAST OF T12 LINE, 11'-10" SOUTH OF J LINE	UNIT 2
SS-36	EL 755.0	2'-0" WEST OF T6 LINE, 3" SOUTH OF h LINE	UNIT 1
SS-36	EL 755.0	2'-0" EAST OF T12 LINE, 3" SOUTH OF h LINE	UNIT 2
SS-37	EL 755.0	g, 6'-0" WEST OF T6 LINE	UNIT 1
SS-37	EL 755.0	g, 6'-0" EAST OF T12 LINE	UNIT 2
SS-38	EL 755.0	f, 6'-0" WEST OF T6 LINE	UNIT 1
SS-38	EL 755.0	f, 6'-0" EAST OF T12 LINE	UNIT 2
SS-39	EL 755.0	T6, 12" NORTH OF e LINE	UNIT 1
SS-39	EL 755.0	T12, 12" NORTH OF e LINE	UNIT 2
SS-40	EL 755.0	6'-0" WEST OF T6 LINE, 5'-2" NORTH OF C LINE	UNIT 1
SS-40	EL 755.0	6'-0" EAST OF T12 LINE, 5'-2" NORTH OF C LINE	UNIT 2
SS-41	EL 755.0	4'-6" WEST OF T6 LINE, 10'-9½" NORTH OF d LINE	UNIT 1
SS-41	EL 755.0	4'-6" EAST OF T12 LINE, 10'-9½" NORTH OF d LINE	UNIT 2
SS-42	EL 710.0	e, T11	WEST FACE OF COL
SS-43	EL 710.0	e, T11	EAST FACE OF COL
SS-44	EL 710.0	6'-0" WEST OF T6 LINE, 1'-9" NORTH OF C LINE	NORTH FACE OF WALL
SS-45	EL 729.0	2'-0" WEST OF A11 LINE, 4'-8" NORTH OF Y LINE	
SS-46	EL 729.0	2'-0" WEST OF A11 LINE, 43'-8" NORTH OF Y LINE	
SS-47	EL 729.0	2'-0" EAST OF A6 LINE, 43'-8" NORTH OF Y LINE	
SS-48	EL 729.0	2'-0" EAST OF A6 LINE, 4'-8" NORTH OF Y LINE	



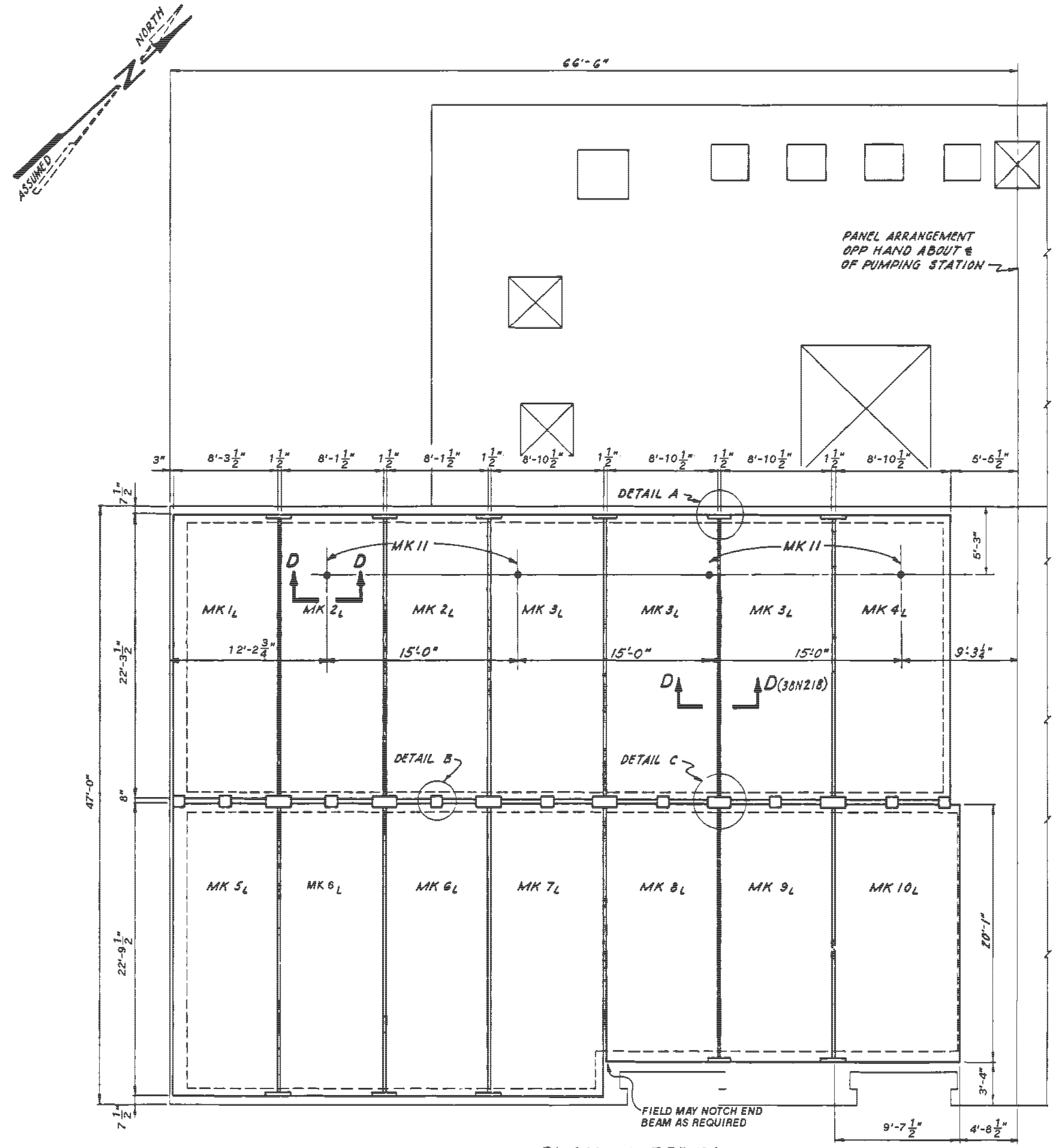
DIESEL GENERATOR BUILDING

NOT TO SCALE
EXCEPT AS NOTED
COMPANION DRAWING: 10N203-1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

GENERAL
LOCATION
OF
SETTLEMENT STATIONS
TVA DWG NO. 10N203-2 RE
FIGURE 3.8.4-67

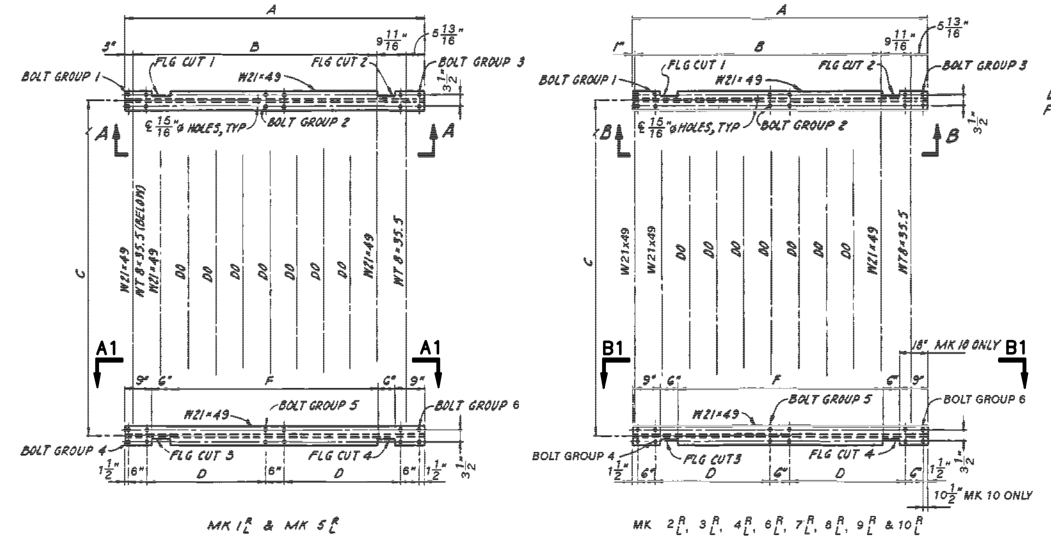
CAD MAINTAINED DRAWING



PLAN EL 755.74

3/16" = 1'-0"

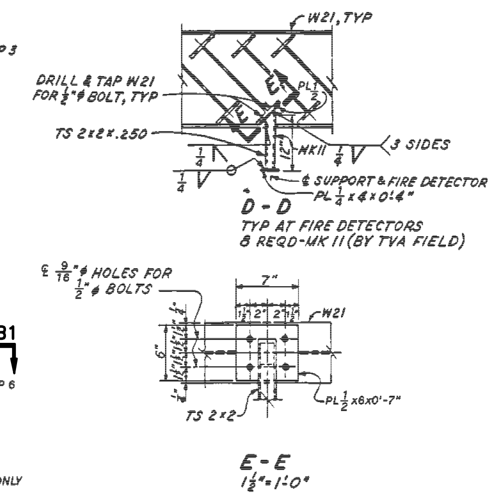
NOTE:
HOLES CUT FOR NEW ANCHOR BOLT ASSEMBLIES MK 1 MAY BE TORCH CUT FOR PANELS MK 8L 9L & 10L. HOLES FOR ABANDONED BOLT ASSEMBLIES NEED NOT BE REPAIRED.



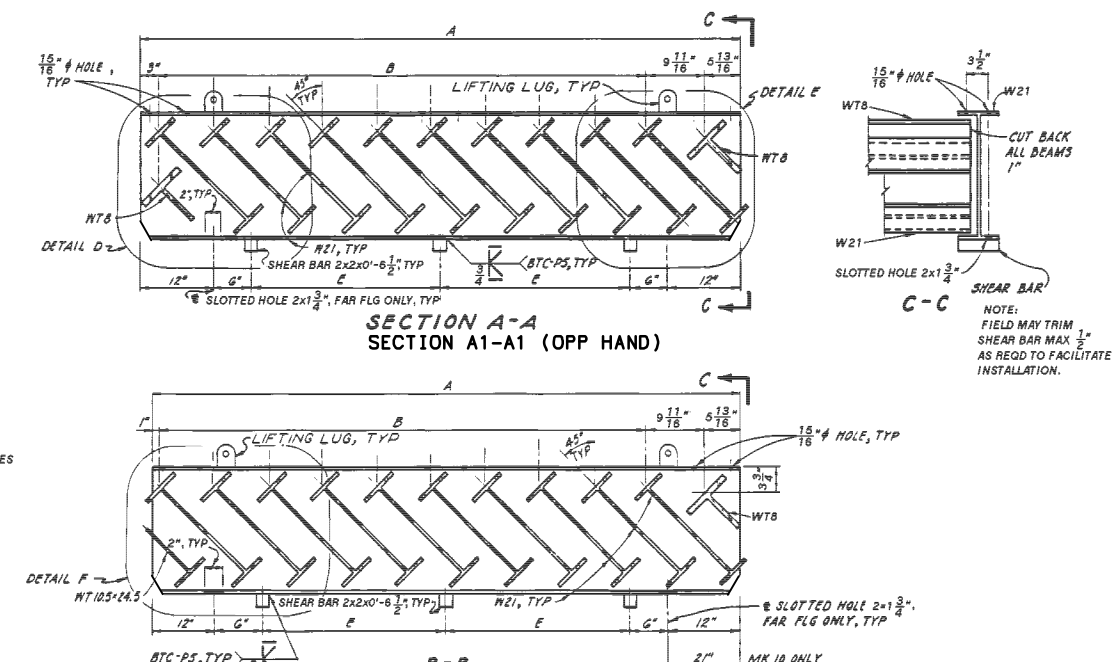
MISSILE PROTECTION FRAMING PANELS

ALL MARKS NOTED "L" ARE AS SHOWN, THOSE NOTED "R" ARE OPP HAND

1/2" = 1'-0"



NOTES:
1. FOR LIST SEE 38N216.
2. FOR SECTIONS AND DETAILS NOT SHOWN SEE 38N218.
3. FOR DIMENSIONS DENOTED BY LETTERS SEE TABLE.
4. ALL MARKS THIS SHEET TO HAVE PREFIX 38N217.
5. THE W21x49 BEAMS USED IN THE FABRICATION OF PANELS MK. NO. 1L THROUGH 10L AND MK. NO. 1R THROUGH 10R WERE IDENTIFIED AS HAVING DEFICIENCIES IN WBPER 950012 R0. THE DEFICIENCIES WERE LINEAR INDICATIONS EVALUATED AS ROLLING MARKS IN THE FLANGES OF THE BEAMS. THE DISPOSITION OF THESE DEFICIENCIES ARE ACCEPTABLE AS IS BASED UPON THE ANALYSIS PERFORMED PER CALCULATION WCG-1-622 R2.



SECTION A-A

SECTION A1-A1 (OPP HAND)

NOTE:
FIELD MAY TRIM SHEAR BAR MAX 1/2" AS REQD TO FACILITATE INSTALLATION.

B-B

B1-B1 (OPP HAND)

MARKS		DETAIL DIMENSIONS										BOLT GROUP							FLANGE CUT				
MK NO.	REQD	A	B	C	D	E	F	G	H	J		1	2	3	4	5	6		1	2	3	4	
MK 1R/L	1 EA	8'-3 1/2"	9 SPACES @ 9" = 6'-9"	22'-3 1/2"	3'-3 1/4"	2'-7 3/4"	5'-9 1/2"	2'-1 3/4"	6 SPACES @ 3'-0" = 18'-0"	2'-1 3/4"		NA	NA	X	X	X	X		NA	NA	X	X	
MK 2R/L	2 EA	8'-1 1/2"	9 SPACES @ 9" = 6'-9"	22'-3 1/2"	3'-2 1/4"	2'-6 3/4"	5'-7 1/2"	2'-1 3/4"	6 SPACES @ 3'-0" = 18'-0"	2'-1 3/4"		X	NA	X	X	X	X		NA	NA	X	X	
MK 3R/L	3 EA	8'-10 1/2"	10 SPACES @ 9" = 7'-6"	22'-3 1/2"	3'-6 3/4"	2'-11 1/4"	6'-4 1/2"	2'-1 3/4"	6 SPACES @ 3'-0" = 18'-0"	2'-1 3/4"		X	NA	X	X	X	X		NA	NA	X	X	
MK 4R/L	1 EA	8'-10 1/2"	10 SPACES @ 9" = 7'-6"	22'-3 1/2"	3'-6 3/4"	2'-11 1/4"	6'-4 1/2"	2'-1 3/4"	6 SPACES @ 3'-0" = 18'-0"	2'-1 3/4"		X	NA	NA	X	X	X		NA	NA	X	X	
MK 5R/L	1 EA	8'-3 1/2"	9 SPACES @ 9" = 6'-9"	22'-9 1/2"	3'-3 1/4"	2'-7 3/4"	5'-9 1/2"	10 3/4"	7 SPACES @ 3'-0" = 21'-0"	10 3/4"		X	X	X	NA	NA	X		X	X	NA	NA	
MK 6R/L	2 EA	8'-1 1/2"	9 SPACES @ 9" = 6'-9"	22'-9 1/2"	3'-2 1/4"	2'-6 3/4"	5'-7 1/2"	10 3/4"	7 SPACES @ 3'-0" = 21'-0"	10 3/4"		X	X	X	X	NA	X		X	X	NA	NA	
MK 7R/L	1 EA	8'-10 1/2"	10 SPACES @ 9" = 7'-6"	22'-9 1/2"	3'-6 3/4"	2'-11 1/4"	6'-4 1/2"	10 3/4"	7 SPACES @ 3'-0" = 21'-0"	10 3/4"		X	X	X	X	NA	NA		X	X	NA	NA	
MK 8R/L	1 EA	8'-10 1/2"	10 SPACES @ 9" = 7'-6"	20'-1"	3'-6 3/4"	2'-11 1/4"	6'-4 1/2"	14 1/4"	6 SPACES @ 3'-0" = 18'-0"	10 3/4"		X	X	X	NA	NA	X		X	X	NA	NA	
MK 9R/L	1 EA	8'-10 1/2"	10 SPACES @ 9" = 7'-6"	20'-1"	3'-6 3/4"	2'-11 1/4"	6'-4 1/2"	14 1/4"	6 SPACES @ 3'-0" = 18'-0"	10 3/4"		X	X	X	X	NA	X		X	X	NA	NA	
MK 10R/L	1 EA	9'-7 1/2"	11 SPACES @ 9" = 8'-3"	20'-1"	3'-6 3/4"	2'-11 1/4"	6'-4 1/2"	14 1/4"	6 SPACES @ 3'-0" = 18'-0"	10 3/4"		X	X	X	X	NA	NA		X	X	NA	NA	

NA = NOT APPLICABLE

X = APPLIES

ECN 1261

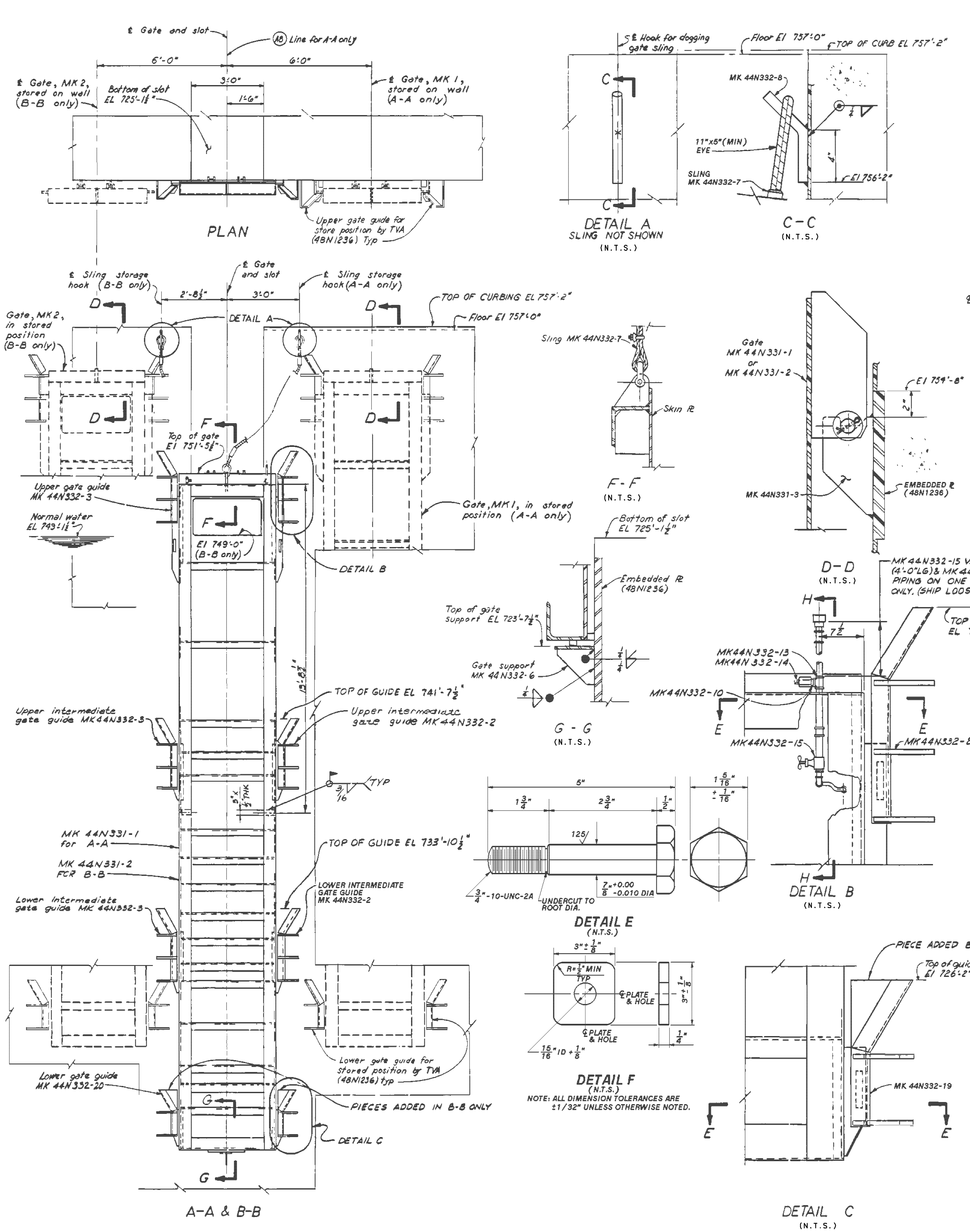
SCALE 1" = 1'-0"

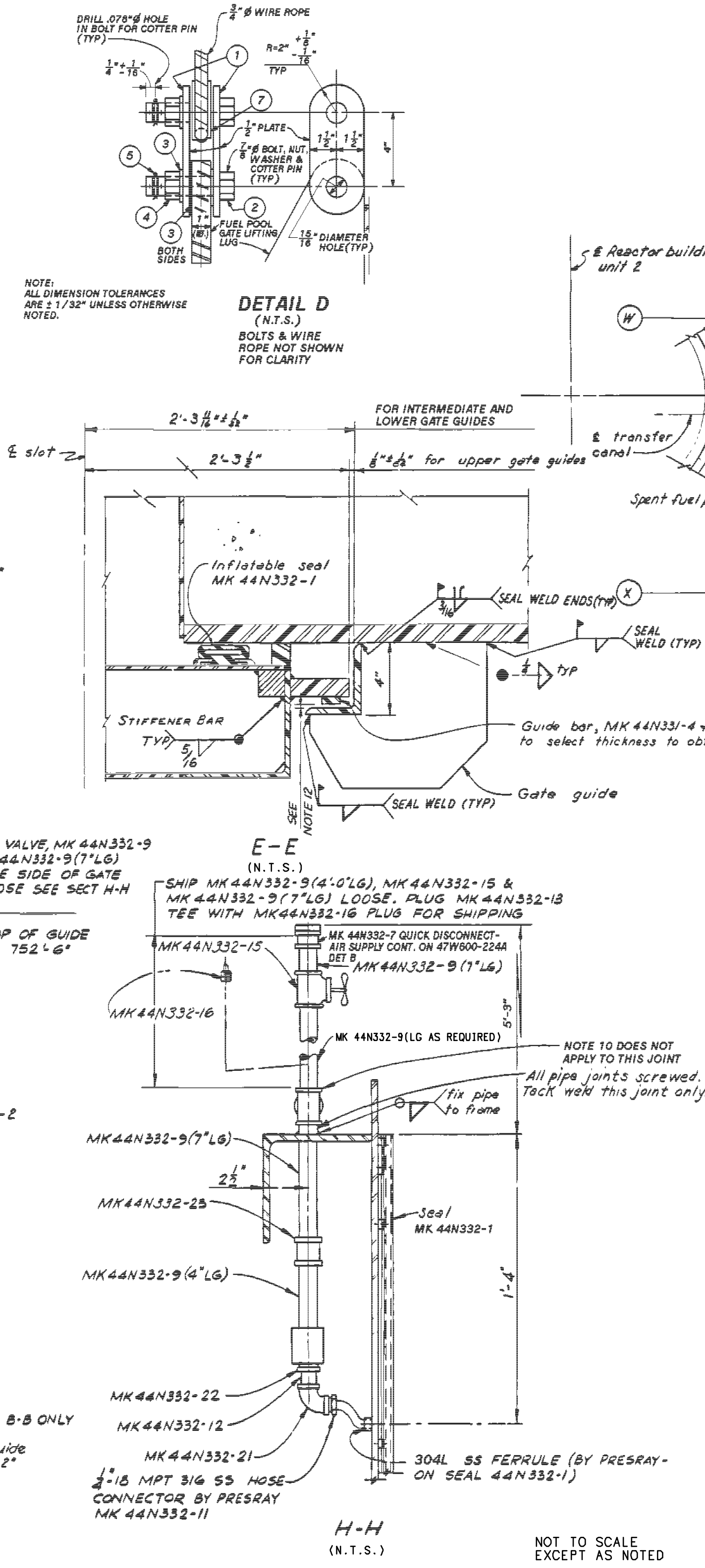
COMPANION DWG: 38N216 & 38N218

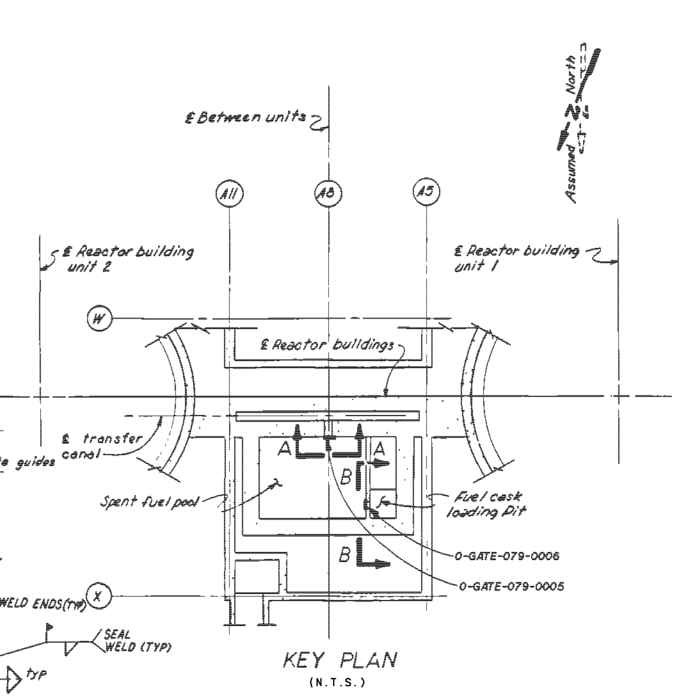
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

WATER SUPPLY
UNITS 1 & 2
STRUCTURAL STEEL
ERCW INTAKE PUMPING STA
MISSILE PROT FRAMING SH-2
TVA DWG NO. 38N217 RF
FIGURE 3.8.4-68

CAD MAINTAINED DRAWING







NOTES:

1. TWO GATES REQUIRED. IDENTICAL EXCEPT MK44N331-2 HAS 3'-0" WIDE BY 1'-6" HIGH OPENING IN SKIN PLATE AND 3" X 1/2" THK BACK-UP PLATE AS SHOWN IN A-A & B-B.
2. ALL MARK NUMBERS ON THIS DRAWING TO HAVE PREFIX 44N331 UNLESS OTHERWISE NOTED.
3. BOTH GATES ARE OPERATED ONLY UNDER BALANCED HEAD.
4. GATE, MK 44N331-1, (UNID 0-GATE-079-0005) SERVES TO DEWATER THE TRANSFER CANAL FOR REPAIRS AND MAINTENANCE. GATE, MK 44N331-2, (UNID 0-GATE-079-0006) IS UTILIZED IN ACCORDANCE WITH SPENT FUEL POOL COOLING AND CLEANING SYSTEM DESCRIPTION, N3-78-4001.
5. IN SETTING A GATE TO CLOSE A SLOT, THE UPPER GUIDES ARE ABOVE WATER AND ENGAGE BEFORE THE LOWER GUIDES. AS THE GATE IS LOWERED SLOWLY, THE SMALL CLEARANCES AT THE UPPER GUIDES ALIGN THE GATE FOR PROPER ENGAGEMENT OF THE LOWER GUIDES.
6. TO SEAL GATE, INFLATE SEAL TO 35±5 PSI. RELIEF VALVES FOR SEAL TO BE SET AT 40 PSI.
7. ALL MATERIAL DESIGNATED STAINLESS STEEL OR SS TO BE ASTM A 276, A 240 OR A 167, TYPE 304, COLD REDUCED, ANNEALED, AND PICKLED FOR NUCLEAR POOL LINER APPLICATION. EXCEPT HOT REDUCED PERMISSIBLE WITH APPROVAL OF THE ENGINEER.
8. WHERE ANGLES ARE SHOWN, EQUIVALENT BENT OR WELDED PLATES MAY BE USED. SEE NOTE 6D ON DWG 44N332.
9. THE FUEL POOL GATES AND GUIDES HAVE BEEN QUALFIED AS SEISMIC CATEGORY I FOR THE ECN 3475 REVISION OF DRAWINGS 44N330, 44N331, AND 44N332
10. TVA FIELD TO SEAL WELD ALL THREADED CONNECTIONS.
11. FIELD REQUIREMENTS; WELDING TO BE IN ACCORDANCE WITH GEN. CONST. SPEC. G-29C AND G-29M. VISUAL INSPECTION TO BE IN ACCORDANCE WITH GEN. CONST. SPEC. G-29C. NO VISUAL POROSITY PERMITTED. SEE NOTE 6 ON 44N332 FOR ADDITIONAL REQUIREMENTS.
12. GAP TO BE AS REQUIRED TO ALLOW EASY INSTALLATION OF GATE WITHOUT BINDING, UP TO A MAXIMUM OF 3/16" OVERALL.

FOR FIELD NOTES SEE DWG 44N332

NOT TO SCALE EXCEPT AS NOTED

ITEM NO.	QTY	BILL OF MATERIAL	MATERIAL
1	2	1/2" PLATE	ASTM A276 TYPE 304, CF, COND A
2	2	7/8"Ø BOLT SEE DET E	ANSI B18.2.1 DIMENSIONS MAT'L ASTM A564 TYPE 630
3	4	WASHER - 3"O.D. x 15/16"I.D. x 1/4" DETAIL F	ASTM A240 TYPE 304
4	2	HEAVY HEX NUTS 3/4"Ø	ASTM A564 TYPE 630 DIM. PER ANSI B18.2.2
5	2	1/16" COTTER PIN x 1 1/2" LONG	ANSI B18.8.1, ASTM A493 TYPE 304, 1/16"Ø x 1 1/2" LONG
6	1	3/4"Ø IWRC	CORROSION RESISTANT 6x19 OR 6x37
7	1	3/4"Ø THIMBLE SS TYPE 304	CROSBY WIRE ROPE THIMBLE TYPE SS 414

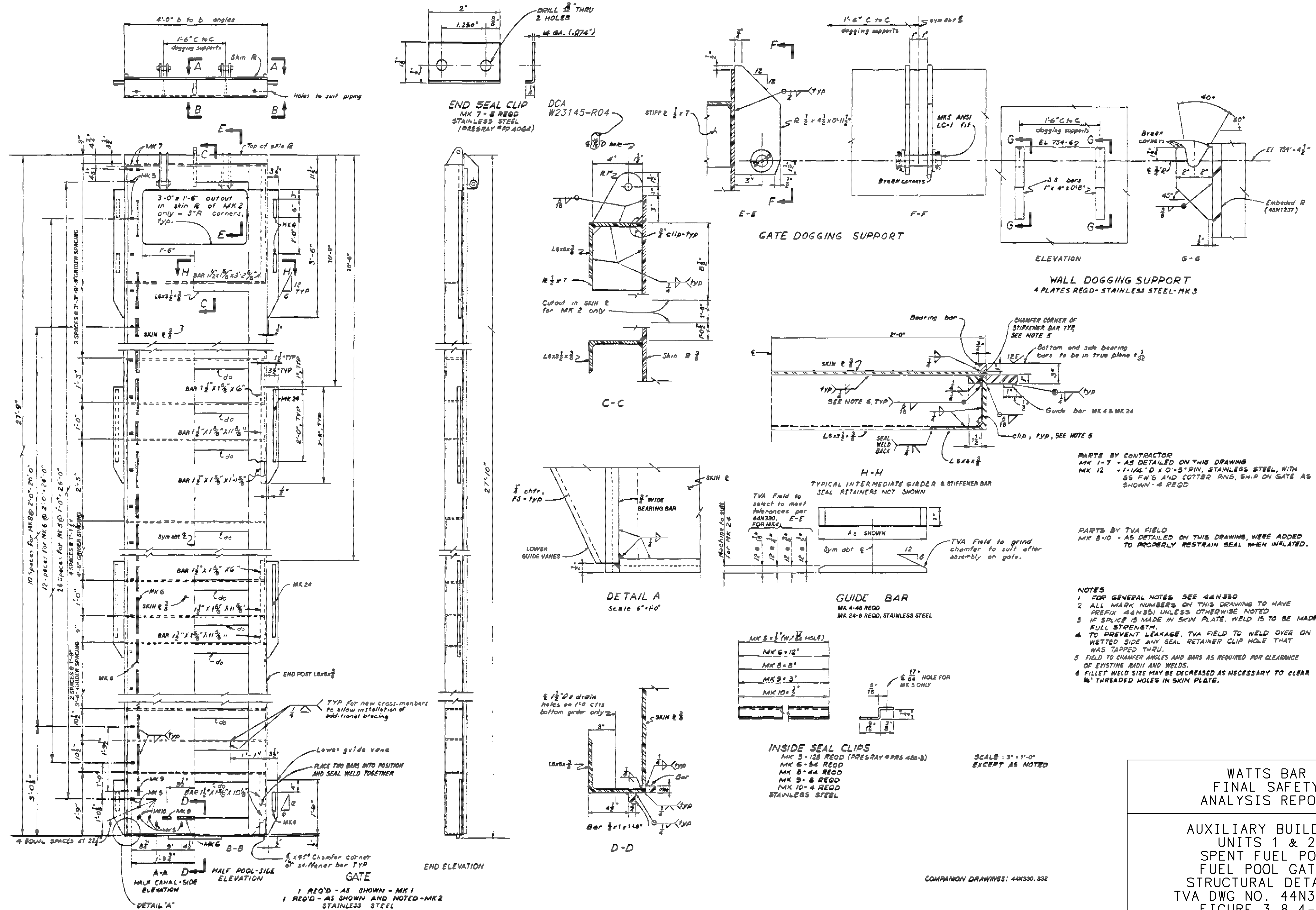
NOTE: QUANTITIES LISTED ARE FOR ONE ASSEMBLY - TWO ASSEMBLIES ARE REQUIRED.

COMPANION DRAWINGS: 44N331, 332

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNITS 1 & 2
SPENT FUEL POOL
FUEL POOL GATES
ARRANGEMENTS
TVA DWG NO. 44N330 RG
FIGURE 3.8.4-69

CAD MAINTAINED DRAWING



WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

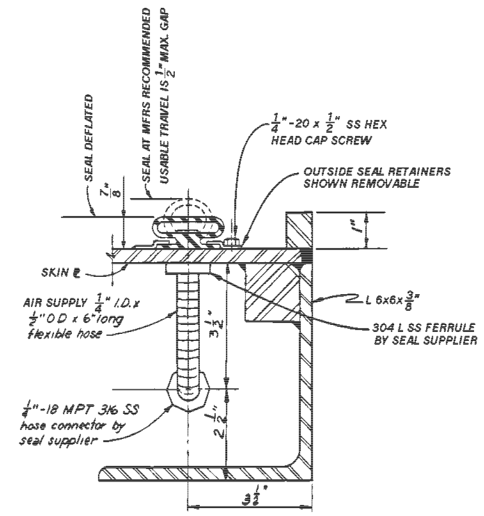
AUXILIARY BUILDING
UNITS 1 & 2
SPENT FUEL POOL
FUEL POOL GATES
STRUCTURAL DETAILS
TVA DWG NO. 44N331 RE
FIGURE 3.8.4-70

CAD MAINTAINED DRAWING

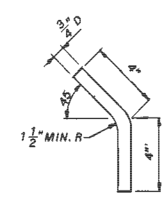
FIELD NOTES:

THE FOLLOWING MODIFICATIONS UNDER ECN 3475 WILL QUALIFY THE GATES FOR SEISMIC CATEGORY I (REF: WB-DC-40-43.)

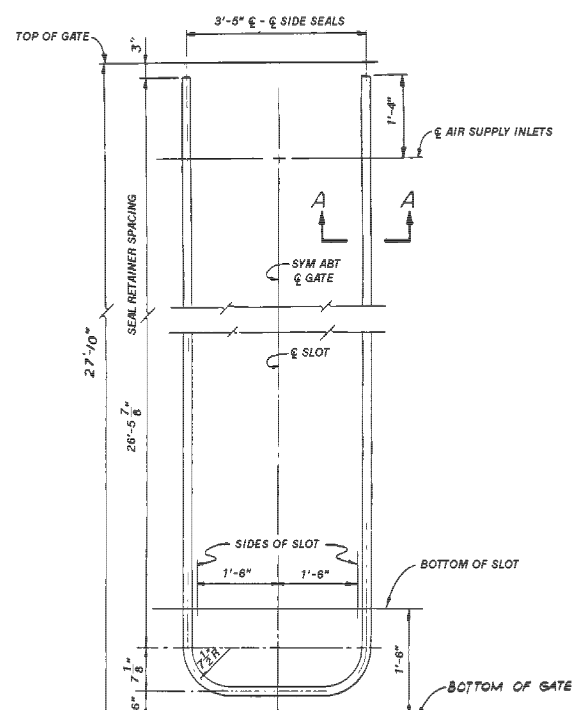
1. VERIFY THAT ALL PREVIOUS FIELD WORK (WELDING, MATERIALS ETC.) IS IN ACCORDANCE WITH DESIGN DRAWINGS. IF A NONCONFORMANCE EXISTS IT MUST BE CORRECTED.
2. REMOVE THE EXISTING MIDDLE GATE RESTRAINT GUIDE VANES FROM GATES.
3. ADD TWO INTERMEDIATE GUIDE VANES TO EACH SIDE OF THE GATES AND FIVE ADDITIONAL CROSS MEMBERS TO EACH GATE AS SHOWN IN THE POOL SIDE ELEVATION ON DRAWING 44N331.
4. ADD STIFFENER BARS TO THE INSIDE CORNER OF THE GATES OPPOSITE EACH GUIDE VANE (INCLUDING THE UPPER AND LOWER GUIDE VANES) AS SHOWN IN H-H ON 44N331.
5. ADD INTERMEDIATE GATE GUIDES TO THE WALLS EIGHT PLACES TO MATE WITH THE INTERMEDIATE GUIDE VANES WHEN THE GATES ARE IN THE OPERATING POSITION.
6. ALL MATERIALS AND WELDING TO BE IN ACCORDANCE WITH QUALITY LEVEL II OF CONSTRUCTION SPECIFICATION N3G-881.
   - A. ALL MATERIALS TO BE ASTM A-276, A240, OR A167; TYPE 304 STAINLESS STEEL; COLD REDUCER; AND ANNEALED.
   - B. WELD ELECTRODES TO BE AWS A5.4, E308 SERIES OR EQUIVALENT.
   - C. T.I.G. WELDING TO BE USED AS MUCH AS POSSIBLE.
   - D. WHERE PLATES ARE USED TO FORM ANGLES, WELDS ARE TO BE FULL PENETRATION WITH PASS ON BACK SIDE.



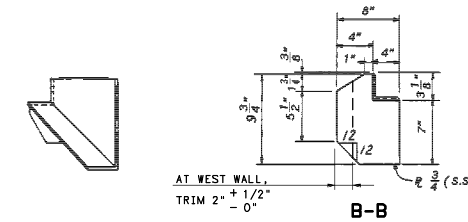
SECTION A-A
AIR SUPPLY
Scale 6"=1'-0"



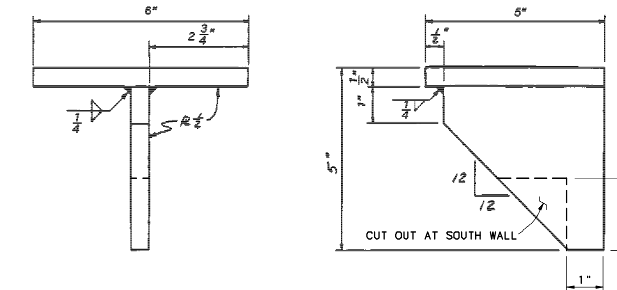
SLING STORAGE HOOK
2 REQ'D - STAINLESS STEEL - MK 8
Scale 3"=1'-0"



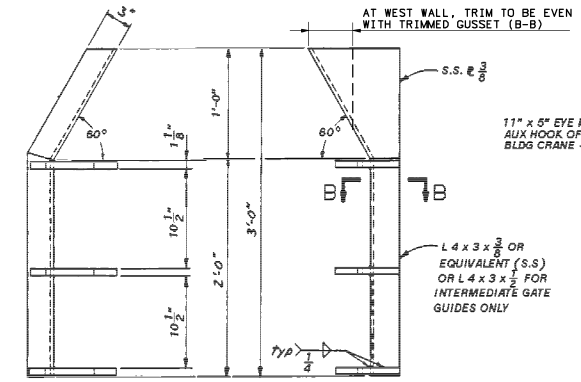
SEAL
3 REQD - MK 1
SCALE 3/4" = 1'-0"



B-B

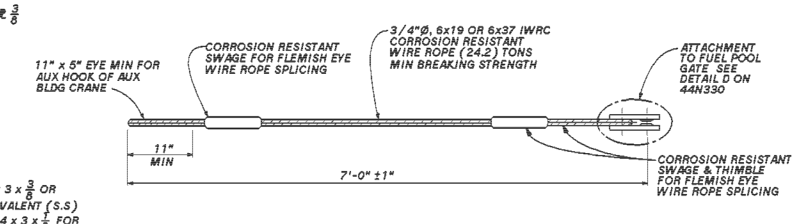


GATE SUPPORT
2 REQ'D - STAINLESS STEEL - MK 6
Scale 6"=1'-0"



UPPER & INTERMEDIATE GATE GUIDES
6 REQD - AS SHOWN - MK 2
6 REQD - OPPOSITE HAND - MK 3
STAINLESS STEEL

NOTE:
THE TRIMMING OF THE GUSSET PLATES IN SECTION B-B ABOVE INCLUDES ALL THE GUSSET PLATES ASSOCIATED WITH THE (6) UPPER AND INTERMEDIATE GATE GUIDES ON THE CASK LOADING PIT GATE ON THE WEST WALL OF THE SPENT FUEL POOL. REF (3) MK 2 AND (3) MK 3.



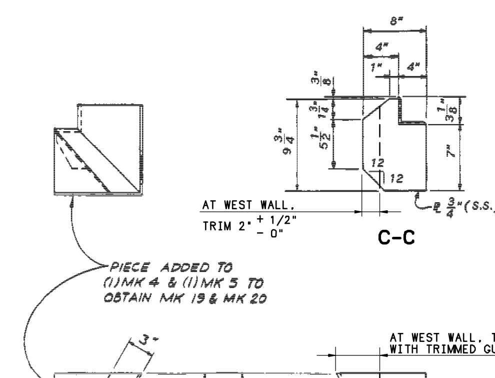
ASSEMBLY
44N332 MK-7
(2 REQUIRED)

PARTS
MK 1 - INFLATABLE SEAL PRS 582-6-4907-1 AS SHOWN ON PRESRAY DRAWING PR 4907RI, MINIMUM BURST PRESSURE OF 150 PSI - 3 REQD.
MK 2-8,17-20 - AS DETAILED ON THIS DRAWING.
MK 9-17 - MATERIAL TO BE ASTM A376, TYPE 304 (18-8) OR APPROVED EQUAL.
* MK 9 - 3/4" PIPE - 12' REQD.
* MK 10 - 3/4" PIPE NIPPLE - 4 REQD.
MK 11 - 1/2" MPT X 1/4" FPT BUSHING - 4 REQD.
MK 12 - 1/2" PIPE NIPPLE - 4 REQD.
MK 13 - 3/4" PIPE TEE - 4 REQD.
MK 14 - 3/4" PRESSURE RELIEF VALVE, GOOD FOR 200 PSI OPERATING PRESSURE, ADJUSTMENT RANGE TO INCLUDE 10-50 PSI - 4 REQD.
MK 15 - 3/4" SHUTOFF VALVE, GOOD FOR 200 PSI OPERATING PRESSURE - 6 REQD.
MK 16 - 3/4" PIPE PLUG - 4 REQD.

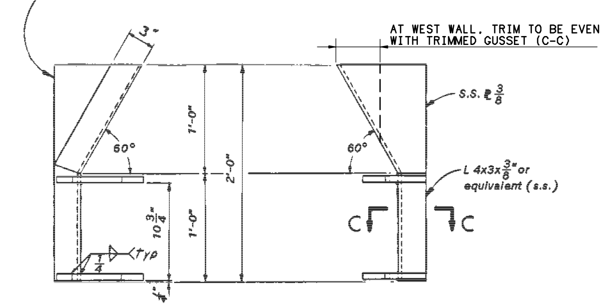
* 3/4" SCH. 80 PIPE ASME-SA312/376/304/304L

BY TVA FIELD
MK 7 - QUICK DISCONNECT WITH 3/4" IPT - 2 REQD.

MK 19 & 20 - LOWER GATE GUIDES - TVA FIELD ADD PIECE TO CONTRACTOR FURNISHED MKS 4 & 5 - 1 EA REQD.
MK 21 - 90° ELBOW - 1/2" PIPE - 4 REQD.
MK 22 - 3/4" X 1/2" HEX HEAD BUSHING - 4 REQD.
MK 23 - 3/4" SCREWED COUPLING - 4 REQD.
MK 24 - GUIDE BAR - 8 REQD.



C-C



LOWER GATE GUIDE
1 REQD - AS SHOWN - MK 19 - CASK LOADING PIT
1 REQD - OPPOSITE HAND - MK 20 - CASK LOADING PIT
1 REQD W/O ADDED PC - MK 4 - TRANSFER CANAL
1 REQD W/O ADDED PC - OPPOSITE HAND - MK 5 - TRANSFER CANAL

NOTE:
THE TRIMMING OF THE GUSSET PLATES IN SECTION C-C ABOVE INCLUDES ALL THE GUSSET PLATES ASSOCIATED WITH THE (2) LOWER GATE SUPPORT GUIDES ON THE CASK LOADING PIT GATE ON WEST WALL OF THE SPENT FUEL POOL. REF MK 19 & MK 20.

NOTES
1. FOR GENERAL NOTES SEE 44N330.
2. ALL MARK NUMBERS DETAILED ON THIS DRAWING TO HAVE PREFIX 44N332 UNLESS OTHERWISE NOTED.
3. SEAL TO BE TESTED AFTER ASSEMBLY TO A PRESSURE OF 40 PSI.
4. ALL SLINGS TO BE DESIGNED TO ANSI STANDARD B30.9-1971.

QUALITY ASSURANCE NOTES:
1. MK 7 TO BE PROCURED UNDER A QUALITY ASSURANCE PROGRAM AS DESCRIBED IN TVA REQUISITION.

Scale 1 1/2"=1'-0"
Except as noted

COMPANION DRAWINGS: 44N330, 331

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

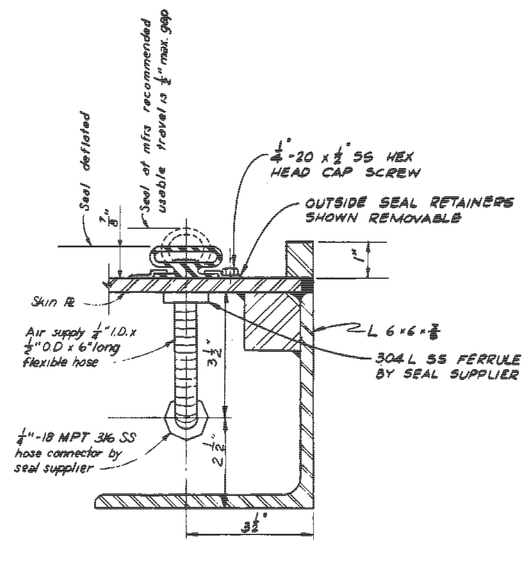
AUXILIARY BUILDING
UNIT 1
SPENT FUEL POOL
FUEL POOL GATES
SEAL DETAILS
TVA DWG NO. 44N332 RF
FIGURE 3.8.4-71

CAD MAINTAINED DRAWING

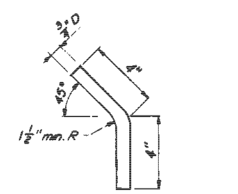
<u>FIELD NOTES:</u>

THE FOLLOWING MODIFICATIONS UNDER ECN 3475 WILL QUALIFY THE GATES FOR SEISMIC CATEGORY I (REF: WB-DC-40-43.)

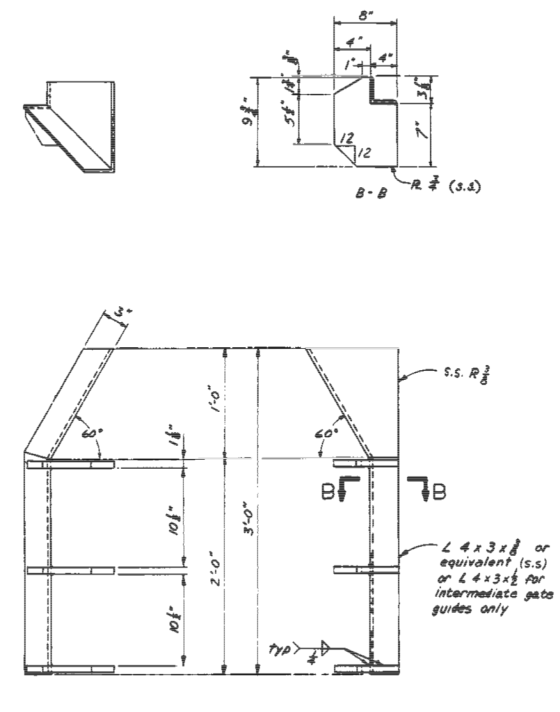
1. VERIFY THAT ALL PREVIOUS FIELD WORK (WELDING, MATERIALS ETC.) IS IN ACCORDANCE WITH DESIGN DRAWINGS. IF A NONCONFORMANCE EXISTS IT MUST BE CORRECTED.
2. REMOVE THE EXISTING MIDDLE GATE RESTRAINT GUIDE VANES FROM GATES.
3. ADD TWO INTERMEDIATE GUIDE VANES TO EACH SIDE OF THE GATES AND FIVE ADDITIONAL CROSS MEMBERS TO EACH GATE AS SHOWN IN THE POOL SIDE ELEVATION ON DRAWING 44N331.
4. ADD STIFFENER BARS TO THE INSIDE CORNER OF THE GATES OPPOSITE EACH GUIDE VANE (INCLUDING THE UPPER AND LOWER GUIDE VANES) AS SHOWN IN H-H ON 44N331.
5. ADD INTERMEDIATE GATE GUIDES TO THE WALLS EIGHT PLACES TO MATE WITH THE INTERMEDIATE GUIDE VANES WHEN THE GATES ARE IN THE OPERATING POSITION.
6. ALL MATERIALS AND WELDING TO BE IN ACCORDANCE WITH QUALITY LEVEL II OF CONSTRUCTION SPECIFICATION N3G-881.
   A. ALL MATERIALS TO BE ASTM A-276, A240, OR A167; TYPE 304 STAINLESS STEEL; COLD REDUCER; AND ANNEALED.
   B. WELD ELECTRODES TO BE AWS A5.4, E308 SERIES OR EQUIVALENT.
   C. T.I.G. WELDING TO BE USED AS MUCH AS POSSIBLE.
   D. WHERE PLATES ARE USED TO FORM ANGLES, WELDS ARE TO BE FULL PENETRATION WITH PASS ON BACK SIDE.



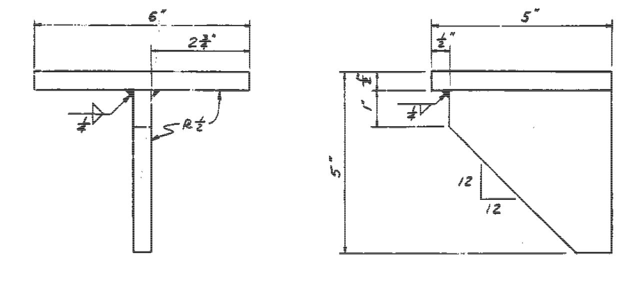
SECTION A-A
AIR SUPPLY
Scale 6"=1'-0"



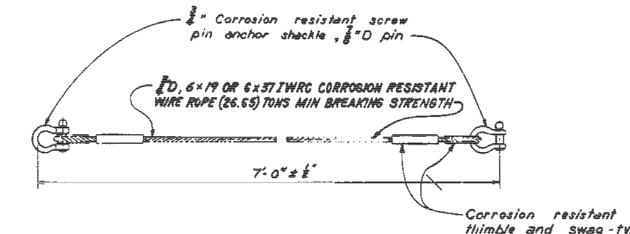
SLING STORAGE HOOK
2 REQ'D - STAINLESS STEEL - MK 8
Scale 3"=1'-0"



UPPER & INTERMEDIATE GATE GUIDES
6 REQD - AS SHOWN - MK 2
6 REQD - OPPOSITE HAND - MK 3
STAINLESS STEEL



GATE SUPPORT
2 REQ'D - STAINLESS STEEL - MK 6
Scale 6"=1'-0"



SLING
2 REQD - MK 7

<u>PARTS</u>

MK 1 - INFLATABLE SEAL PRS 582-6-4907-1 AS SHOWN ON PRESRAY DRAWING PR 4907R1, MINIMUM BURST PRESSURE OF 150 PSI - 3 REQD.
MK 2-8, 17-20 - AS DETAILED ON THIS DRAWING.
MK 9-17 - MATERIAL TO BE ASTM A376, TYPE 304 (18-8) OR APPROVED EQUAL.
MK 9 - 3/4" PIPE - 12' REQD.
MK 10 - 3/4" PIPE NIPPLE - 4 REQD.
MK 11 - 1/2" MPT X 1/4" FPT BUSHING - 4 REQD.
MK 12 - 1/2" PIPE NIPPLE - 4 REQD.
MK 13 - 3/4" PIPE TEE - 4 REQD.
MK 14 - 3/4" PRESSURE RELIEF VALVE, GOOD FOR 200 PSI OPERATING PRESSURE, ADJUSTMENT RANGE TO INCLUDE 10-50 PSI - 4 REQD.
MK 15 - 3/4" SHUTOFF VALVE, GOOD FOR 200 PSI OPERATING PRESSURE - 6 REQD.
MK 16 - 3/4" PIPE PLUG - 4 REQD.

<u>BY TVA FIELD</u>

MK 7 - QUICK DISCONNECT WITH 3/4" IPT - 2 REQD.

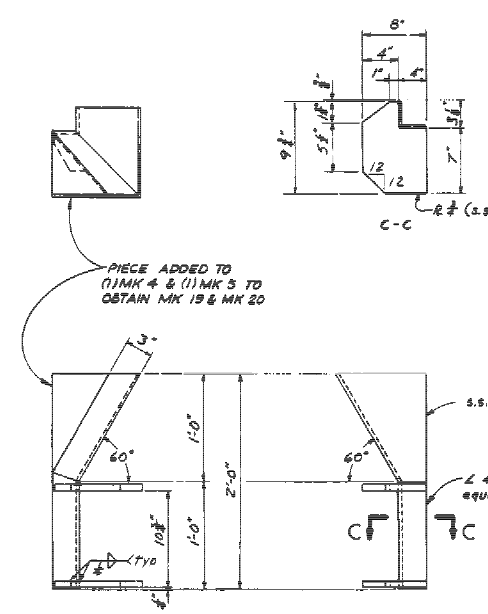
MK 19 & 20 - LOWER GATE GUIDES - TVA FIELD ADD PIECE TO CONTRACTOR FURNISHED MKS 4 & 5 - 1 EA REQD.
MK 21 - 90° ELBOW - 1/2" PIPE - 4 REQD.
MK 22 - 3/4" X 1/2" HEX HEAD BUSHING - 4 REQD.
MK 23 - 3/4" SCREWED COUPLING - 4 REQD.
MK 24 - GUIDE BAR - 8 REQD.

<u>NOTES</u>

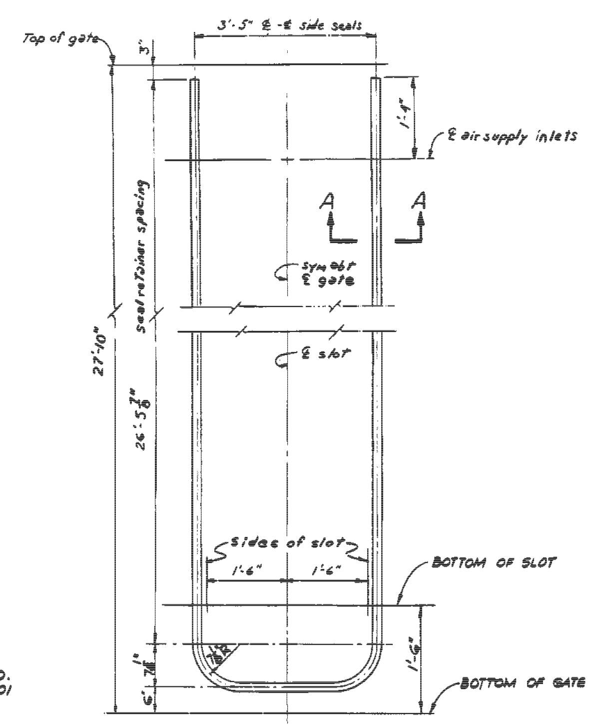
1. FOR GENERAL NOTES SEE 44N330.
2. ALL MARK NUMBERS DETAILED ON THIS DRAWING TO HAVE PREFIX 44N332 UNLESS OTHERWISE NOTED.
3. SEAL TO BE TESTED AFTER ASSEMBLY TO A PRESSURE OF 40 PSI.
4. ALL SLINGS TO BE DESIGNED TO ANSI STANDARD B30.9-1971.

<u>QUALITY ASSURANCE NOTES:</u>

1. MK 7 TO BE PROCURED UNDER A QUALITY ASSURANCE PROGRAM AS DESCRIBED IN TVA REQUISITION.



LOWER GATE GUIDE
1 REQD - AS SHOWN - MK 19 - CASK LOADING PIT
1 REQD - OPPOSITE HAND - MK 20 - CASK LOADING PIT
1 REQD W/O ADDED PC - MK 4 - TRANSFER CANAL
1 REQD W/O ADDED PC - OPPOSITE HAND - MK 5 - TRANSFER CANAL



CONTRACT NO. 76K70-820G01 (PENTE INDS.)

SEAL
3 REQD - MK 1
Scale 1/4"=1'-0"

COMPANION DRAWINGS: 44N330, 331

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

AUXILIARY BUILDING
UNIT 2
SPENT FUEL POOL
FUEL POOL GATES
SEAL DETAILS
TVA DWG NO. 2-44N332 R0
FIGURE 3.8.4-71(U2)

Figure 3.8.4-72 THRU Figure 3.8.4-80

Deleted

3.8.5.1.2 Foundations of Other Category I Structures

Auxiliary-Control Building and Associate Structure

All of the Auxiliary-Control Building, except the waste packaging structure, and the condensate demineralizer waste evaporator structure portion is supported by a reinforced concrete slab placed on a 4-inch-minimum-thick concrete subpour which caps the top of the irregular rock surface.

The Auxiliary Building portion of the base slab is 7 feet thick while the control bay portion is 5 feet thick. The entire base slab is located on three different levels with continuity between these levels being provided through thick walls. The thicknesses of the slab were selected primarily to provide sufficient rigidity to minimize differential vertical movements of columns and walls and secondarily to reduce shearing stresses in the slab itself. Due to the thickness of the slab, anchorage into rock was not required to resist hydrostatic up-lift pressures from maximum flood conditions.

The waste packaging structure is separated from the rest of the Auxiliary Building by 2 inches of fiberglass expansion joint material. The 45-inch-thick base slab at grade Elevation 728 is supported below Elevation 725.25 by crushed stone backfill placed in 4-inch layers and compacted to a minimum of 70% relative density.

The base slab of the condensate demineralizer waste evaporator structure is 2-feet, 9-inches thick, except for the access tunnel part of the building which is 2-feet, 3-inches thick. The structure is supported on H-bearing piles. The access tunnel is separated from the rest of the Auxiliary Building by two inches of fiberglass expansion joint material.

Intake Pumping Station

The intake structure is supported by a reinforced concrete slab placed on a 4-inch-minimum-thick concrete subpour which caps the top of the irregular rock surface. The base slab is 4 feet thick with a 6-foot-wide by 10-foot deep key located at the back of the structure. This key extends the full width of the structure. The base slab extends 10 feet past the back wall and has two areas of 26 feet by 29 feet on each side that extend beyond the walls.

The concrete retaining walls at the intake structure are designed to protect the forebay of the intake against earth slides during an earthquake. The base slabs of these cantilevered walls are keyed into rock. The walls are separated from the structure with expansion joint material.

North Steam Valve Room

The north steam valve room is supported by a grillage of reinforced concrete foundation walls to base rock. These walls span vertically from base rock at Elevation 683.0 to the bottom of the valve room base slab at Elevation 722.0. There are four 4-foot thick walls running in a north-south direction and these walls are tied together by a singular 4-foot thick wall running in an east-west direction. Three closed cells are formed by these walls in combination with the Reactor Building wall. These closed cells are backfilled with a non-compacted crushed stone. The valve room foundation walls are separated from the Reactor Building foundation and wall by a 2-inch fiberglass expansion joint material.

Diesel Generator Building

The base slab of the Diesel Generator Building is discussed in Section 3.8.5.5.2. Based on soils laboratory tests, it could not be assured that the existing material between the top of firm gravel at Elevation 713 and base slab was capable of safely supporting the structure. Therefore, this material was removed and replaced with crushed stone fill placed in 4-inch layers and compacted to a minimum of 70% relative density (see Section 2.5.4.5.2, historical information). A slope stability analysis was performed in order to assure stability of the slope below the building.

Refueling Water Storage Tank

The refueling water storage tank foundation is a solid, circular reinforced concrete structure placed on engineered granular fill over firm natural granular soil. The foundation is constructed with shear keys to prevent sliding displacement and with retaining walls to contain a reservoir of borated water after a postulated rupture of the storage tank. The foundation is protected from missiles by a concrete apron.

Discharge Overflow Structure

See Section 3.8.4.1.7 for a description of the discharge overflow structure foundation.

Class 1E Electrical System Manholes and Duct Banks

The manholes and a portion of the duct banks are supported on in-situ soil. The duct banks at the intake pumping station are supported on in-situ soil, piles, and a bracket on the pumping station wall, see Section 3.8.4.1.4 for additional information.

ERCW Standpipe Structures

See Section 3.8.4.1.7 for the standpipe structures.

ERCW Pipe Supporting Slabs and Beams

See Section 3.8.4.1.7 for a description of the beams and slab.

ERCW Valve Covers

See Section 3.8.4.1.7 for a description of these structures.

Additional Diesel Generator Building

The base slab of the additional Diesel Generator Building is discussed in Section 3.8.4.4.8. Similar to the Diesel Generator Building, it could not be assured that the existing soil between the top of firm gravel at Elevation 713.0 and the bottom of the base slab at Elevation 730.0 could safely support this structure. Therefore, the building was supported on end bearing steel H-Piles driven to refusal in sound rock or other suitable material. For additional information on this structure, see Section 3.8.4.1.8.

3.8.5.2 Applicable Codes, Standards, and Specifications

See Sections 3.8.1.2, 3.8.3.2, and 3.8.4.2.

3.8.5.3 Loads and Loading Combinations

The loads and loading combinations are described in Sections 3.8.1.3, 3.8.3.3, and 3.8.4.3. For loads and loading combinations on the Additional Diesel Generator Building, see Table 3.8.4-22.

3.8.5.4 Design and Analysis Procedure

3.8.5.4.1 Primary Containment Foundation

The foundation was analyzed as a slab on a rigid foundation. The slab was analyzed using computer code Gendek 3 Finite Element Analysis of Stiffened Plates.

Maximum tangential and radial moments were obtained using the finite element analysis of the various load combinations. Shear stresses were obtained by conventional analysis for the containment vessel anchorage and major equipment loadings.

3.8.5.4.2 Auxiliary-Control Building

The reinforced concrete base slab of the Auxiliary-Control Building was designed in compliance with the ACI Building Code 318-63. It was analyzed by the ICES STRUDL-II finite element method as a slab on an elastic foundation. In the ICES STRUDL-II program the foundation material was modeled by assigning a vertical spring to each node of the grid system which was used to represent the base slab. The base slab was divided into elements with wall stiffnesses being recognized by introducing flexural rigidity along the wall and torsional rigidity being recognized by including a rotational spring. Superposition of the various loading conditions were used to obtain maximum stresses. Manual calculations gave results for the bending moments which checked reasonably close with those obtained from the ICES STRUDL-II analysis. A standard frame analysis was also performed in order to determine the shearing forces in the slab.

Shear walls fixed to the base slab transmit lateral force to the slab; the base slab itself is keyed and anchored into foundation rock to transmit shear from the structure into the rock.

The 45-inch-thick slab of the waste packaging area was designed for a uniform distribution of base pressure to span as a flat plate between the load bearing walls. Walls were thicker than necessary for structural purposes because of shielding requirements.

The base slab of the condensate demineralizer waste evaporator building portion was designed as a pile supported foundation. Batter piles were used around the perimeter of the structure to transmit lateral loads from the structure to the foundation media.

3.8.5.4.3 Intake Pumping Station

The design of the base slab was controlled for the most part by uplift considerations under assumed unwatered conditions with one bay dry and full uplift over 100% of the area between the slab and the base rock. The backfilled portion of the base slab was controlled by the load from the saturated fill.

3.8.5.4.4 Soil-Supported Structures

A uniform or linear distribution of base pressure was assumed in the design of all soil-supported structures and all base slabs were essentially designed as flat plates.

3.8.5.4.5 Pile Supported Structures

Pile supported structures were designed using conventional frame analysis or through the use of ICES STRUDL-II finite element computer program.

3.8.5.5 Structural Acceptance Criteria

3.8.5.5.1 Primary Containment Foundation

The base slab design contained the following conservative features:

1. No allowance was made for the additional spread of reactions under the walls or the additional section modulus due to the 3-foot structural fill over the base slab.

2. In the outer area of the slab, where the additional depth is in excess of the 2-foot, 8-inch recess in the upper surface, no allowance has been made for the additional thickness which increases the stiffness of the slab and thus lowers the stresses.

3.8.5.5.2 Foundations of Other Category I Structures Auxiliary-Control Building

The base slab as designed has its maximum flexural stresses and shearing stresses within the allowable working stress design limits of Table 3.8.4-1 for all loading combinations. Design Case I (dead load plus live load), which generally controlled the design, was investigated by the ICES STRUDL-II program for several loading conditions created by the three different levels of the slab and by the early conditions were superimposed in various combinations to ensure that the slab was designed for the maximum possible stresses. The maximum calculated compression of the base slab was approximately 12 ksf. The maximum allowable compression on rock is 26 ksf (180 psi). In probable maximum flood conditions, with the dead load of the structure alone assumed to resist the buoyant force, the factor of safety against floatation is 1.51.

Intake Pumping Station

The base slab of the intake pumping station serves as a water barrier under maintenance conditions with one bay unwatered. It also adds to the stability of the structure. Backfill on the extended areas of the slab add weight to the structure and the key provides resistance to sliding. The maximum calculated compression on the base slab was approximately 12 ksf. The maximum allowable compression on rock is 26 ksf (180 psi).

North Steam Valve Room

The valve room foundation walls were designed to resist the maximum overturning effect on the building. This effect was due to pressure as the result of the rupture of a main steam pipe, its associated jet impingement load, and the Safe Shutdown Earthquake. This resistance to overturning was obtained by converting the maximum overturning moment on the structure into a resisting active soil pressure on the foundation walls. For overturning in the east-west direction, four of the foundation walls were considered effective. For overturning in the north-south direction, the singular cross-wall was considered to be resisting the overturning. Using this pressure as a load on the walls, they were modeled as plate structures utilizing the STRUDL-II Finite Element computer program. The walls were considered to span between bedrock, the bottom of the valve room base slab and other foundation walls framing into them.

Waste Packaging Structure

This structure is situated on well-compacted crushed stone backfill above rock and was designed for a normal allowable uniform bearing pressure of 6.5 ksf and a maximum allowable pressure with 70% or more of the base in compression of 10 ksf under maximum overturning forces. Actual calculated bearing pressures were 1.4 ksf for uniform loading and 6.7 ksf with 72% of the base in compression for maximum overturning forces.

Diesel Generator Building

The structure is situated as described in Section 3.8.5.1.2. The base slab of the Diesel Generator Building is 9 feet 9 inches thick founded on crushed stone backfill and located above the probable maximum flood elevation. The structure was designed for a normal allowable uniform bearing pressure of 6.5 ksf and a maximum allowable pressure of 11.5 ksf under maximum overturning forces. Actual calculated bearing pressures for the Diesel Generator Building were 2.0 ksf for uniform loading and 4.9 ksf for maximum overturning forces with 100% of the base in compression.

Additional Diesel Generator Building

For discussions on this pile supported structure see Section 3.8.4.4.8. Also, rotational restraint from the piles was not considered due to the large difference in stiffness between the 12 foot thickness of the base slab and that of the steel H-piles.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

General

See Section 3.8.1.6.

3.8.5.6.1 Materials

Concrete and Reinforcing Steel

See Section 3.8.1.6.1

Backfill Materials

Backfill material was taken only from areas designated by the soils investigation program (see Section 2.5.4.5.2, historical information) as suitable for backfill material.

3.8.5.6.2 Quality Control

Concrete and Reinforcing Steel

Concrete production and testing were as in Section 3.8.1.6.2, except some concrete used to protect rock surfaces was purchased as ready mix in conformance with ASTM C94-69.

The protective concrete for rock surfaces was specified as 2,000 psi at 90 days age. It was in conformance to specifications.

The Shield Building base slab and the north steam valve rooms foundation walls used concrete specified as 5,000 psi at 90 days.

Some concrete did not meet specification requirements. This was evaluated and documented in the Report CEB-86-19C "Concrete Quality Evaluation". Results have been documented in affected calculation packages and drawings.

Testing of reinforcing steel was as in Section 3.8.1.6.2.

<u>Base Rock</u>

The base area of all rock-supported structures was inspected by the principal civil design engineer in conjunction with an experienced TVA geologist during final cleanup of rock surfaces to determine its suitability as a foundation.

<u>Backfill</u>

Quality control requirements for backfill material were as specified in Section 2.5.4.5 (historical information).

3.8.5.6.3 <u>Special Construction Techniques</u>

No special construction techniques were used.

REFERENCES

None

3.8.6 Category I(L) Cranes

3.8.6.1 Polar Cranes

3.8.6.1.1 Description

See Figures 3.8.6-1 through 3.8.6-6.

There are two polar cranes, one in each of the Reactor Buildings. Each crane is a single two-part trolley, overhead, electric traveling type; operating on an 86-foot 0-inch-diameter rail at the top of the crane wall and above the reactor. Each crane has a main hoist capacity of 175 tons and an auxiliary hoist capacity of 35 tons.

The Unit 1 polar crane main and auxiliary hoist motions are driven by dc motors with stepless regulated speed control. The bridge and trolley are driven by ac motors with static, stepless regulated speed control.

The Unit 2 polar crane main and auxiliary hoist motions are driven by ac motors with Variable Frequency Drives. The bridge and trolley are driven by ac motors with variable Frequency Drives.

Structural portions of the crane bridges consist of welded box-type girders and welded, haunched, box-type end ties. Structural portions of the trolleys consist of welded box-type trucks and welded cross beams.

Control of each crane is from a cab located below the bridge walkway at one end of a girder.

3.8.6.1.2 Applicable Codes, Standards, and Specifications

The following codes, standards, and specifications were used in the design of the cranes:

National Electric Code, 1971 edition.

National Electrical Manufacturers Association, Motor and Generator Standards, Standard MG-1, 1970 edition.

Crane Manufacturers Association of American, Inc., Specification #70, 1970 edition.

Federal Specification RR-W-410C

American Society for Testing and Materials, 'Material Standards,' 1974 edition.

American Welding Society, D1.1-72 with 1973 Revisions, Structural Welding Code.

Section 1.23, Part 1, 'Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings,' Manual of Steel Construction, Part 5, American Institute for Steel Construction, 7th edition, 1970.

American Gear Manufacturers Association Standards for Spur, Helical, Herringbone, and Beval Gears.

Where date of edition, copyright, or addendum is specified, earlier versions of the listed documents were not used. In some instances, later revisions of the listed documents were used where design safety was not compromised.

The cranes meet applicable requirements of the listed codes, standards, and specifications.

3.8.6.1.3 Loads, Loading Combinations, and Allowable Stresses

Loads, loading combinations, and allowable stresses are shown in Table 3.8.6-1.

3.8.6.1.4 Design and Analysis Procedure

The bridge girders and end ties for each crane were designed as simple beams in the vertical plane and as a continuous frame in the horizontal plane. Stresses in the girders and end ties were computed with the trolley positioned to produce maximum stresses. Seismic restraints are located on the bottom of each girder and these restraints are designed to withstand seismically applied loads to ensure the crane will not fall during an earthquake.

Trolley positions used were the maximum end position, third point, and the point near the center which produces maximum bending moments.

Trolley members were designed as simple beams. Design of the bridge girders and end ties was by TVA. Mechanical parts and structural members except the bridge girders and end ties were designed by the contractor. Calculations and designs made by the contractor were reviewed by TVA design engineers.

In designing for earthquake conditions, forces due to accelerations at the crane bridge rails were used as static loads for determining component and member sizes. After establishing component and member sizes, a dynamic analysis, using appropriate response spectra, was made of the total crane to determine that allowable stresses had not been exceeded.

Earthquake accelerations at the bridge rails were determined by dynamic analysis of the structures supporting the crane rails.

The polar crane was also evaluated for seismic loads based on the Set B seismic response spectra using 2% damping for OBE and 4% damping for SSE. The polar crane was initially evaluated for seismic loads based upon Set A seismic response spectra.

3.8.6.1.5 Structural Acceptance Criteria

Allowable stresses for all load combinations used for the various crane parts are given in Table 3.8.6-1. For normal load conditions, the allowable stresses provide safety factors of 2 to 1 on yield for structural parts and 5 to 1 on ultimate for mechanical parts, except for wire ropes which have a minimum safety factor of 5 to 1 on ultimate. For limiting conditions, such as an SSE earthquake or stall, stresses do not exceed .9 yield.

3.8.6.1.6 Materials, Quality Controls, and Special Construction Techniques

A36 steel was used for the major structural portions of the crane. Design by TVA and erection by TVA were in accordance with TVA's quality assurance program. Design and fabrication by the contractor were in accordance with the contractor's quality assurance program which was reviewed and approved by TVA's design engineers. The contractor's quality assurance program covers the criteria in Appendix B of 10 CFR 50. Fabrication procedures such as welding, stress relieving, and nondestructive testing were included in appendices to the contractor's quality assurance program.

ASTM standards were used for all material specifications and certified mill test reports were provided by the contractor for materials used for all load-carrying members.

This crane is covered by TVA's Augmented Quality Assurance Program for Seismic Category I(L) Structures.

3.8.6.1.7 Testing and In-service Surveillance Requirements

Refer to Section 14.2.7 (historical information), Paragraph 4.A.1.h for Initial Testing.

After the initial test, periodic visual inspections of each crane are to be made. Parts inspected during the visual inspection are to include all bolted parts, couplings, brakes, hoist ropes, hoist blocks, limit switches, and equalizer systems.

3.8.6.1.8 Safety Features

The cranes were designed to withstand an SSE and to maintain any load up to rated capacity during and after the earthquake period.

The bridges are equipped with double flange wheels, spring-set, electrically-released brakes which set and firmly lock two of the wheels when the bridge drive machinery is not operating or when power is lost for any reason, hold down lugs which run under the rail heads, and seismic restraints located on the bottom of each girder. During an earthquake the crane rail will yield before the crane wheels fail, thus allowing the crane to move until the seismic restraints on each girder contact the crane wall. These restraints hold the crane on the runway. Guide rollers, mounted on each extreme corner truck, travel against the outer surface of the bridge rail to assure bridge truck alignment.

The trolleys are each equipped with double flange wheels, spring-set, electrically-released brakes which set and firmly lock the driving wheels when the trolley drive machinery is not operating or when power is lost for any reason, and hold down lugs which run under the rail heads. Positive wheel and bumper stops are provided at both ends of the bridge. During an earthquake, the trolley could be displaced, but it will not leave its rails which are firmly attached to the bridge structure.

Safety features provided for each hoist include two independent gearing systems, connected by a cross shaft to prevent windup, two brakes with each of the brakes operating through one of the independent gearing systems, two upper travel limit switches, one lower travel limit switch, over-speed switches set to trip at 120% of maximum rated speed. The Unit 1 polar crane has emergency dynamic braking for controlled lowering in case of simultaneous failure of ac power source and holding brakes. For the Unit 2 polar crane, upon loss of AC power, load is manually lowered by manipulating the holding brakes. In addition, each hoist incorporates a symmetrical cross reeving system designed to hold the load level with either rope. Each hoist is also provided with a hydraulic equalizing system to prevent dropping the load and to limit shock loading in case of a single rope failure. Holding brakes for the hoists are the spring-set, electrically released type with provisions for manual release of the brakes. The capacity of each main hoist brake is sufficient to stop a 100% rated load traveling at the maximum rated hoisting speed within a distance of 6 inches.

Safety control features provided for all motions consist of overcurrent protection, undervoltage protection, control actuators which return to the stop position when released, and an emergency-stop push button.

3.8.6.2 Auxiliary Building Crane

3.8.6.2.1 Description

The crane in the Auxiliary Building is a single trolley, overhead, electric traveling type with a span of 77 feet. The crane has a main hoist capacity of 125 tons and an auxiliary hoist capacity of 10 tons. The main hoist meets NUREG-0554 single failure proof criteria for compliance with 10 CR 72.124(a), “Design for Criticality Safety” for handling spent fuel casks.

The main and auxiliary hoists are driven by AC Variable Frequency motor drives with eddy current lowering and stepless (infinite) speed control. The bridge and trolley travel motions are AC operated with stepless control.

Structural portions of the crane bridge consist of welded, box-type girders and welded, haunched, box-type end ties. Structural portions of the trolley consist of welded, box-type trucks and welded cross beams.

Control of the crane is from a control console in the operator cab which is located at mid-span of the crane beneath the south girder.

The one crane serves the needs of two reactor units. It handles the fuel casks, new fuel shipments to the new fuel storage, shield plugs at the equipment access doors, and any large pieces of equipment going into or out of the Reactor Buildings via the Auxiliary Building.

3.8.6.2.2 Applicable Codes, Standards, and Specifications

The following codes, standards, and specifications were used in the original design of the crane:

National Electric Code, 1971 Edition.

NEMA Standard MG1, 1970 Edition.

Crane Manufacturers Association of American, Inc., Specification No. 70, 1970 Edition.

Federal Specification RR-W-410C. Auxiliary (10 Ton) Hoist Only

ASTM Material Standards, 1974 Edition.

AWS, D1.1-72 with 1973 Revisions, Structural Welding Code.

Section 1.23, Part I, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings; AISC Manual of Steel Construction, 7th Edition, 1970.

American Gear Manufacturers Association Standards for Spur, Helical, Herringbone, and Bevel Gears.

DIN EN (10264-2/12385-2), “Steel Wire and Wire Products- Steel Wire for Ropes, Part 2/Steel Wire Ropes - Safety - Part 2 (Main Hoist Wire Rope only)

The following codes and Standards were used to qualify the 125 Ton Auxiliary Building Crane Main Hoist as NUREG-0554 single-failure-proof:

CMAA #70 (2010)
NUREG-0554
NUREG-0612

Where date of edition, copyright, or addendum is specified, earlier versions of the listed documents were not used. In some instances, later revisions of the listed documents were used where design safety was not compromised.

The cranes meet applicable requirements of the listed codes, standards, and specifications.

3.8.6.2.3 Loads, Loading Combinations, and Allowable Stresses

Loads, loading combinations, and allowable stresses are shown in Table 3.8.6-2.

3.8.6.2.4 Design and Analysis Procedure

The 125 Ton Auxiliary Building Bridge Crane is designed to retain control of and hold the load, and the bridge and trolley are designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event. The Maximum Critical Load (125 Tons) plus operational and seismically induced pendulum and swinging load effects on the crane were considered in the design of the trolley, and they were added to the trolley weight for the design of the bridge.

The crane was analyzed with the trolley located at mid-span, one-third span, close approach, and with the hook in the high and low position at each location. These analyses were competed using 4% acceleration response spectra which were conservatively derived using the average of the 3% and 5% response spectra.

It was determined from this analysis that all structural components analyzed are capable of withstanding the seismically induced loadings.

The Auxiliary Building 125/10 Ton Crane Main Hoist is designed to NUREG-0554 single-failure-proof requirements. The head and load block, reeving, and hook are designed to support a static load of 200% of the Maximum Critical Load (MCL) of 125 Tons. The main hoist drive train, bridge, and trolley components have a 15% margin relative to the maximum critical load. The auxiliary hoist is designed to CMAA #70 requirements.

3.8.6.2.5 Structural Acceptance Criteria

Calculated stresses for the crane are all within CMAA #70 (2010) limits. Additionally, the individual component parts of the vertical hoisting system components, which include the head block, rope reeving system, load block, and sister hook, are designed to support a static load of 200% of the Maximum Critical Load of 125 Tons for the main hook.

Drive train components for the main hoist, bridge drive and trolley drive, including requirements to deal with degradation due to wear and exposure.

3.8.6.2.6 Materials, Quality Controls, and Special Construction Techniques

ASTM A 36 steel was used for the major structural portions of the crane. Design by TVA and erection by TVA were in accordance with the TVA quality assurance program. Design and fabrication by the contractor were in accordance with the contractor's quality assurance program which was reviewed and approved by TVA's design engineers. The contractor quality assurance program covers the criteria in Appendix B of 10 CFR 50. Fabrication procedures such as welding, stress relieving, and nondestructive testing, were included in appendices to the contractor's quality assurance program.

ASTM standards were used for all material specifications and certified mill tests reports were provided by the contractor for materials used for all load-carrying members.

This crane is covered by TVA's Augmented Quality Assurance Program for Seismic Category I(L) Structures.

3.8.6.2.7 Testing and In-service Surveillance Requirements

Upon completion of erection and adjustments on the crane, all crane motions and operating parts were thoroughly tested with crane handling 125% of rated capacity. Tests were made to prove the ability of the crane to handle its rated capacity and smaller loads smoothly at any speed within the specified speed range. Each brake was tested to demonstrate its ability to hold the required load.

After the initial test, periodic visual inspections of the crane are to be made. Parts inspected during the visual inspection are to include all bolted parts, couplings, brakes, hoist ropes, hoist blocks, limit switches, and equalizer systems.

3.8.6.2.8 Safety Features

The crane was designed to withstand an SSE and to maintain any load up to rated capacity during and after the earthquake period.

The bridge is equipped with double flange wheels, hold down lugs which run under the rail heads two spring-set electrically released brakes which set and firmly lock the wheels when the bridge drive machinery is not operating or when power is lost for any reason. During an earthquake the crane rail will yield before failure of the crane wheels and allow the end ties to contact the adjacent concrete wall, thus restraining the crane and preventing it from falling. Positive wheel and bumper stops are provided at each end of the bridge travel.

The trolley is equipped with double flange wheels, two spring-set, electrically released brakes which set and firmly lock the driving wheels when the trolley drive machinery is not operating or when power is lost for any reason, and hold down lugs which run under the rail heads. Positive wheel and bumper stops are provided at both ends of the bridge. During an earthquake, the trolley could be displaced, but it will not leave the rails which are firmly attached to the bridge structure.

Safety features provided for each hoist include two independent gearing systems, connected by a cross shaft to prevent windup, two brakes with each of the brakes operating through one of the independent gearing systems, two upper traveling limit switches, one lower travel limit switch, over-speed switches set to trip at 120% of maximum rated speed, and emergency dynamic braking for controlled lowering in case of simultaneous failure of ac power source and holding brakes. In addition, the main hoist incorporates a symmetrical cross reeving system designed to hold the load level with either rope and to limit the shock loading in case of a single rope failure, and a hydraulic sheave equalizing system to prevent dropping the load and to limit shock loading in case of a single rope failure. The auxiliary hoist has a two-part whip-style reeving so that a single rope failure will not drop the load. Holding brakes for the hoists are the spring-set, electrically released type with provisions for manual release of the brakes. The capacity of each main hoist brake is sufficient to stop at 100% rated load traveling at the maximum rated hoisting speed within a distance of 6 inches.

The interlocks will not be bypassed for any heavy loads except the fuel transfer gates and for new fuel handling. All loads in excess of 2,059 lbs, or which would have a kinetic energy greater than that of a spent fuel assembly from its normal handling height, will be transported around the spent fuel pit, rather than over, with the interlocks activated, via the normal paths used for heavy loads.

The Main Hoist of the 125/10 Ton Auxiliary Building Crane meets NUREG-0554 requirements as single-failure-proof.

Safety control features provided for all motions consist of overcurrent protection, undervoltage protection, control actuators which return to the stop position when released, and an emergency-stop pushbutton.

The electrical interlocks and mechanical stops will be administratively bypassed to allow use of the crane for handling the fuel transfer canal gate. The bypass is accomplished by means of a keyed switch, operation of which bypasses all interlocks controlling crane movements and activates a green indicating light located beneath the operator's cab. The indicating light is visible from any point on the operating floor. Control of the bypass key by administrative personnel and the ability of administrative personnel to stop the crane by means of any one of three pushbutton stations ensure that administrative personnel control all bypass operations.

Two pushbutton stations are located on the west wall and one pushbutton station is located on the east wall of the Auxiliary Building about four feet above the Elevation 757.0 operating floor. These stations are readily accessible to administrative personnel on the operating floor.

Testing of bypass interlocks is accomplished on a periodic basis in accordance with approved WBNP surveillance instructions. Testing must occur within seven calendar days prior to initial use, and every seven calendar days during continued regular usage. Each limit switch is manually operated to ascertain proper functioning of interlock circuits. To verify that the interlock system is functioning properly, each limit switch is moved to its actuated position, and all affected crane controls operated to ensure that crane movement does not occur.

REFERENCES

None

TABLE 3.8.6-1 (Sheet 1 of 3)

POLAR CRANES
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES

No.	Load Combinations	Allowable Stresses (psi)		
		Tension	Compression[1]	Shear
	Bridge Structure			
I	Dead Live Impact Trolley tractive	$0.50\ F_y$	$0.48\ F_y$	$0.33\ F_y$
II	Dead Live Impact Bridge tractive	$0.50\ F_y$	$0.48\ F_y$	$0.33\ F_y$
III	Dead Live Trolley collision	$0.62\ F_y$	$0.59\ F_y$	$0.41\ F_y$
IV	Dead Trolley weight Stall at 275% capacity	$0.90\ F_y$	$0.90\ F_y$	$0.50\ F_y$
V	Dead Live at 100% capacity SSE	$0.90\ F_y$	$0.90\ F_y$	$0.50\ F_y$

TABLE 3.8.6-1 (Sheet 2 of 3)

POLAR CRANES
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

No.	Load Combinations	Allowable Stresses (psi) Tension	Compression[1]	Shear
	Trolley Structure			
I	Dead Live Impact	$0.5\ F_Y$	$0.48\ F_Y$	$0.33\ F_Y$
II	Dead Stall at 275% capacity	$0.9\ F_Y$[3] $0.62\ F_Y$[2]	$0.9\ F_Y$ $0.59\ F_Y$	$0.5\ F_Y$ $0.41\ F_Y$
III	Same as case V for bridge			

Mechanical Parts

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[1]	Shear
	Parts Other Than Wheel Axles and Saddle Truck Connecting Pins		
I	Dead Live	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
II	Dead Stall at 275% capacity	$0.9\ F_Y$	$0.5\ F_Y$
	Wheel Axles and Connecting Pins		
I	Dead Live Impact	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$
II	Dead Live Collision	$\frac{Ult}{5}$	$\frac{2 \times Ult}{15}$

TABLE 3.8.6-1 (Sheet 3 of 3)

POLAR CRANES
LOADS, LOADING COMBINATIONS, AND ALLOWABLE STRESSES (Cont'd)

No.	Load Combinations	Allowable Stresses (psi) Tension and Compression[1]	Shear
	Wheel Axles and Connecting Pins (Continued)		
III	Dead Stall at 275% capacity	0.40 F_Y	0.50 F_Y
IV	Dead Live at 100% capacity SSE	0.90 F_Y	0.50 F_Y

Notes:

(1) The value given for allowable compression stress is the maximum value permitted, when buckling does not control. The critical buckling stress, F_{cr}, shall be used in place of F_Y when buckling controls.

(2) For sheave frames, cross beams, and their respective connections.

(3) For all other members.

WBN-1

TABLE 3.8.6-2

DELETED

APPENDIX 3.8A

SHELL TEMPERATURE TRANSIENTS

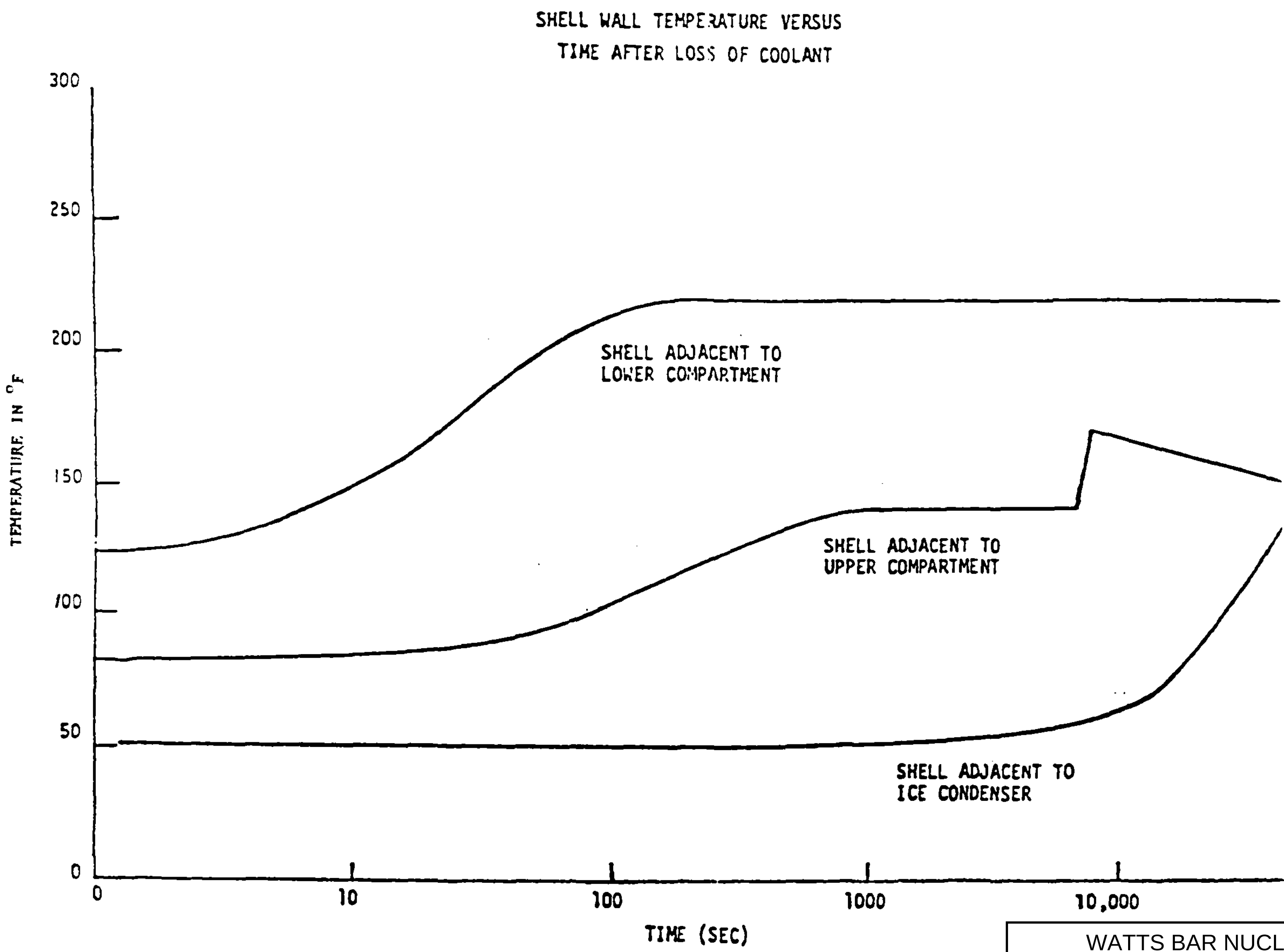
Figure 3.8A-1 presents average shell temperatures adjacent to the three compartments as a function of time after the DBA. The DBA is a double end rupture of the reactor coolant pipe with the reactor decay heat released into the lower compartment as steam. Initially the steam is condensed in the ice compartment. After the ice melts, the steam is condensed in the upper compartment by a water spray.

The lower compartment temperature rises to 250°F, essentially instantaneously, then is reduced to 220°F very shortly after the blowdown is completed. The blowdown is completed before the shell adjacent to the lower compartment reaches 220°F, as illustrated by the smooth curve presented in Figure 3.8A-1.

The upper compartment temperature rises essentially instantaneously due to compression of the noncondensable gases into the upper compartment. The sharp rise at 7,000 seconds simulates the disappearance of the ice from the ice compartment. The shell temperature will rise at a maximum of 0.11 degree per second during the rise from 140°F to 190°F. The subsequent temperature decrease of the shell adjacent to the upper compartment is due to the reduction in decay heat.

The curve labeled shell adjacent to the ice compartment indicates the temperature of the shell adjacent to the ice compartment. The shell is separated from the ice compartment with a thick layer of insulation, hence the rather slow response for the temperature of the shell adjacent to the ice compartment. After the ice is all melted the temperature inside the ice compartment will be the same as the temperature in the lower compartment; however, the shell temperature adjacent to the ice compartment will always be less than the temperature in the ice compartment because of insulation. The temperature of the shell adjacent to the ice compartment will peak at less than 220°F.

The curves in Figure 3.8A-1 are an average shell temperature representative for the bulk of the shell. Some areas near boundaries between compartments and near the base will differ significantly from the bulk. The lower portion of the lower compartment shell will be insulated for the purpose of minimizing the transient effects. Figure 3.8A-2 is a plot of shell temperature versus distance above Elevation 702.78 for various times after a LOCA. In establishing these curves it was assumed that top of the concrete slab is at Elevation 702.78 inches, and that the top of the insulation is at Elevation 707.11, and the top 8 inches of insulation is tapered from 2 inches thick to 1/4-inch thick.



WATTS BAR NUCLEAR PLANT
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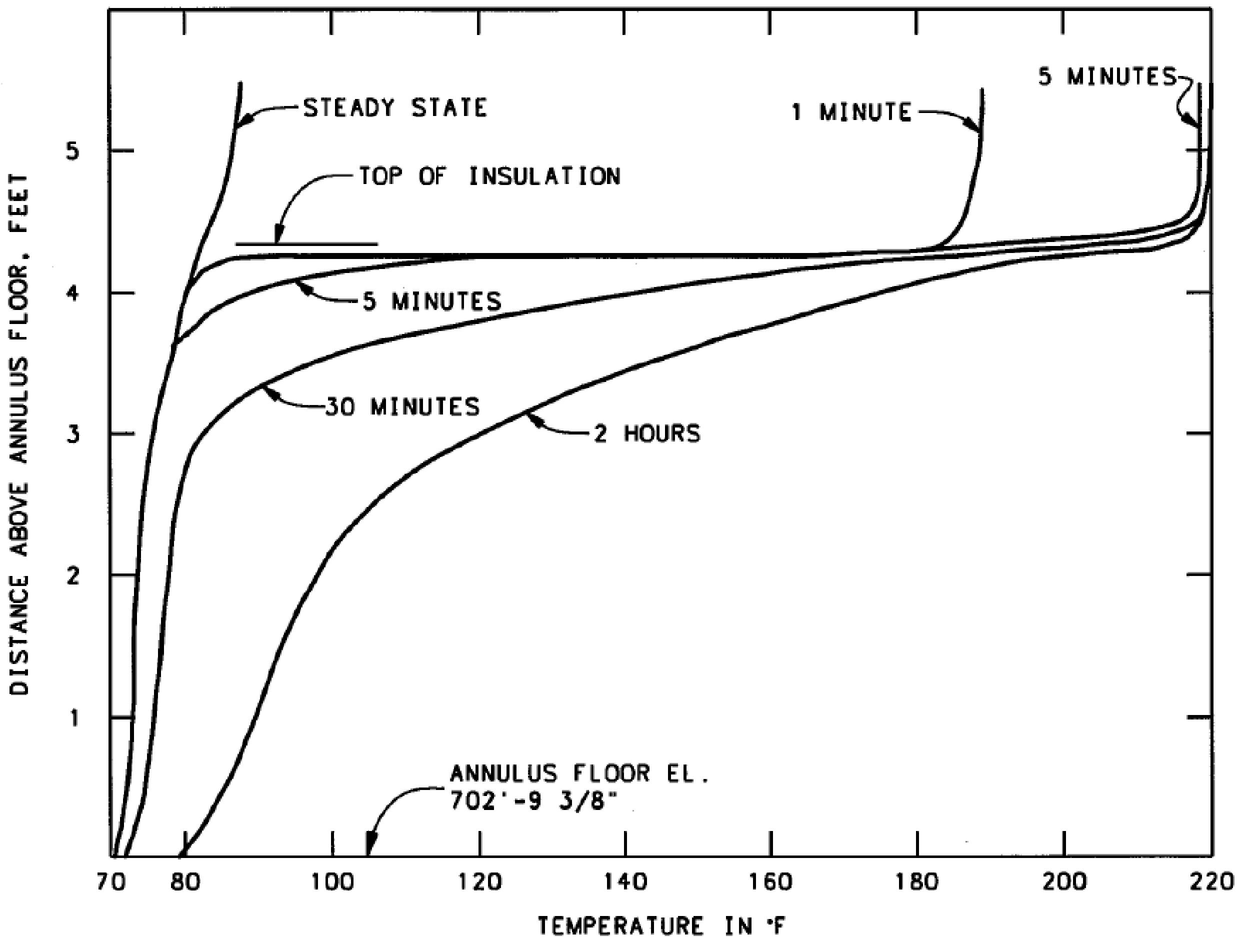
Shell Wall Temperature Versus
Time After Loss of Coolant

FIGURE 3.8A-1

TYPICAL TEMPERATURE TRANSIENT
LOWER COMPARTMENT WALL



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Temperature Transient
Lower Compartment

FIGURE 3.8A-2

APPENDIX 3.8B

BUCKLING STRESS CRITERIA

3.8B.1 INTRODUCTION

The buckling design criteria in this appendix are applicable to stiffened circular cylindrical and spherical shells. Section 2.0 sets forth the buckling design criteria for shells stiffened with circumferential stiffeners. Because of existing penetrations, interferences, or large attached masses, it may be expedient to further analyze some areas of the vessel as independent panels. Section 3.0 sets forth the criteria for shells stiffened with a combination of circumferential and vertical stiffeners. Section 4.0 deals with the criteria for a spherical dome. The procedures and data presented were adapted primarily from Chapter 3 of the Shell Analysis Manual, by E. H. Baker, A. P. Cappelli, L. Kovalevsky, F. L. Rish, and R. M. Verette, National Aeronautics and Space Administration, Washington, D.C., Contractor Report CR-912, April 1968. The criteria given in this section cover only the range of variables needed for the structural steel containment vessel for which these specifications were prepared.

The buckling criteria are specified in terms of unit stresses and membrane forces in the shell. Stresses caused by multiple loads must be combined according to provisions of Table 3.8B-1 for use in these criteria. The values of the load factors and factors of safety used in the buckling criteria are given in Section 5.0. The method of applying the factors of safety to the criteria is shown in Table 3.8B-2.

3.8B.2 SHELLS STIFFENED WITH CIRCUMFERENTIAL STIFFENERS

3.8B.2.1 Circular Cylindrical Shells Under Axial Compression

The critical buckling stress for a cylinder under axial compression alone is determined by the equation

$$\sigma_{cr}^{(1)} = \frac{C_c Et}{R}$$

for various ranges of cylinder length defined by

$$Z = \frac{L^2}{Rt}\sqrt{1-\mu^2}$$

The constant C_c is determined from Figure 3.8B-l for the appropriate value of R/t.

The critical buckling stress in a cylinder under axial compression and internal pressure is determined by:

$$\sigma_{cr}^{(1)} = (C_c + \Delta C_c)\frac{Et}{R}$$

The constants C_c and ΔC_c are determined from Figures 3.8B-1 and 3.8B-2, respectively. The constant ΔC_c given in Figure 3.8B-2 depends only upon the internal pressure and R/Et.

3.8B.2.2 Circular Cylindrical Shells in Circumferential Compression

A circular cylindrical shell under a critical external radial or hydrostatic pressure will buckle in circumferential compression. The critical circumferential compressive stress is given by:

$$\sigma_{cr}^{(2)} = \frac{K_p \pi^2 E}{12(1-u^2)} \left(\frac{t}{L}\right)^2$$

for various values of Z given in Section 2.1. Curves for determining the constant K_p for both radial and hydrostatic pressure are given in Figure 3.8B-3.

3.8B.2.3 Circular Cylindrical Shells Under Torsion

The shear buckling stress of the cylinder subject to torsional loads is given by:

$$\sigma_{cr}^{(3)} = C_s \frac{Et}{RZ^{1/4}}$$

The shear buckling stress of the cylinder subject to torsion and internal pressure is determined by

$$\sigma_{cr}^{(3)} = (C_s + \Delta C_s) C_s \frac{Et}{RZ^{1/4}}$$

where constants, C_s and ΔC_s, are determined from Figures 3.8B-4 and 3.8B-5. Values of ΔC_s are given for internal radial pressure alone and internal pressure plus an external load equal to the longitudinal force produced by the internal pressure.

Figure 3.8B-4 is applicable for values of:

$$Z = \frac{L^2}{Rt} \sqrt{1-\mu^2} > 100$$

For cylinders with length constant Z less than 100, the shear buckling stress is determined by:

$$\sigma_{cr}^{(3)} = \frac{K'_s \pi^2 E}{12 (1 - \mu^2)} \left(\frac{t}{a}\right)^2 \quad a \le b$$

for values of:

$$Z = \frac{a^2}{Rt} \sqrt{1 - \mu^2}$$

where a is the effective length and b is the circumference of the cylinder. The coefficient K_s' is given in Figure 3.8B-10.

3.8B.2.4 Circular Cylindrical Shells Under Bending

The critical buckling stress for the cylinder under bending is computed by the equation:

$$\sigma_{cr}^{(4)} = C_b \frac{Et}{R}$$

where the buckling constant, C_b is given by Figure 3.8B-6.

The critical buckling stress for the cylinder under internal pressure and bending is computed by:

$$\sigma_{cr}^{(4)} = (C_b + \Delta C_b) \frac{Et}{R}$$

where C_b and ΔC_b are given by Figures 3.8B-6 and 3.8B-7, respectively.

Figure 3.8B-7 is a function of the internal pressure and the geometry.

3.8B.2.5 Circular Cylindrical Shell Under Combined Loads

The criterion for buckling failure of the cylindrical shell under combined loading is expressed by an interaction equation of stress-ratios of the form:

$$R_1^x + R_2^y + R_3^z \langle 1$$

Note that

$$R_n = \frac{N_1 \; {}^{(n)}F_1}{\sigma_{cr} \; {}^{(n)}t} + \frac{N_2 \; {}^{(n)}F_2}{\sigma_{cr} \; {}^{(n)}t} + \ldots \frac{N_m \; {}^{(n)}F_m}{\sigma_{cr} \; {}^{(n)}t} \ldots \frac{N_k \; {}^{(n)}F_k}{\sigma_{cr} \; {}^{(n)}t}$$

where N_m is the compressive or shear membrane force and F_m is the appropriate load factor, given in Section 5.0, for individual loading components in any loading combination. The superscript n refers to the particular type of loading. Superscripts n = 1, 2, 3, and 4 represent respectively axial compression, circumferential compression, torsion, and bending loads.

The following interaction equations were used in the design of the cylindrical shell.

a. Axial Compression and Circumferential Compression

$$\sum_{m=o}^{m=k} \frac{N_m^{(1)} \ F_m}{\sigma_{cr}^{(1)} \ t} + \sum_{m=o}^{m=k} \frac{N_m^{(2)} \ F_m}{\sigma_{cr}^{(2)} \ t} < 1$$

b. Axial Compression and Bending

$$\sum_{m=o}^{m=k} \frac{N_m^{(1)} F_m}{\sigma_{cr}^{(1)} t} + \sum_{m=o}^{m=k} \frac{N_m^{(4)} F_m}{\sigma_{cr}^{(4)} t} < 19$$

c. Axial Compression and Torsion

$$\sum_{m=o}^{m=k} \frac{N_m^{(1)} \ F_m}{\sigma_{cr}^{(1)} \ t} + \left(\sum_{m=o}^{m=k} \frac{N_m^{(3)} \ F_m}{\sigma_{cr}^{(3)} \ t} \right)^2 < 1$$

d. Axial Compression, Bending, and Torsion

$$\sum_{m=o}^{m=k} \frac{N_m^{(1)} \ F_m}{\sigma_{cr}^{(1)} \ t} + \sum_{m=o}^{m=k} \frac{N_m^{(4)} \ F_m}{\sigma_{cr}^{(4)} \ t} + \left(\sum_{m=o}^{m=k} \frac{N_m^{(3)} \ F_m}{\sigma_{cr}^{(3)} \ t} \right)^2 < 1$$

e. Axial Compression, Circumferential Compression, and Torsion

$$\sum_{m=o}^{m=k} \frac{N_m^{(1)} \ F_m}{\sigma_{cr}^{(1)} \ t} + \sum_{m=o}^{m=k} \frac{N_m^{(2)} \ F_m}{\sigma_{cr}^{(2)} \ t} + \left(\sum_{m=o}^{m=k} \frac{N_m^{(3)} \ F_m}{\sigma_{cr}^{(3)} \ t} \right)^2 < 1$$

The longitudinal membrane stresses produced by the nonaxisymmetric pressure loads (NASPL) were considered as caused by bending loads in the interaction equations.

3.8B.3 SHELLS STIFFENED WITH A COMBINATION OF CIRCUMFERENTIAL AND VERTICAL STIFFENERS

3.8B.3.1 The shell was provided with permanent circumferential and vertical stiffeners. The circumferential stiffeners were designed to have a spring stiffness at least great enough to enforce nodes in the vertical stiffeners so as to preclude a general instability mode of buckling failure, thus ensuring that if buckling occurs, it will occur in stiffened panels between the circumferential stiffeners. An acceptable procedure for determining the critical buckling stresses in the vertical stiffeners and stiffened panels is outlined in Section 3.43 Shell Analysis Manual, by E. H. Baker A. P. Cappelli, L. Kovalevsky, F. L. Rish, and R. M. Verette, National Aeronautics and Space Administration, Washington, D.C., Contractor Report CR-912, April 1968.

3.8B.3.2 In addition for shells stiffened with a combination of circumferential and vertical stiffeners under combined load, the criterion for buckling failure of the shell plate is expressed by an interaction equation of stress ratios in the form similar to the interaction equations of Section 2.5.

$$R_1^x + R_2^y + R_3^z < 1$$

The critical buckling stresses for the shell plates between the circumferential and vertical stiffeners were determined by the following equations.

a. Curved Panel under Axial Compression.

The critical buckling stress for a curved cylindrical panel under axial compression alone is determined by the equation:

$$\sigma_{cr} = \frac{K_c \ \pi^2 \ E}{12(1-\mu^2)} \left(\frac{t}{b}\right)^2$$

for various ranges of cylinder length given by:

$$Z = \frac{b^2}{Rt} \sqrt{1 - \mu^2}$$

The constant K_c is determined from Figure 3.8B-8.

b. Curved Panel in Circumferential Compression

The critical buckling stress of a curved cylindrical panel under circumferential compression was determined by Section 2.2.

c. Curved Panel Under Torsion

The shear buckling stress of a curved cylindrical panel subjected to torsional loads is given by:

$$\sigma_{cr} = \frac{K_s \pi^2 E}{12 (1 - \mu^2)} \left(\frac{t}{b}\right)^2 \quad a \geq b$$

for values of:

$$Z = \frac{b^2}{Rt} \sqrt{1 - \mu^2}$$

The coefficient, K_s, is given in Figure 3.8B-9. For cylindrical panels with length, a, less than the arc length, b, the shear buckling stress is determined by:

$$\sigma_{cr} = \frac{K'_s \pi^2 E}{12 (1 - \mu^2)} \left(\frac{t}{a}\right)^2 \quad a \leq b$$

for values of:

$$z = \frac{a^2}{Rt} \sqrt{1 - \mu^2}$$

Curves for determining K_s' are given in Figure 3.8B-10.

d. Curved Panels Under Bending

The critical buckling stress for a curved panel in bending was computed using the equation for axial compression given in (a) of this section.

3.8B.3.3 The critical buckling stress in a stiffened hemispherical shell for the analysis was not treated in the Shell Analysis Manual, and except for external pressure, was determined by the following equation:

$$\sigma_{cr} = 0.125 E \frac{t}{R}$$

where t = thickness of shell
E = modulus of elasticity
R = radius of shell

3.8B.4 SPHERICAL SHELLS

3.8B.4.1 The critical buckling stress in the spherical dome, except for external pressure, was determined by the following equation:

$$\sigma_{cr} = 0.125 \frac{Et}{R}$$

where t = thickness of shell
E = modulus of elasticity
R = radius of shell

3.8B.4.2 Spherical Shell Under Combined Loads

The criterion for buckling failure of the dome is expressed by an interaction equation of the stress ratios in the form:

$$R_1^x + R_2^y + R_3^z < 1$$

similar to the interaction equation of Section 2.5.

A set of interaction equations similar to those in Section 2.5 was used in the design except that the effects due to torsion were considered.

3.8B.5 FACTOR OF SAFETY

The buckling stress criteria were evaluated to determine the factors of safety against buckling inherent in the criteria. The factors which affect stability were determined and the criteria were evaluated to account for these factors. The basis used to evaluate the criteria to account for the factors were (1) how well established are the effects of the factors on stability of these shells (2) amount of supporting data in the literature and (3) margins marked by the critical stresses and interaction equations used in the criteria. The buckling criteria were found to be very conservative and judged to provide at least a factor of safety of 2.0 against buckling for all loading conditions for which the vessels were designed.

In addition, a load factor of 1.1 was applied to load conditions which include the Safe Shutdown Earthquake (SSE). A load factor of 1.25 was used with all other load conditions.

TABLE 3.8.B-1 (Sheet 1 of 4)

MULTIPLE LOAD COMBINATIONS
VARIOUS PLANT CONDITIONS

LOADING CONDITIONS

Load Components	Const. Cond.	Test. Cond.	Normal Design	Norm. Oper. 1/2 SSE	Accident 1/2 SSE	MSLB Accident (Static) 1/2 SSE	Accident SSE	MSLB Accident (Static) SSE	Post Accident Flooding
Personnel Access Lock Load			X	X					
Penetration Loads			X	X	X	X	X	X	
Containment Vessel and Appurtenances Dead Loads	X	X	X	X	X	X	X	X	X
Walkway Live Loads				X					
Spray Header and Lighting Fixtures Live Loads				X		X		X	
Safe Shutdown Earthquake (SSE) Lateral and Vertical Loads							X	X	
Design Internal Pressure or Design External Pressure		X	X						
One-half Safe Shutdown Earthquake (1/2 SSE) Lateral and Vertical Loads			X	X	X	X			

TABLE 3.8.B-1 (Sheet 2 of 4)

MULTIPLE LOAD COMBINATIONS VARIOUS PLANT CONDITIONS

LOADING CONDITIONS

Load Components	Const. Cond.	Test. Cond.	Normal Design	Norm. Oper. 1/2 SSE	Accident 1/2 SSE	MSLB Accident (Static) 1/2 SSE	Accident SSE	MSLB Accident (Static) SSE	Post Accident Flooding
Design Internal Pressure or Pressure Transient Loads					X		X		
Design Temperature			X						
Internal Temperature Range of 60°F to 120°F				X	X (for Pressure Transient Loads)		X (for Pressure Transient Loads)		
Internal Temperature Range of 80°F to 250°F					X (for Design Internal Pressure)		X (for Design Internal Pressure)		
Thermal Stress Loads Including Shell Temperature Transients					X		X		
Hydrostatic Load Case 1A or 1B (See Note 1)					X		X		

TABLE 3.8.B-1 (Sheet 3 of 4)

MULTIPLE LOAD COMBINATIONS
VARIOUS PLANT CONDITIONS

LOADING CONDITIONS

Load Components	Const. Cond.	Test. Cond.	Normal Design	Norm. Oper. 1/2 SSE	Accident 1/2 SSE	MSLB Accident (Static) 1/2 SSE	Accident SSE	MSLB Accident (Static) SSE	Post Accident Flooding
Hydrostatic Load Case II (See Note 1)									X
Wind Loads (See Note 2)	X								
Snow Loads (See Note 2)	X								
Temporary Construction Loads	X								
Internal Test Pressure		X							
Weight of Contained Air		X							
MSLB Pressure						X		X	
Internal Temperature Range of 80°F to 327°F						X		X	
Airlock Live Load			X	X					
Ice Condenser Duct Loads						X		X	

TABLE 3.8.B-1 (Sheet 4 of 4)

MULTIPLE LOAD COMBINATIONS
VARIOUS PLANT CONDITIONS

Notes:

1. Hydrostatic loads case 1A & 1B, and load case II are shown on TVA drawing 48N400.

2. Wind and snow loads do not act simultaneously.

3. For allowable stress condition see Table 3.8.B-2.

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TABLE 3.8B-2

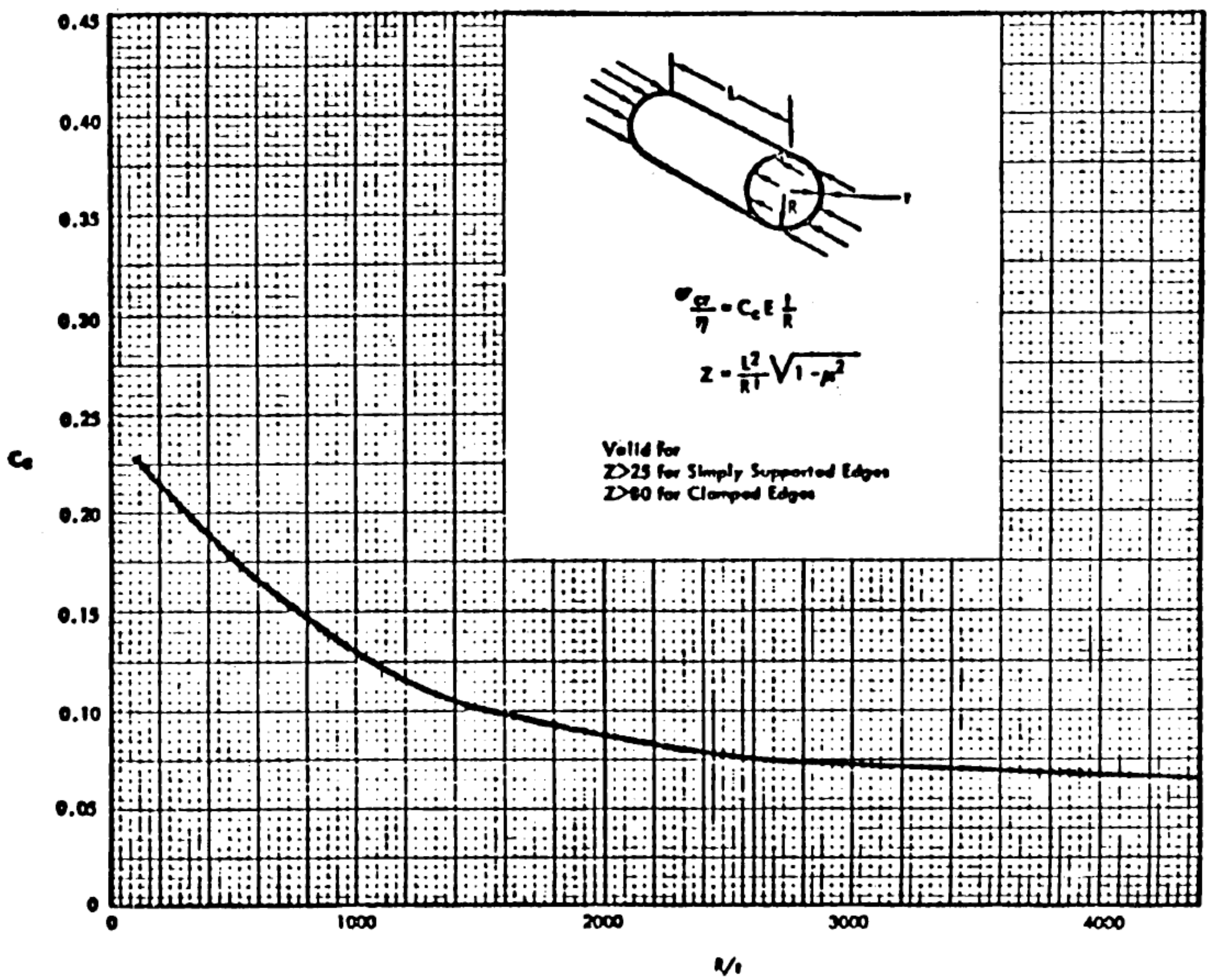
ALLOWABLE STRESS INTENSITIES

PLUS BUCKLING LOAD FACTORS

Loading Conditions	Applicable ASME Code Reference (1) for Stress Intensity	Buckling Load Factors
Normal Design Condition Construction Condition	NB-3221	In accordance with ASME Code, Section VIII
Normal Operation Condition	NB-3222	Load factor = 1.25 for both cylindrical portion and hemispherical head
Upset Operation Condition	NB-3223	Load factor = 1.25 for both cylindrical portion and hemispherical head
Emergency Operation Condition	NB-3224	Load factor = 1.10 for both cylindrical portion and hemispherical head
Test Condition	NB-3226	NA
Post-Accident Fuel Recovery Condition	NB-3224	NA

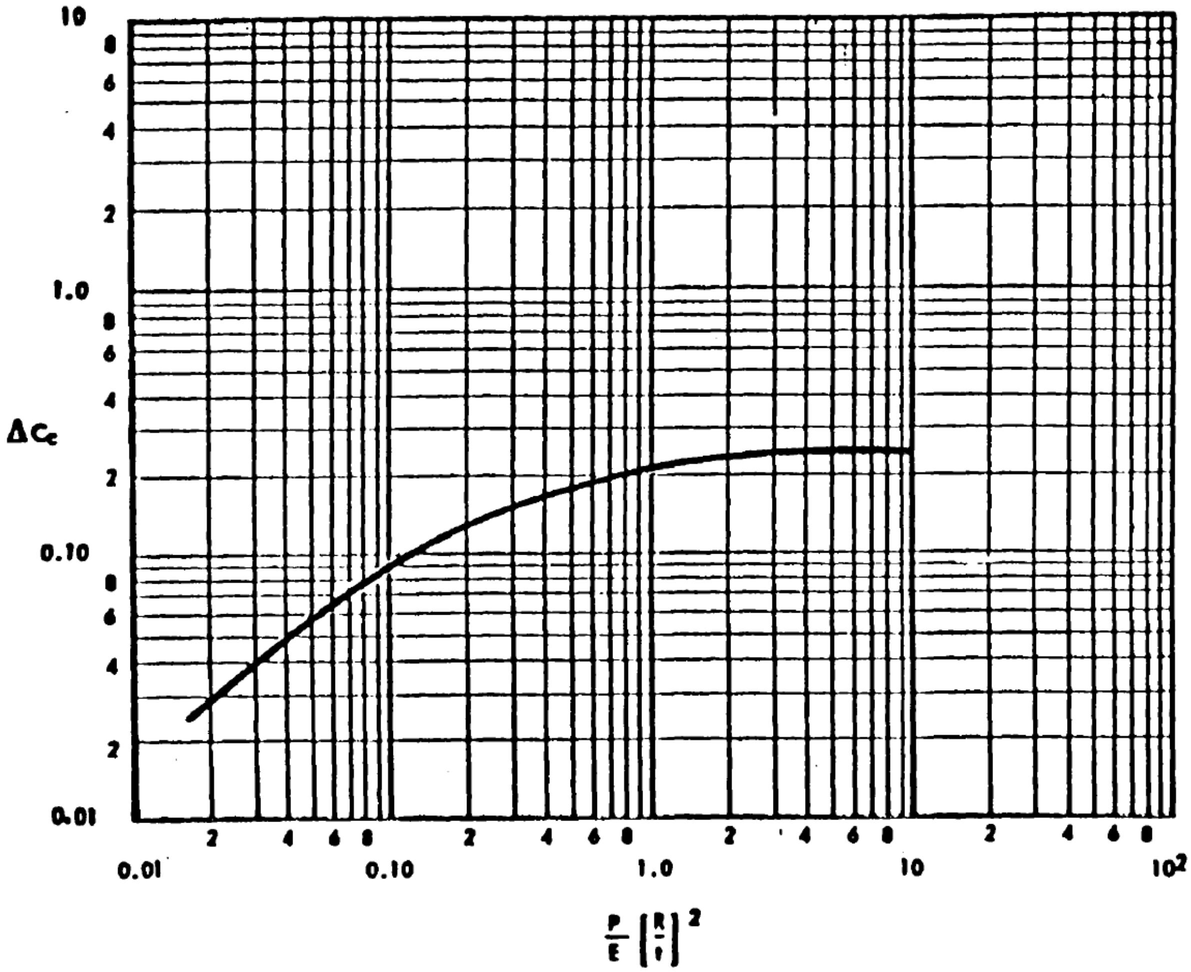
(1) All code references are to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971 Edition, with Winter 1971 Addenda.

BUCKLING-STRESS COEFFICIENT, C_C, FOR UNSTIFFENED UNPRESSURIZED CIRCULAR CYLINDERS SUBJECTED TO AXIAL COMPRESSION



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Buckling-Stress Coefficient FIGURE 3.8B-1

INCREASE IN AXIAL-COMPRESSIVE BUCKLING-STRESS COEFFICIENT OF CYLINDERS DUE TO INTERNAL PRESSURE



WATTS BAR NUCLEAR PLANT
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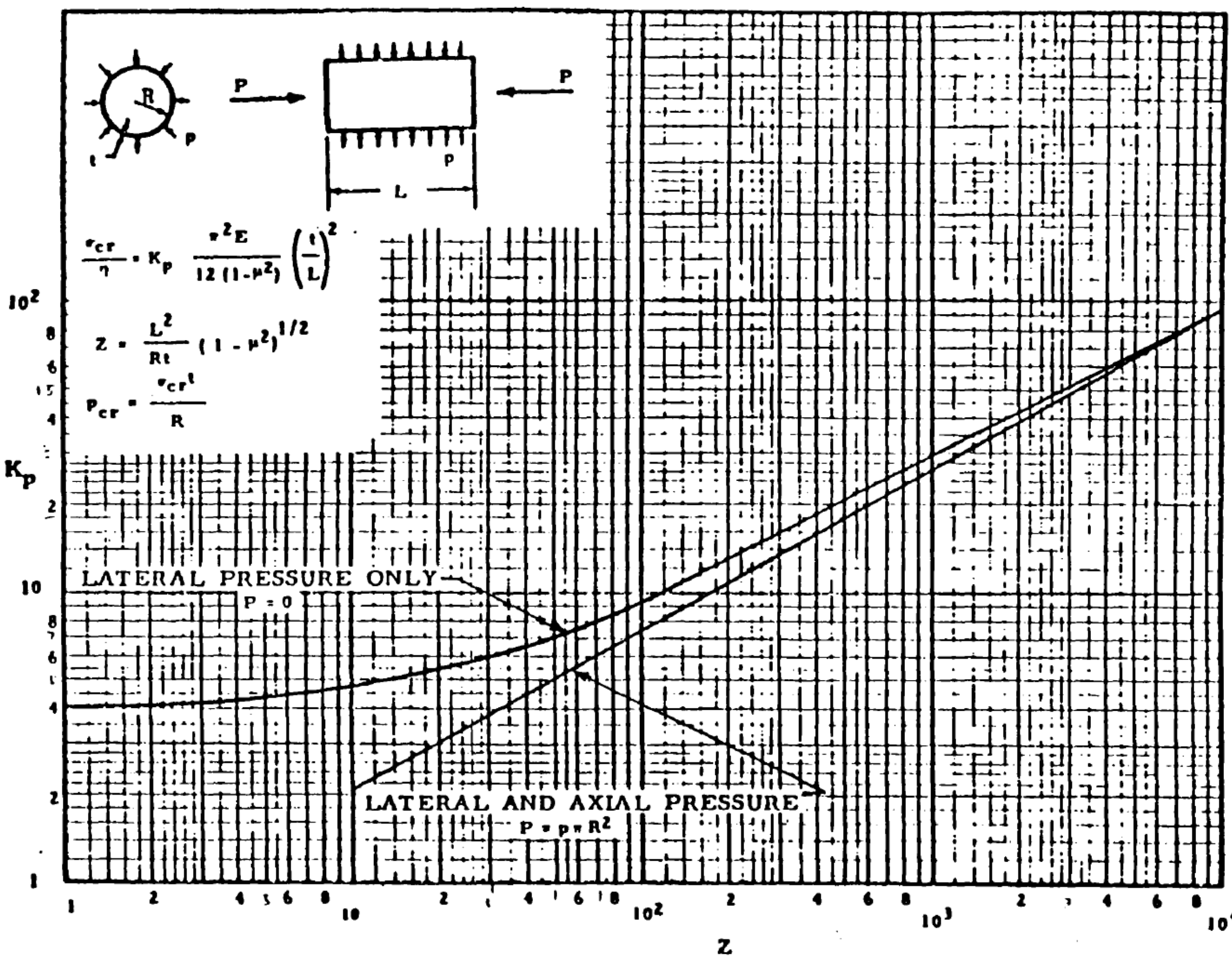
ANALYSIS REPORT

Axial-Compressive Buckling Stress

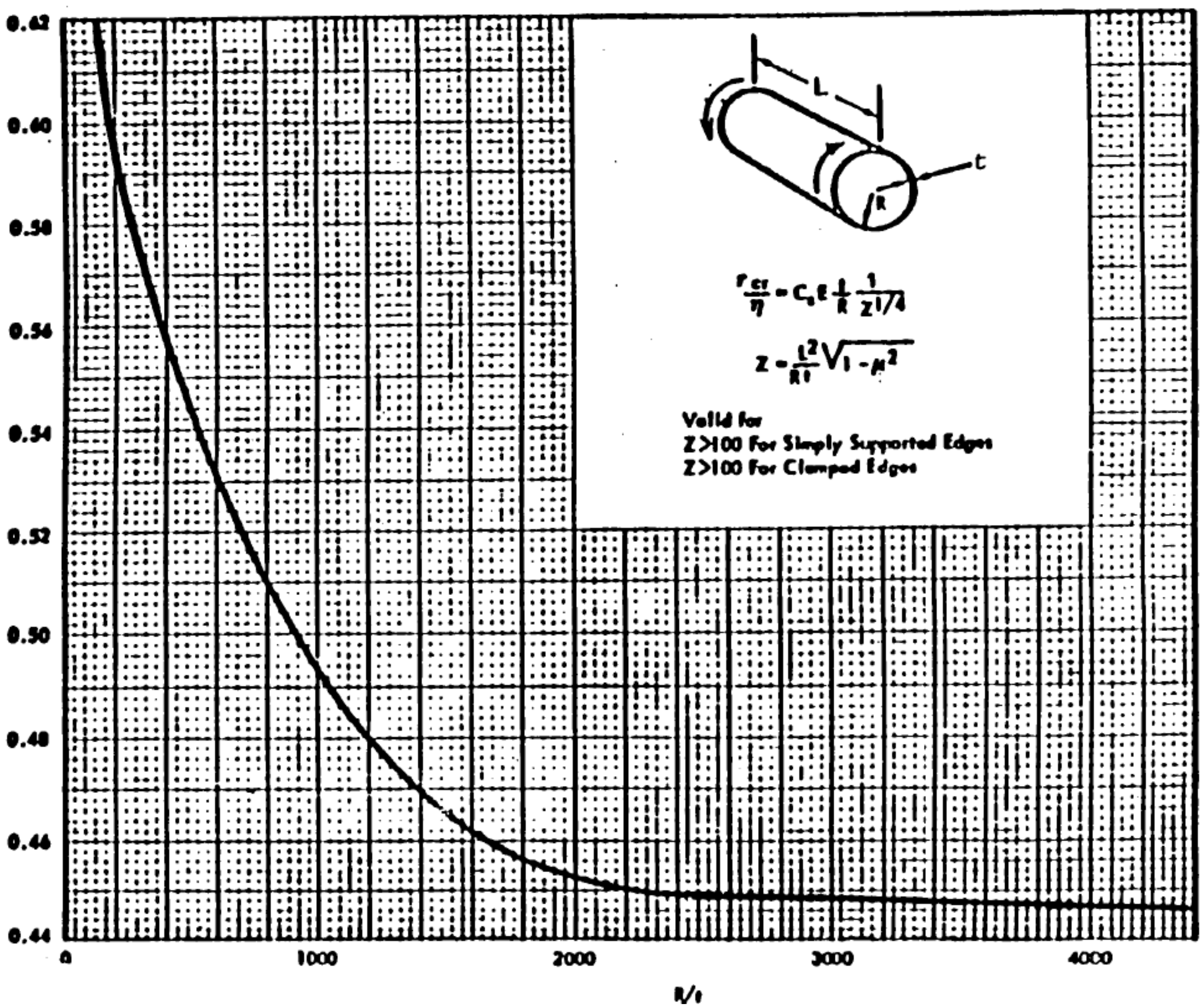
FIGURE 3.8B-2

BUCKLING COEFFICIENTS FOR CIRCULAR CYLINDERS SUBJECTED TO EXTERNAL PRESSURE



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Buckling Coefficients for Circular Cylinders FIGURE 3.8B-3

BUCKLING-STRESS COEFFICIENT, CS FOR UNSTIFFENED UNPRESSUREIZED CIRCULAR CYLINDERS SUBJECTED TO TORSION

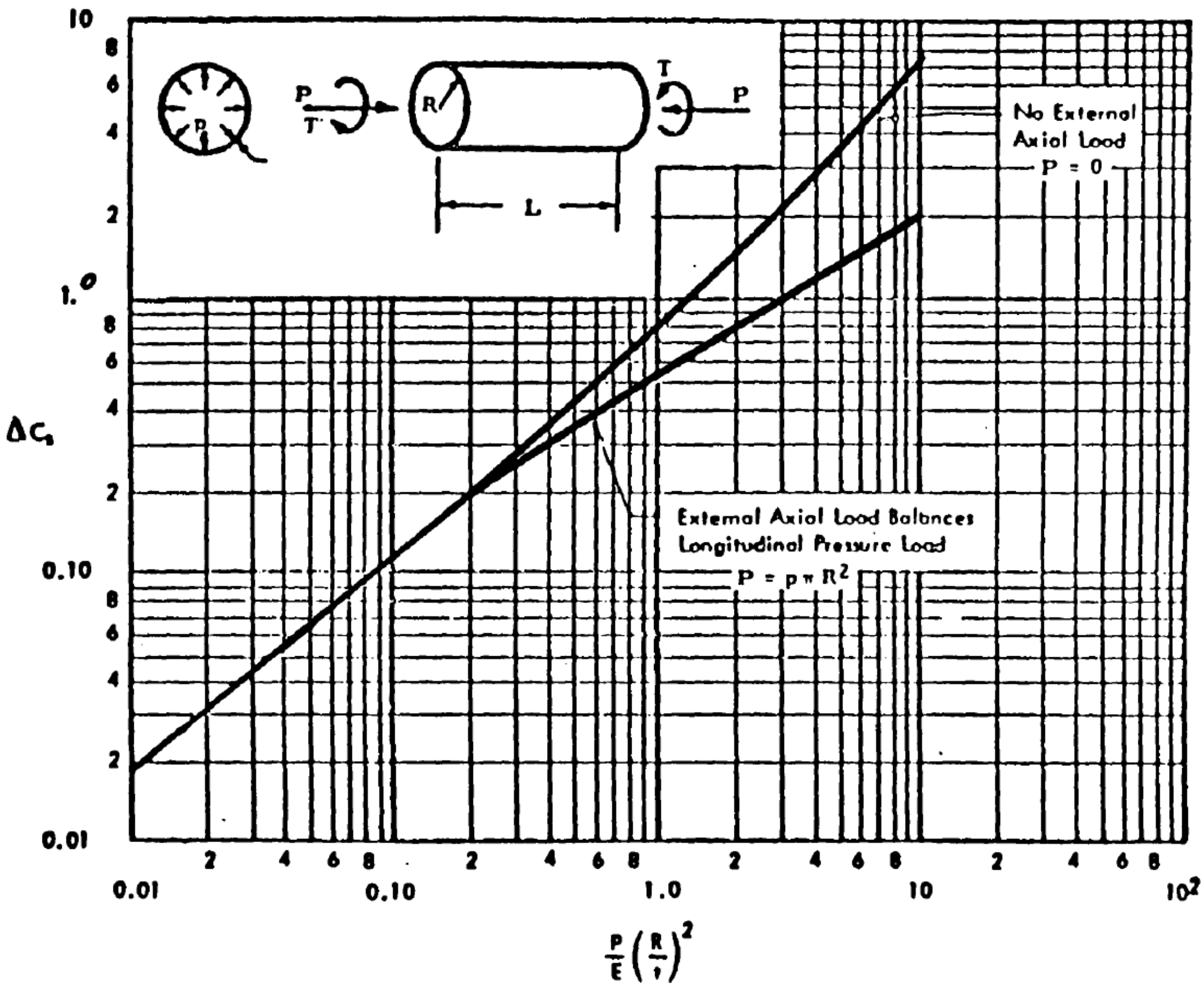


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Buckling-Stress Coefficients

FIGURE 3.8B-4

INCREASE IN TORSIONAL BUCKLING-STRESS COEFFICIENT OF CYLINDERS DUE TO INTERNAL PRESSURE

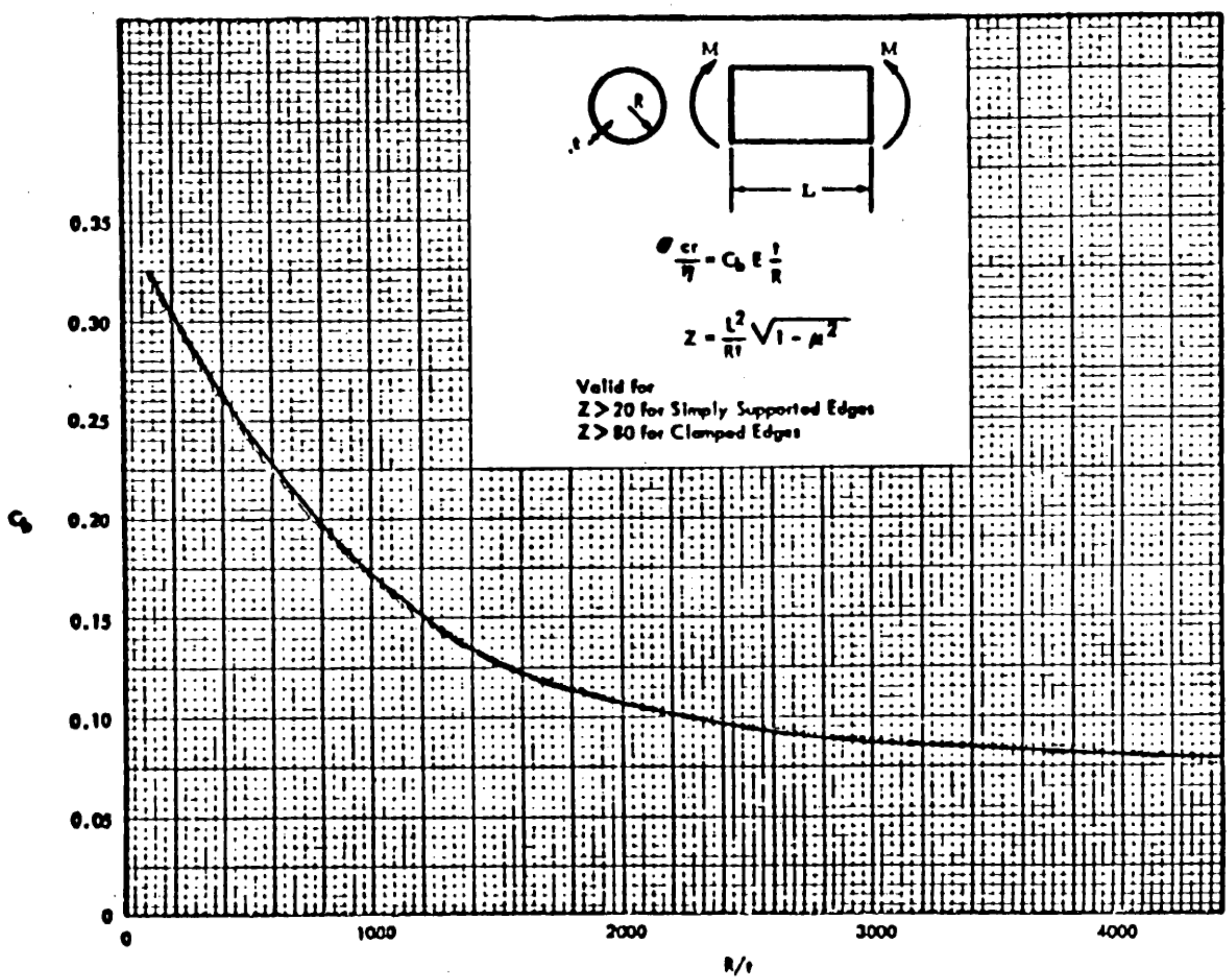


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Torsional Buckling-Stress Coefficients

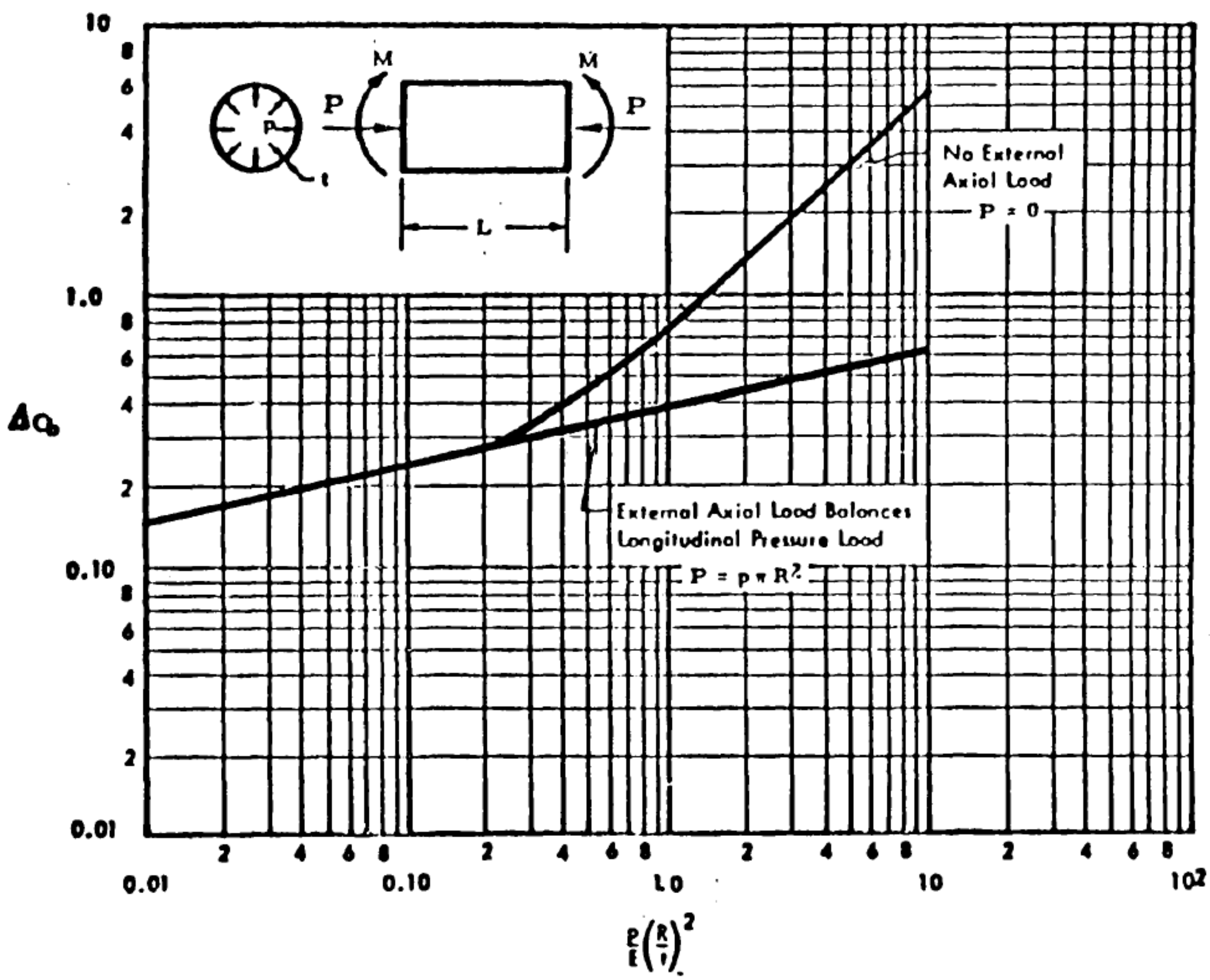
FIGURE 3.8B-5

BUCKLING-STRESS COEFFICIENT, C_b, FOR UNSTIFFENED UNPRESSURIZED CIRCULAR CYLINDERS SUBJECTED TO BENDING



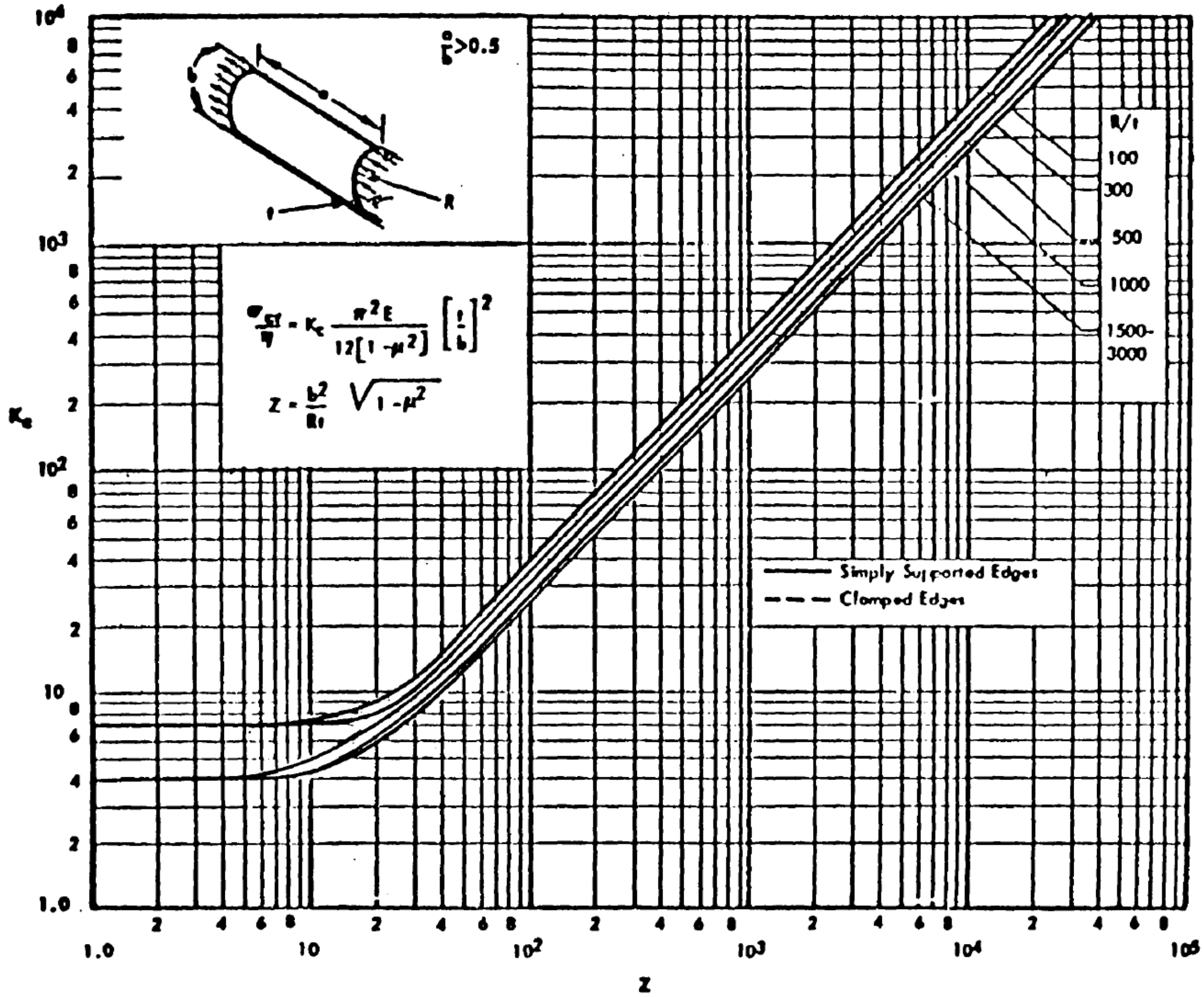
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Buckling-Stress Coefficient, C_b FIGURE 3.8B-6

INCREASE IN BENDING BUCKLING-STRESS COEFFICIENT
OF CYLINDERS DUE TO INTERNAL PRESSURE



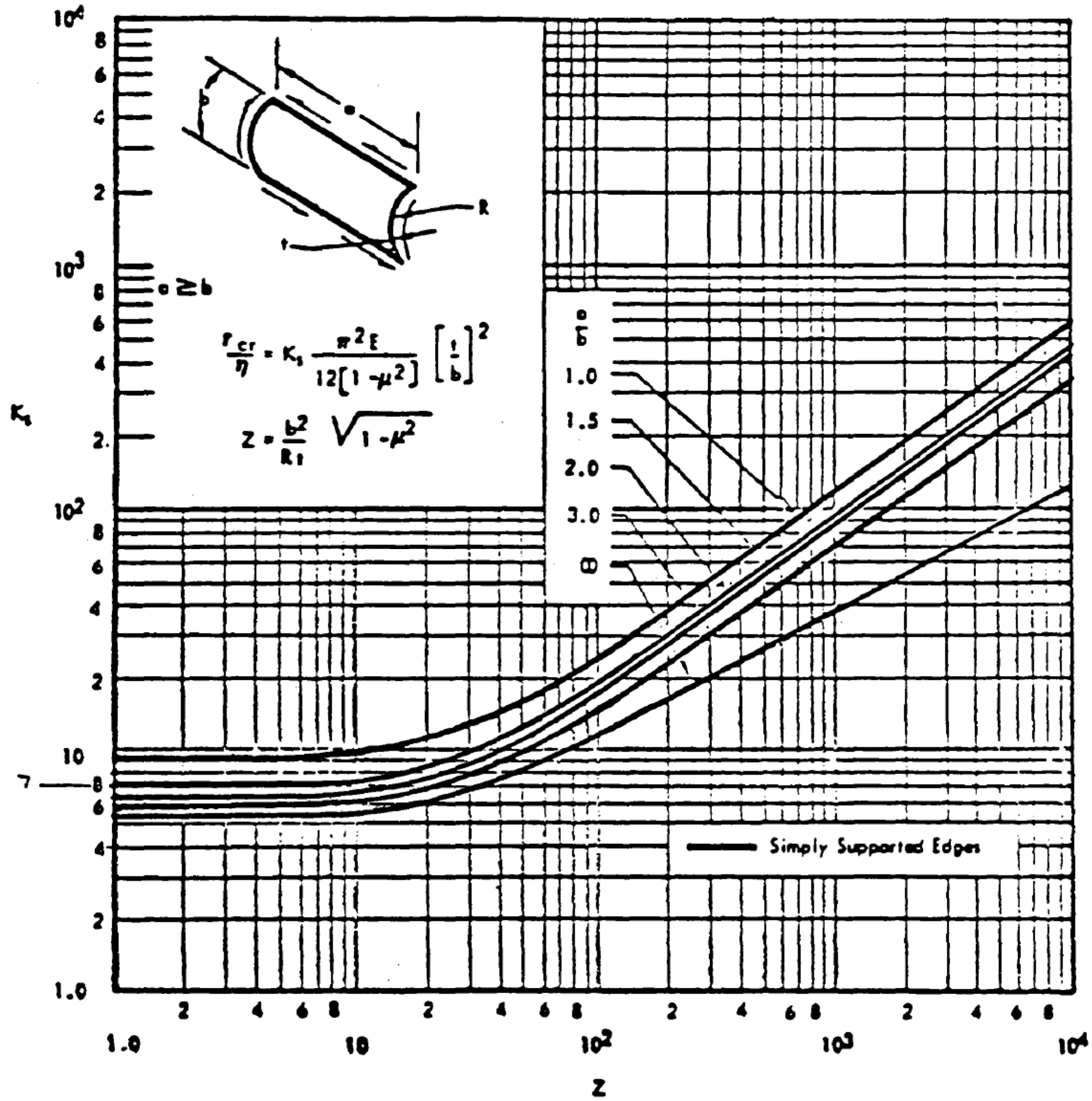
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Bending Buckling-Stress Coefficient FIGURE 3.8B-7

BUCKLING-STRESS COEFFICIENT, K_C, FOR UNPRESSURIZED CURVED PANELS SUBJECTED TO AXIAL COMPRESSION



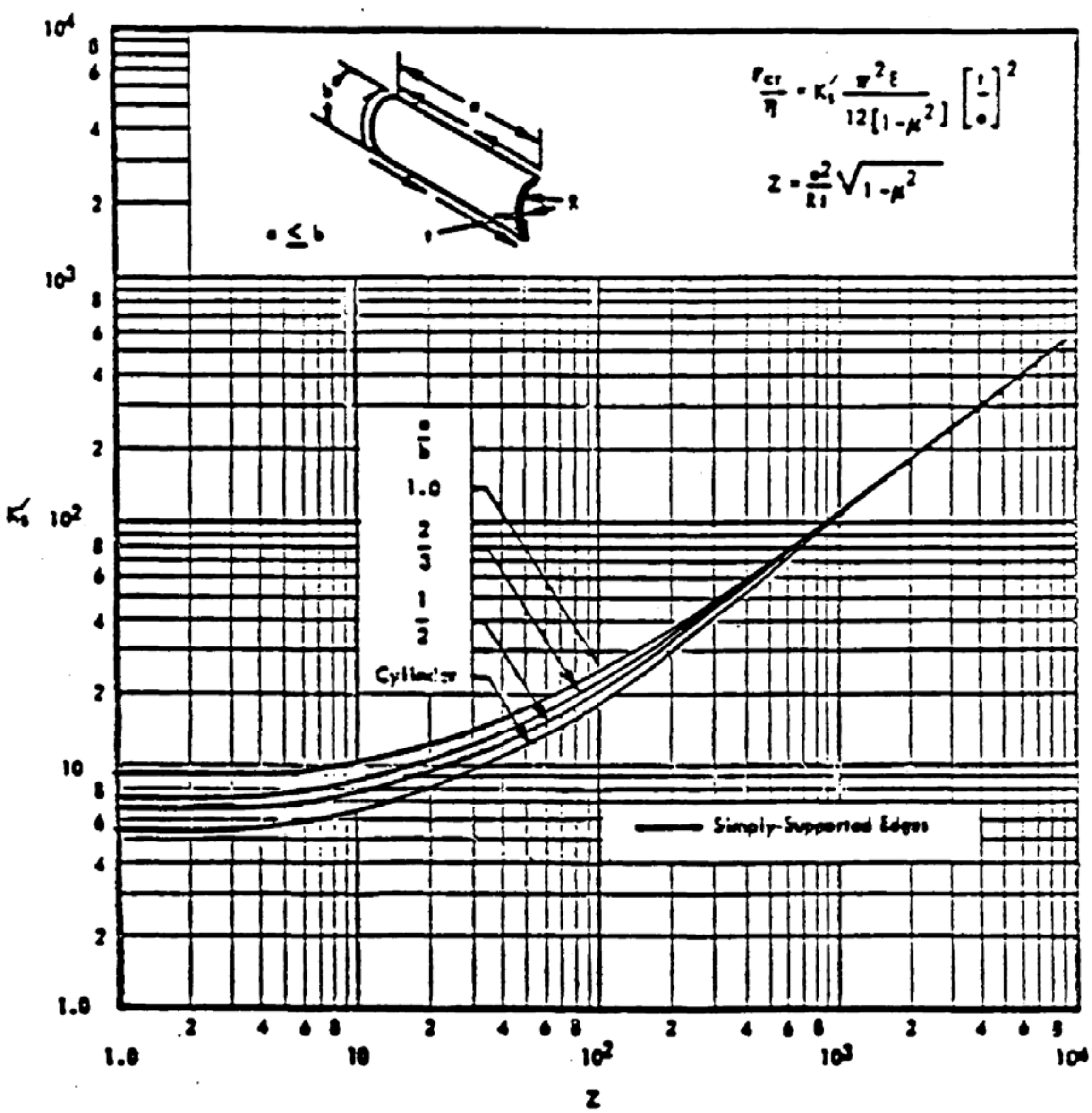
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Buckling-Stress Coefficient, K_C FIGURE 3.8B-8

BUCKLING-STRESS COEFFICIENT, K_S, FOR UNPRESSURIZED CURVED PANELS SUBJECTED TO SHEAR



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Buckling-Stress Coefficient, K_s FIGURE 3.8B-9

BUCKLING-STRESS COEFFICIENT, K'$_S$, FOR UNPRESSURIZED CURVED PANELS SUBJECTED TO SHEAR



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Buckling-Stress Coefficient, K'$_s$ FIGURE 3.8B-10

APPENDIX 3.8C

DOCUMENTATION OF CB&I COMPUTER PROGRAMS

3.8C.1 INTRODUCTION

This appendix presents abstracts of the computer programs employed in the design and analysis of the Watts Bar containment vessels. These abstracts explain the purpose of the program and give a brief description of the methods of analysis performed by the program.

Analytical derivations are not contained herein, but are in the CB&I Stress Report.

3.8C.2 PROGRAM 1017-MODAL ANALYSIS OF STRUCTURES USING THE EIGEN VALUE TECHNIQUE

The purpose of this program is three-fold:

1. To calculate the mass and stiffness matrices associated with the structural model.

2. To determine the undamped natural periods of the model.

3. To calculate the maximum modal responses of the structure; i.e., deflections, shears, and moments.

The stiffness and mass matrices may be required in order to perform a dynamic analysis of the structure. The maximum modal responses may be used to perform a spectral analysis.

The program has the following options:

1. Vertical translation.
2. Torsional modes.
3. Soil-structure interaction.
4. Liquid sloshing.
5. Direct introduction of stiffness and mass matrices.

3.8C.3 PROGRAM 1044-SEISMIC ANALYSIS of VESSEL APPENDAGES

Appendages to a vessel may not significantly contribute structurally to the dynamic responses of a model of a vessel. However, appendages can effect the vessel locally by vibrating differently from the model of the vessel at the point of attachment.

The response spectrum method of analysis is not a strictly adequate way of obtaining the maximum appendage accelerations since it does not include the possible consequences of near resonance between the vessel model and the appendage model.

This paper describes the method used to evaluate the maximum elastic differential accelerations between an independently vibrating appendage model and an elastic beam vessel model at the appendage elevation due to known excitations of the elastic beam model.

The method involves two distinct steps. Firstly, the necessary time-absolute acceleration records are computed at appendage elevations due to model excitations. Secondly, the maximum differential accelerations between each appendage model and the vessel model at the appendage elevation are obtained.

The time-absolute acceleration records at the appendage elevation are computed by use of a step-by-step matrix analysis procedure. The equations of motion for the vessel model are of the form:

$$[M]\ \{\ddot{u}\} + (AT/\pi)\ [K]\ \{\acute{u}\} + [K]\ \{u\} = -[M]\ \{\ddot{u}_g\}$$

where

[M]	=	Mass matrix, order n x n obtained from a modal analysis.
[K]	=	Stiffness matrix, order n x n, obtained from a modal analysis.
A	=	Portion of first mode critical damping for the model
T	=	First mode of the model
[M]	=	A diagonal matrix, order n x n, with diagonal elements corresponding to elements of the mass excited by translational accelerations.
{d}	=	n x 1 matrix of relative accelerations between the model base and the n degrees of freedom.
{b}	=	n x 1 matrix of velocities corresponding to {d}
{u}	=	n x 1 matrix of displacements corresponding to {d}
$\{d_g\}$	=	n x 1 matrix of translation base acceleration.
n	=	Degrees of freedom of vessel model.

By taking a small time increment (smaller than the smallest period obtained from the model analysis) and letting accelerations vary linearly within the selected increment, the equations of motion can be integrated for the quantities {u}, {b}and {d} over the expected time increment[1]. The values obtained are superimposed upon the values of these quantities existing at the beginning of the time increment. This process is repeated for the duration of the excitation. The time-absolute acceleration records for each translational degree of freedom are the sums of {d} and $\{d_g\}$ taken throughout the history of the excitation.

The second step is similar to the first step. The equation of motion (n = 1) is written for the appendage as a single degree-of-freedom elastic model using the time-absolute acceleration record obtained in Step 1 at the appendage elevation as the excitation. This equation is solved in the same manner used in Step 1. The maximum absolute value of {d} obtained is the quantity desired. It is the maximum differential acceleration between the appendage model and the vessel model due to a known excitation of the vessel mode.

For any appendage, this two-step procedure should be executed three times. This is required to evaluate normal, tangential and vertical appendage accelerations with respect to a vessel cross-section.

3.8C.4 PROGRAM E1668-SPECTRAL ANALYSIS FOR ACCELERATION RECORDS DIGITIZED AT EQUAL INTERVALS

Program E1668 evaluates dynamic response spectra at various periods and presents the results on a printed plot. Given the time-acceleration record, the program numerically integrates the normal convolution time integral for various natural periods and damping ratios. The computed relative displacements, relative and pseudo-relative velocities, and absolute and pseudo-absolute accelerations are tabulated for periods from 0.025 seconds to 1 second.

3.8C.5 PROGRAM 1642-TRANSIENT PRESSURE BEAM ANALYSIS

The program was developed to perform the numerical integration required for the transient pressure beam analysis. The pressure transient curve for each compartment is read in and stored as a series of coordinates. At any time instant the total force acting at each compartment is calculated by multiplying the pressures by the corresponding areas of the shell over which they act. Each force is then distributed to the vessel model masses directly above and below. The proportion applied to each mass is based upon their respective distances from the force.

For a given increment the program checks the current time, determines the current pressure in each compartment, calculates the current force on each mass and applies a recurrence formula. The deflection values, y(t) and y(t-Δt) are updated, the current time incremented, and the process is repeated.

The values of the orthogonal deflections are stored and also printed out for a prescribed number of times, every ten increments or so, and the equivalent static forces determined. Equivalent static forces are those which produce deflections identical to the calculated kinetic deflections; they are obtained by multiplying the deflection vector by the stiffness matrix.

$$[F]\ \text{equivalent} = [K]\ [Y]$$

The shears and moments at the particular time are then determined from statics:

$$Q = \Sigma F$$
$$M = \Sigma F \cdot \text{moment arm}$$

The maximum moments, M = $\sqrt{M_x^2 + M_y^2}$

and maximum shears, Q = $\sqrt{Q_x^2 \ + \ Q_y^2}$ are then printed out at selected locations.

In addition the program will also print out an acceleration trace at the mass points.

3.8C.6 PROGRAM E1623-POST PROCESSOR PROGRAM FOR PROGRAM E1374

Program E1623 was written specifically for the TVA Watts Bar Containment Vessels. It performs the following operations:

1. Using Fourier data generated by Program El374 (Dynamic Shell Analysis), the summed displacements, forces and stresses found for various points around the shell circumference at each output point on the meridian.

2. The maximum of the summed values along with the associated time and azimuth are saved for each elevation and printed out at the end of the problem.

3. The following tables are printed:

    1. Radial deflection, ω, at each elevation versus azimuth
    2. Longitudinal force, Nϕ, at each elevation versus azimuth
    3. Longitudinal moment, Mϕ, at each elevation versus azimuth
    4. Circumferential force, $N_{\phi\checkmark}$, at each elevation versus azimuth

    The time basis for these tables is the occurrence of the minimum longitudinal force at the base.

4. Ring forces are calculated and then the maximums are pointed out.

5. Displacement traces at several elevations can be saved on a tape or disk unit.

6. The membrane stress resultants are saved on either a tape or disk unit for input into the buckling check program.

Program E1374 writes the Fourier amplitude results of the fundamental variables (ω, $☒_\phi$, B_ϕ, $☒_\checkmark$, Q, Nϕ, Mϕ, N) on a labeled tape after each timestop. Program E1623 reads this tape, interpolates to obtain the values at the output times, and calculates the remaining forces and all the stresses.

The amplitudes are then summed using the following equation:

$$f(\chi, \theta, t) = \sum_{n=1}^{m} g_n (x, t) \cos n\,\theta + \sum_{n=1}^{m} h_n (x, t) \sin n\,\theta$$

where:

x = meridinal coordinate
t = time
$g_n(x,t)$ = amplitudes of cosine harmonics
$h_n(x,t)$ = amplitudes of sine harmonics
θ = azimuth
$f(x,\theta,t)$ = Fourier sum
m = maximum number of circumferential waves

3.8C.7 PROGRAM E1374-SHELL DYNAMIC ANALYSIS

1. Introduction

Program E1374 is CBI's shell dynamic analysis program. Presently, it is capable of extracting eigenvalues and performing undamped transient analyses. Non-axisymmetric loads can be handled through the use of appropriate Fourier series.

The equation of motion for a particular Fourier harmonic n of an undamped system is

$$[M_n] [\ddot{U}_n] + [K_n] [U_n] = [P_n]$$

where: $[M_n]$ = Mass matrix

$[K_n]$ = Stiffness matrix

$[P_n]$ = Applied load

$[U_n]$ = Displacement

$[\ddot{U}_n]$ = Acceleration

Note that all of the above are functions of n.

In order to calculate free vibration frequencies and mode shapes the applied load is set equal to zero, $[U_n]$ is assumed to be a harmonic function of time, and the eigenvalues and eigenvectors of the resulting equation obtained.

If the transient response due to a time-varying load is required, a numerical integration technique is used.

Since Program E1374 is not set up to handle longitudinal stiffeners, the integration for this portion of the shell is performed using Program 781. The influence values are then converted to stiffness matrix form and stored on disc. After Program E1374 has set up the stiffness matrices for the unstiffened shell, the matrices for segments with stiffening are replaced with the Program 781 matrices from disc. The solution in Program E1374 then continues in the standard manner. This consists of assembling the overall stiffness matrix [K_n] and load vector [C_n], reducing to upper triangular form, and back-substituting.

3.8C.8 PROGRAM E1622-LOAD GENERATION PREPROCESSOR FOR PROGRAM E1374

In order to perform non-axisymmetric analyses on shells, the load must often be defined using Fourier series representation. The purpose of Program E1622 is to calculate and store on magnetic tape a time history of the Fourier pressure amplitudes. The format of this tape is designed specifically for use with Program E1374.

In order to calculate the amplitudes of the harmonics several assumptions are made in the program.

1. A linear function in the circumferential direction is assumed between given points.

2. Only distributed loads are considered.

3. The model consists of a cylindrical shell and optional hemispherical top head.

4. The pressure has a block type distribution in the longitudinal direction.

5. Any initial pressure acting on the shell can be subtracted from the input pressure histories.

6. Amplitudes for both sine and cosine terms can be calculated with the user supplying the range of harmonics to be output.

3.8C.9 PROGRAM E1624 SPCGEN-SPECTRAL CURVE GENERATION

Program E1624 reads the Fourier amplitudes of the deflection transients stored on magnetic tape from the output of Program E1374. The program calculates the accelerations at uniform time intervals and evaluates the response spectra. From the deflection transient for each harmonic, the acceleration traces are computer generated using three point central difference for the first and last three time steps, and a seven point central difference elsewhere.

3.8C.10 PROGRAM 781, METHOD OF MODELING VERTICAL STIFFENERS

N = No. of vertical stiffeners around

E = Modulus of elasticity

The shell shown in Figure 3.8C-I is modeled using 2 layers. The inside layer represents the shell and, therefore, has the normal isotropic material properties. The outer layer, on the other hand is described as an orthotropic material having the following properties.

$t_2 = d$

$$E_{\phi_2} = \frac{bN}{2\pi R} E$$

$E_{\theta 2} = 0$

$G_{\phi\theta 2} = 0$

where:

t_2 = Thickness of outer layer

$E_{\phi 2}$ = Modulus of elasticity of outer layer in longitudinal direction

$E_{\theta 2}$ = Modulus of elasticity of outer layer in circumferential direction.

$G_{\phi\theta 2}$ = Shear modulus of outer layer.

3.8C.11 PROGRAM 119-CHECK of FLANGE DESIGN

This program is used for the design of bolted flanges. The program checks the flange design based on Appendix II of ASME Code, Section VIII. Bolt and flange stresses are computed for both the bolt-up and design conditions. If the bolt and gasket are not overstressed, the computer automatically calculates the required flange thickness or checks any supplied thickness. The minimum gasket width required to prevent crushing, and the maximum pressure that the flange is capable of resisting under the design conditions are automatically calculated.

3.8C.12 PROGRAM 772-NOZZLE REINFORCEMENT CHECK

This is a program for checking nozzle reinforcing. It is designed essentially for containment vessels, and adheres to area replacement criteria specified by ASME Section III and VIII. The program does no design work, merely checking the adequacy of pre-selected reinforcing plate dimensions and weld sizes.

3.8C.13 PROGRAM 1027-WRC 107 STRESS INTENSITIES AT LOADED ATTACHMENTS FOR SPHERES OR CYLINDERS WITH ROUND OR SQUARE ATTACHMENT

This program determines the stress intensities in a sphere or cylinder at a maximum of 12 points around an externally loaded round or square attachment. Stresses resulting from external loads are superimposed on an initial pressure stress situation. The program computes stresses at three levels of plate thicknesses: outside, inside, and centerline of plate. The 12 points investigated are shown in Figure 3.8C-2. Four points at the edge of attachment, at ½RT from the edge of attachment and at the edge of reinforcement.

The program determines 3 components for each stress intensity:

1. σ_X = A normal stress parallel to the vessel's longitudinal axis

2. σ_ϕ = A normal stress in a circumferential direction

3. τ = A shear stress

The program has an option, whereby the influence coefficients can be calculated directly. The program uses the methodology from the "Welding Research Council Bulletin #107", of December 1968. Additionally, the program contains extrapolations of the curves for cylinders in WRC 107 for gamma up to 600. It should be noted that the use of the program requires complete familiarity with WRC 107 publication.

3.8C.14 PROGRAM 1036M-STRESS INTENSITIES IN JUMBO INSERT PLATES

This program determines the stress intensities in a "Jumbo" insert plate (a reinforcing plate with multiple penetrations) in a cylindrical vessel at 8 points around one of these penetrations due to the loading on that penetration plus the loadings on the 4 adjacent penetrations all as superimposed on an initial stress situation. It does this at three levels of plate thickness: outside, inside, and centerline of plate. The 8 points investigated are shown in Figure 3.8C-3. The 4 points on radius Rθ are at the junction of the penetration and the insert plate. The other 4 points are other points of interest; normally, they will be at the midpoints in the clear space between penetrations or at the edge of reinforcing. Although 5 penetrations are considered, each point is analyzed as though it were only influenced by 2 (the central penetration plus the penetration on the same axis as the point concerned).

The program also determines 3 components for each stress intensity:

1 σ_x = a normal stress parallel to the vessel's longitudinal axis

2 σ_ϕ = a normal stress in a vessel's circumferential direction

3 τ = a shear stress

Each of these is composed of 3 subcomponents:

1. One due to the central penetration's loading

2. One due to the loading on the next adjacent penetration

3. An initial stress component (input)

The program has an option whereby the penetration loads will be considered reversible or nonreversible in direction. Under the reversible option, (see Figure 3.8C-4) only the data associated with the most severe loading situations is printed out.

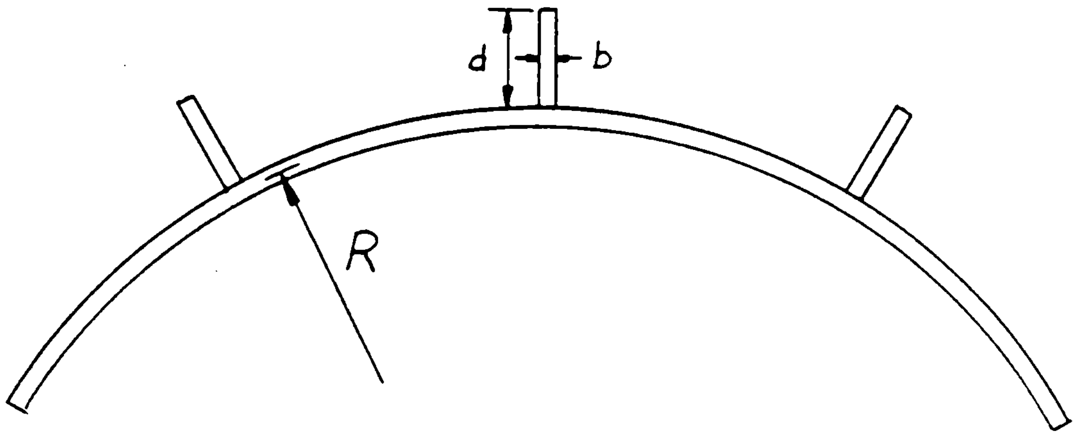
Most of the analysis and notation used in the program is taken directly from the "Welding Research Council Bulletin #107" of December 1968. Use of the program requires complete familiarity with this publication.

The analysis in WRC 107 is for a single penetration. This program analyzes the several penetrations individually, using WRC 107 techniques verbatim, and then through superposition obtains the composite results. The adjacent penetrations must be on a cardinal line of the central penetration in order to use WRC 107 methods. This has required a very conservative extension of the WRC 107 analysis. WRC 107 analysis applies only to the points on the penetration to shell juncture. This program makes stress determinations at points removed from the junction by fictitiously extending the radius of any penetration to any point at which a stress determination is desired. This disregards the statement in WRC 107 that "these stresses attenuate very rapidly at points removed from the penetration to shell juncture". Furthermore, in some cases, the moment induced stresses at both the juncture and at points removed from the juncture are increased by 20% per discussion in WRC 107. Figure 3.8C-5 shows the cases for the 20% increase and indicates the thickness used for the calculation of the parameters (per WRC 107) and stresses.

The program contains extrapolations of the curves in WRC 107 for T up to 600. The program is limited to the domains and range of Figures 1A through 4C in WRC 107 ($0 < \beta \leq 0.5$ and $5 \leq T \leq 600$).

REFERENCE

1. Wilson & Clough, Dynamic Response by Step-By-Step Matrix Analysis



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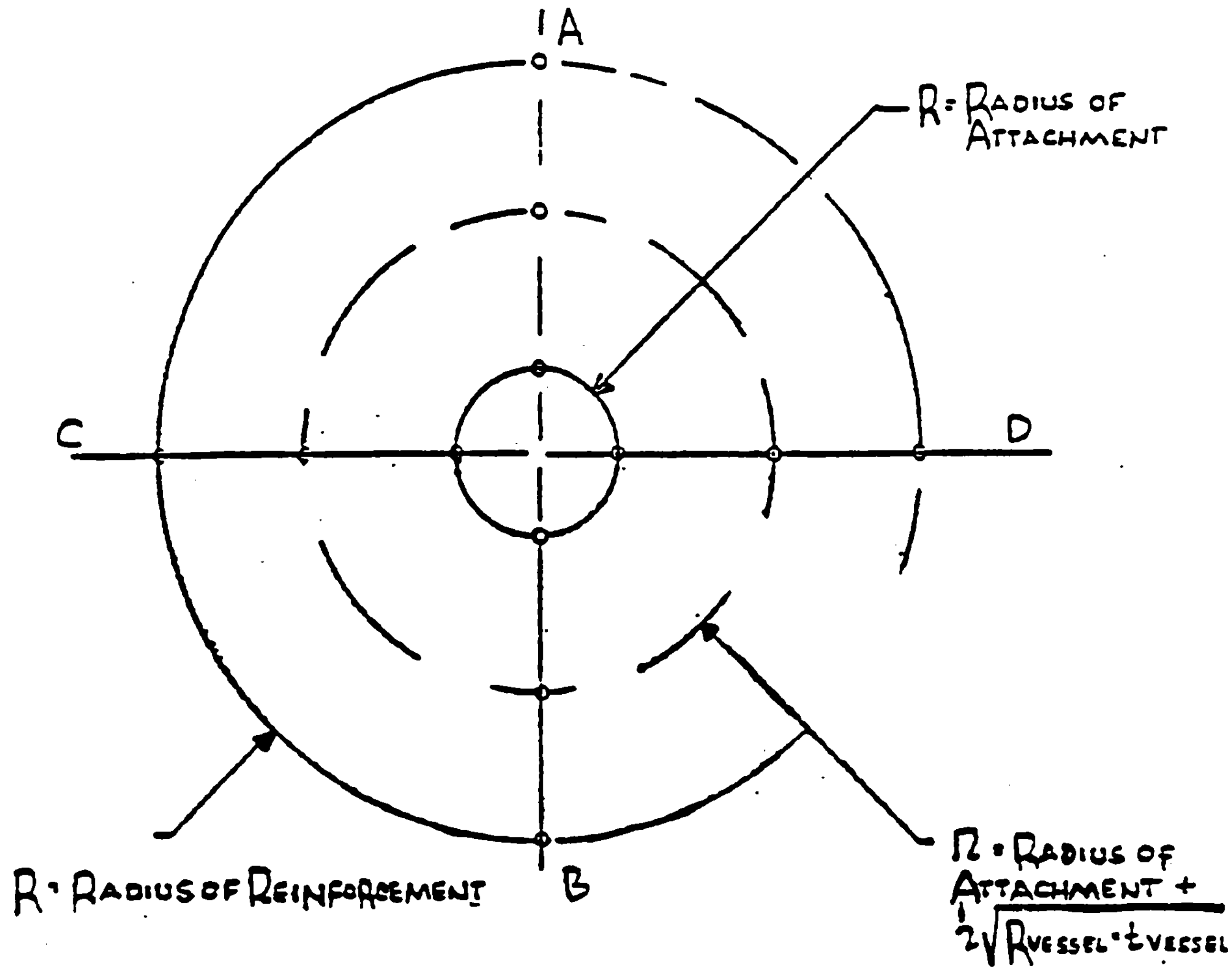
Vertical Stiffener Model

FIGURE 3.8C-1



○ POINTS OF STRESS CALCULATION

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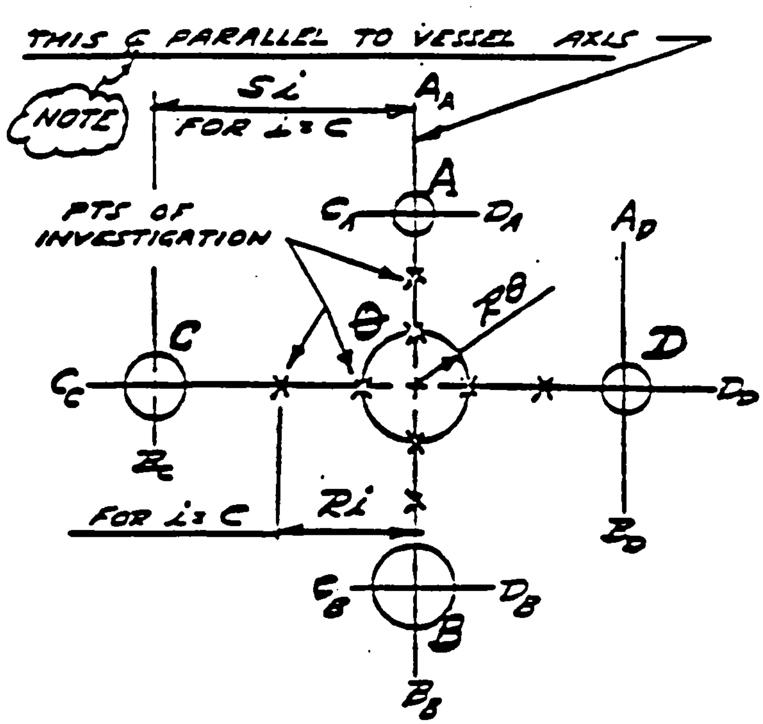
Stress Calculation

FIGURE 3.8C-2



JUMBO INSERT PLATES
POINTS OF STRESS INTENSITIES

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Stress Intensities

FIGURE 3.8C-3

L	P	V_L	V_C	M_L	M_C	M_T
1	+	+	+	±	+	+
2	+	+	+	+	+	−
3	+	+	+	+	−	+
4	+	+	+	+	−	−
5	+	+	+	−	+	+
6	+	+	+	−	+	−
7	+	+	+	−	−	+
8	+	+	+	−	−	−
9	+	+	−	+	+	+
10	+	+	−	+	+	−
11	+	+	−	+	−	+
12	+	+	−	+	−	−
13	+	+	−	−	+	+
14	+	+	−	−	+	−
15	+	+	−	−	−	+
16	+	+	−	−	−	−
17	+	−	+	+	+	+
18	+	−	+	+	+	−
19	+	−	+	+	−	+
20	+	−	+	+	−	−
21	+	−	+	−	+	+
22	+	−	+	−	+	−
23	+	−	+	−	−	+
24	+	−	+	−	−	−
25	+	−	−	+	+	+
26	+	−	−	+	+	−
27	+	−	−	+	−	+
28	+	−	−	+	−	−
29	+	−	−	−	+	+
30	+	−	−	−	+	−
31	+	−	−	−	−	+
32	+	−	−	−	−	−

Note:

L represents the loading situation number.

The program saves the loading situation number that produced a maximum stress stress intensity. Later, it must reconstruct the loading associated with this number for print out purpose.

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Determination of Loads on Center Penetration Associated with Maximum Stress Intensity

FIGURE 3.8C-4

Penetration Analysis

Style or Reinforc.	Location of Analysis	Reference for Curves	Radial M: * Param. / Stress	Radial M: % Incr.	Tang M: * Param. / Stress	Tang M: % Incr.	Radial N: * Param. / Stress	Radial N: % Incr.	Tang N: * Param. / Stress	Tang N: % Incr.	Comments
Insert (Width < $1.65\sqrt{R T_{INS}}$)	Insert to Neck	107	T_{INS} / T_{INS}	20	T_{INS} / T_{INS}	20	T_S / T_{INS}	-	T_S / T_{INS}	-	
	$\frac{1}{2}\sqrt{R T_{INS}}$ from Local Stress	107	T_{INS} / T_{INS}	20	T_{INS} / T_{INS}	20	T_S / T_{INS}	-	T_S / T_{INS}	-	See Note 3
	Insert to Shell	107	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	See Note 4
Pad (Width < $1.65\sqrt{R T_{EQ}}$)	Pad to Neck	107	T_{P+S} / T_{P+S}	20	T_{P+S} / T_{P+S}	20	T_S / T_{P+S}	-	T_S / T_{P+S}	-	
	$\frac{1}{2}\sqrt{R T_S}$ from Local Stress	107	T_{P+S} / T_{P+S}	20	T_{P+S} / T_{P+S}	20	T_S / T_{P+S}	-	T_S / T_{P+S}	-	See Note 3
	Edge of Pad	107	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	See Note 4
Shell (No reinf. or width < $1.65\sqrt{R T_S}$)	Neck to Shell	107	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	
	$\frac{1}{2}\sqrt{R T_S}$ from Neck	107	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	T_S / T_S	-	See Note 3

NOTES:
1) * INDICATES THICKNESS FOR CALCULATION OF PARAMETERS AND STRESSES.
2) ANY ALTERNATE METHOD MAY BE USED FOR A LOCATION OF ANALYSIS IN COMBINATION WITH THE ABOVE METHOD FOR OTHER LOCATIONS.
3) CHECK ONLY IF STRESSES @ OTHER LOCATIONS > 1.1 S_m. USE T_S AND NO INCREASE IN STRESSES IF OUTSIDE REINFORCING.
4) STRESSES DUE TO LOADS MAY BE REDUCED IN ACCORDANCE WITH WRC 95.

NOMENCLATURE

T_{INS} = THICKNESS OF INSERT
T_{EQ} = EQUIVALENT THICKNESS
T_{P+S} = THICKNESS OF PAD PLUS SHELL
T_S = THICKNESS OF SHELL

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Penetration Analysis

FIGURE 3.8C-5

APPENDIX 3.8D

COMPUTER PROGRAMS FOR STRUCTURAL ANALYSIS

Historical Information

[Computer programs used for structural analysis meet the TVA Quality Assurance Program for computer software. The following sections are for historical purposes.]

Computer programs used for structural analysis and design have been validated by one of the following criteria or procedures:

a. The following computer programs are recognized programs in the public domain:

Program	Usage Start Date: Year	Hardware	Source
AMG032	1965	IBM	R&H
AMGO33	1965	IBM	R&H
AMGO34	1965	IBM	R&H
ANSYS	1972	CDC	CDC
ASHSD	1969	IBM	UCB
BASEPLATE II	1982	CDC	CDC
GENDHK 3	1969	IBM	UCB
GENSHL 2	1969	IBM	FIRL
GENSHL 5	1968	IBM	FIRL
GTSTRUDL	1979	CDC	GT
NASTRAN (MSC)	1974	CDC	CDC
SAP IV	1973	CDC	UCB
SAP IV	1974	IBM	USC
SDRC FRAME PACKAGE SAGS/DAGS	1977	CDC	SDRC
SPSTRESS	1977	CDC	CDC
STARDYNE	1977	CDC	CDC
STRESS	1970	EG	CDC
STRUDL (V2 M2)	1972	IBM	ICES
STRUDL (Rel. 2.6) (Dynal)	1974	IBM	MCAUTO
STRUDL (Rel. 4.0)	1975	IBM	MCAUTO
STRUPAK PACKAGE MAP2DF/SAP2DF	1971	CDC	TRW
SUPERB	1977	CDC	CDC
WELDDA	1983	CDC	CDC
WERCO	1978	CDC	AAA

All programs on IBM hardware are run under the MVS operating system, on either a 370/165 machine or a 360150 machine. All programs on CDC hardware are run under the SCOPE 3.3 operating system on a 6600 machine.

The following abbreviations are used for program sources:

CDC - Control Data Corporation, Minneapolis, MI
FIRL - Franklin Institute Research Labs, Philadelphia, PA
GT - Georgia Institute of Technology, Atlanta, GA
ICES - Integrated Civil Engineering System, Worcester, MA
MCAUTO - McDonnell-Douglas Automation Company, St. Louis, MO
R&H - Rohm & Haas Company, Huntsville, AL
SDRC - Structural Dynamics Research Corporation, Cincinnati, OH
TRW - TRW Systems Group, Redondo, CA
UCB - University of California, Berkely, CA
USC - University of Southern California, Los Angeles, CA
AAA - AAA Technology and Specialties Co., Inc., Houston, TX

b. The following programs have been validated by comparison with a program in the public domain:

RESPONSE FOR EARTHQUAKE AVERAGING SPECTRAL RESPONSE

Summary comparisons of results for these computer programs are provided in Figures 3.8D-l and 3.8D-2.

c. The following programs have been validated by comparison with hand calculations:

BIAXIAL BENDING - USD
CONCRETE STRESS ANALYSIS
DL42
PLTDL42
THERMCYL
TORSIONAL DYNANAL
PNA100

The following programs have been validated by comparison with analytical results published in the technical literature:

BAP222
DYNANAL
ROCKING DYNANAL

Summary comparison of results for these computer programs are provided in Tables 3.8D-l through 3.8D-10.

WBN

TABLE 3.8.D-1

BIAXIAL BENDING - USD

Moment Capacity (FT-KIPS)			
M_X		M_Y	
Hand Calculations	Program	Hand Calculations	Program
0	0	409	408
601	603	287	285
850	850	164	165
911	909	77	76
933	932	0	0

Comparison of hand calculations with BIAXIAL BENDING - USD for the moment capacities of a reinforced concrete section for a given direct load.

WBN

TABLE 3.8D-2

CONCRETE STRESS ANALYSIS

Concrete Compression Stress (psi)	
Hand Calculations	Program
436.	436.

	Steel Tensile Stress (psi)	
	Hand Calculations	Program
1	-3833	-3830
2	-2238	-2234
3	- 644	- 639
4	950	957
5	2417	2419
6	3884	3881
7	5478	5477
8	6275	6275
9	11053	11061

Comparison of hand calculations with CONCRETE STRESS ANALYSIS for reinforced concrete beam with 9 rows of steel, subject to combined load of moment and axial force.

WBN

TABLE 3.8D-3

THERMCYL

Dead Load (psi)	Maximum Concrete Compression Stress (psi)		Steel Tensile Stress (psi)	
	Hand Calculations	Program	Hand Calculations	Program
0	770.8	770.9	12,948	12,950
10	848.8	848.3	12,285	12,290
100	1313.0	1316.0	8,336	8,311
1000	2795.0	2793.0	-5,010	-4,990

Comparison of hand calculations with THERMCYL results for stresses in reinforced concrete thin-walled cylinder with non-linear temperature distribution across wall thickness and varying dead load axial stress.

WBN

TABLE 3.8.D-4

TORSIONAL DYNANAL

Pure Torsion Modal Frequencies		
Mode No.	Frequency (RAD./SEC.)	
	Hand Calculations	Program
1	2810	2814
2	8430	8430

Comparison of hand calculations with TORSIONAL DYNANAL results for torsional modes of vibration of a thin-walled steel half-tube.

WBN

TABLE 3.8D-5

DYNANAL

Modal Periods Including Effects of Flexural and Shear Deformations		
Mode No.	Period (SEC)	
	Published Results	Program
1	1.48	1.50
2	.425	.430
3	.216	.222
4	.149	.157
5	.114	.124

Comparison of DYNANAL with analytical procedure presented in Engineering Vibrations, L. S. Jacobsen and R. S. Ayre, McGraw-Hill, 1958, Chapter 10, Modal Analysis of 200 Ft. shear-wall building including effects of flexural and shear deformations.

WBN

TABLE 3.8D-6

ROCKING DYNANAL

Modal Frequencies of Lumped-Mass Shear Beam Including effects of Base Rocking		
Mode No.	Frequency (RAD./SEC.)	
	Published Results	Program
1	5.155	5.339
2	20.52	19.226

Comparison of ROCKING DYNANAL with Analytical Procedure presented in "Earthquake Stresses in Shear Buildings," M. G. Salvadori, ASCE Transactions, 1953, Paper No. 2666. Modal analysis of lumped-mass shear beam including effects of base rocking.

WBN

TABLE 3.8D-7

BAP222

Comparison of BAP222 with analytical procedure presented in A Simple Analysis for Eccentrically Loaded Concrete Sections, L. G. Parker and J. J. Scanion, Civil Engineering, October 1940

	Published Results	Program
Pressure bulb geometry, Z4	12 (in.)	12 (in.)
Pressure bulb geometry, Z5	6.41 (in.)	6.41 (in.)
Pressure bulb geometry, Z7	3.67 (in.)	3.36 (in.)
Concrete pressure force	-14.08 (k)	-14.48 (k)
Anchor load 1	-1.715 (k)	-1.65 (k)
Anchor load 2	5.34 (k)	5.4 (k)
Anchor load 3	-3.22 (k)	-3.44 (k)
Anchor load 4	3.665 (k)	3.61 (k)

WBN

TABLE 3.8D-8

DL42

Comparison of hand calculations with DL42 for the design of a baseplate resisting a given load.

	Hand Calculations	Program
Safety factor (0.5 SSE)	3.232	3.234
Safety factor (SSE)	3.878	3.881
Maximum plate moment	10.535 (k-in)	10.526
Effective section modulus	1.261 (in.)	1.261
Minimum plate thickness	0.417 (in.)	0.417

WBN

TABLE 3.8D-9
(Sheet 1 of 1)

PLTDL42

Comparison of hand calculations with PLTDL42 for the design of a baseplate resisting a given load.

	Hand Calculations	Program
Safety factor (0.5 SSE)	3.232	3.234
Safety factor (SSE)	3.878	3.881
Maximum plate moment	10.535 (k-in)	10.526
Effective section modulus	1.261 (in.)	1.261
Minimum plate thickness	0.417 (in.)	0.417
Plate bending stress	8.355 (k/in)	8.348

WBN

TABLE 3.8D-10

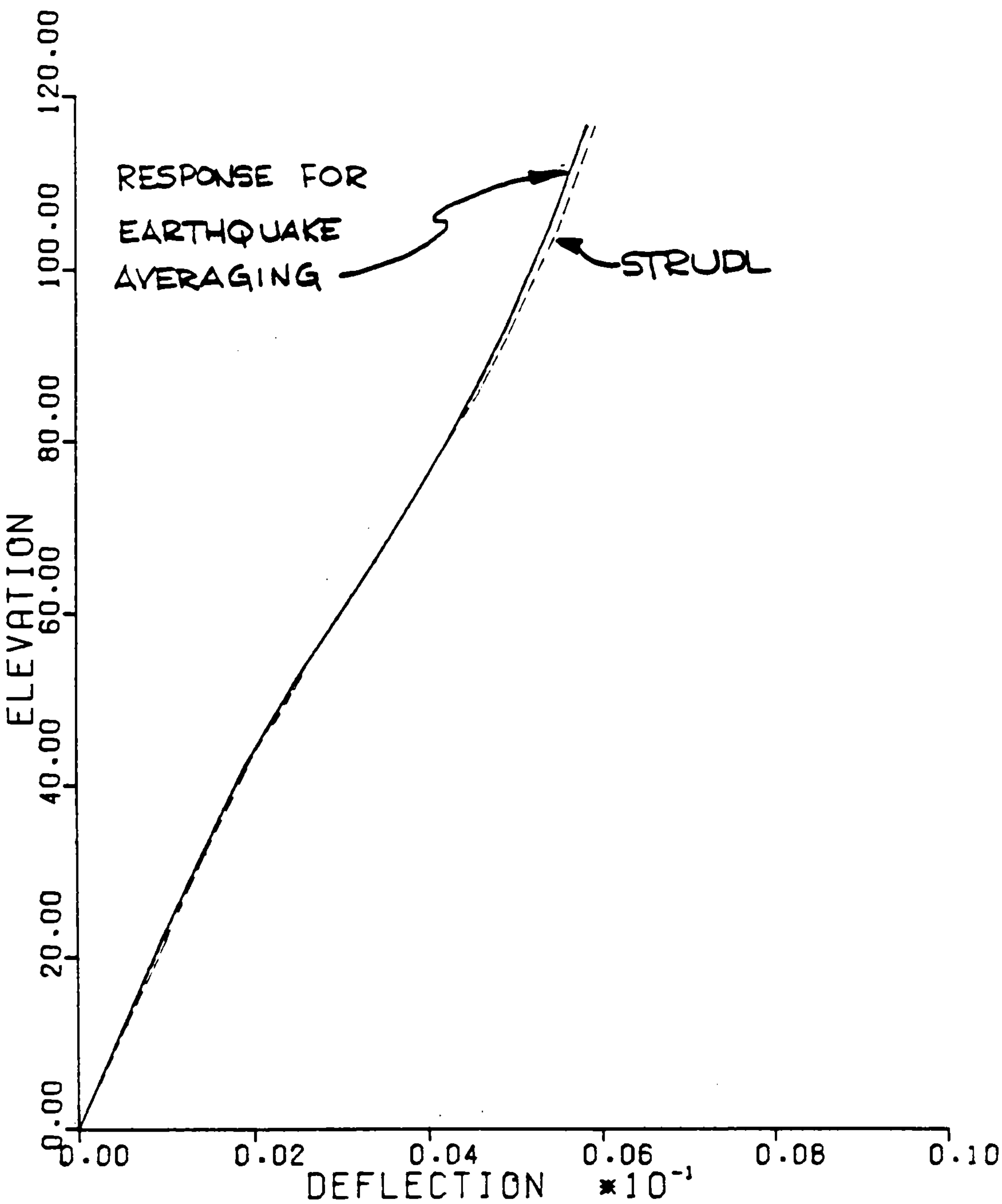
PNA 100
NOZZLE STRESSES (PEN X-57)

NEXT TO SHELL

Case	Calc. Mode	A	B	C	D
1	Program	11,039	16,588	11,224	16,495
	Hand	11,036	16,584	11,221	16,491
4	Program	13,074	19,192	12,417	17,974
	Hand	13,070	19,187	12,412	17,968

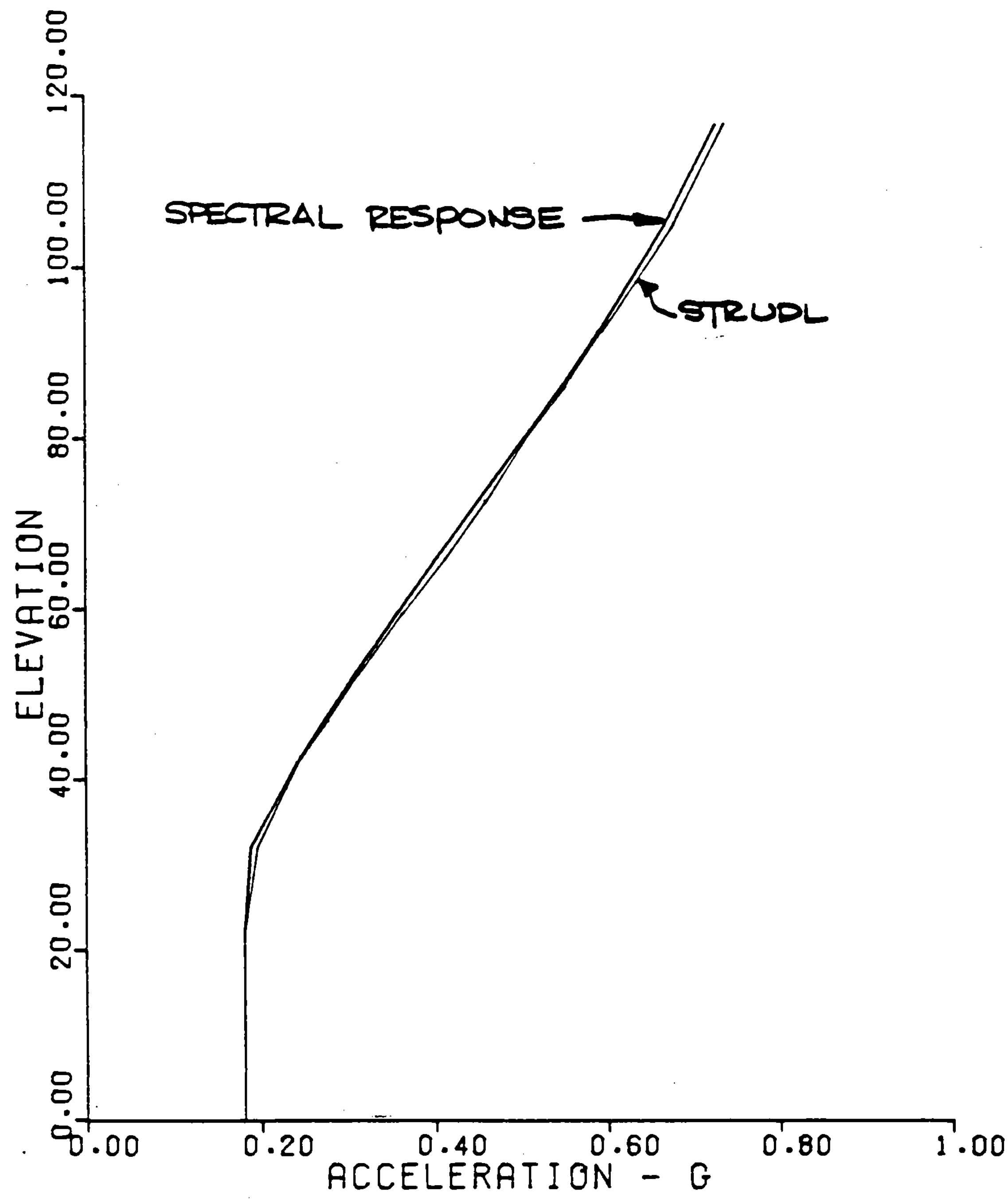
AWAY FROM SHELL

Case	Calc. Mode	A	B	C	D
1	Program	10,358	10,095	10,571	10,330
	Hand	10,354	10,090	10,567	10,327
4	Program	12,944	12,621	12,196	11,915
	Hand	12,939	12,616	12,190	11,908



Comparison of STRUDL with RESPONSE FOR EARTHQUAKE AVERAGING for a normal-mode time-history analysis of a lumped-mass structural model of a nuclear power plant structure subjected to the 1940 El Centro earthquake N-S ground motion.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Response for Earthquake Averaging FIGURE 3.8D-1



Comparison of STRUDL with SPECTRAL RESPONSE for a response spectrum analysis of a lumped-mass structural model of a nuclear power plant structure subjected to the 1940 El Centro earthquake N-S ground motion.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Spectral Response FIGURE 3.8D-2

APPENDIX 3.8E

CODES, LOAD DEFINITIONS AND LOAD COMBINATIONS FOR THE MODIFICATION AND EVALUATION OF EXISTING STRUCTURES AND FOR THE DESIGN OF NEW FEATURES ADDED TO EXISTING STRUCTURES AND THE DESIGN OF STRUCTURES INITIATED AFTER JULY 1979

3.8E.1 Application Codes and Standards

a. American Concrete Institute (ACI) 318-77, "Building Code Requirements for Reinforced Concrete"

b. American Institute of Steel Construction (AISC), "Specification for the Design Fabrication, and Erection of Structural Steel for Buildings," 7th edition adopted February 12, 1969, as amended through June 12, 1974 or later editions, except welded construction is in accordance with Item d below.

c. American Society for Testing and Materials (ASTM) Standards

d. American Welding Society (AWS), Structural Welding Code, AWS D1.1-72, with Revisions 1-73 and 2-74 except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders.

Visual inspection of structural welds will meet the minimum requirements of Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 as specified on the design drawings or other engineering design output (See Section 3.8.4.1.1, Item 18).

e. National Fire Protection Association Standard NFPA 13

f. National Fire Protection Association Standard NFPA 14

g. National Fire Protection Association Standard NFPA 15

h. National Fire Protection Association Standard NFPA 24

i. National Fire Protection Association Standard NFPA 30

j. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections III, VIII, and IX

k. American Nuclear Standard Institute (ANSI) B31.1, "Power Piping"

l. AWS D1.1-81, "Structural Welding Code"

m. AISC-ANSI-N690-1984 "Nuclear Facilities Steel Safety-Related Structures for Design, Fabrication and Erection"

3.8E.2 Load Definitions

The following terms are used in the load combination equations for structures.

Normal loads, which are those loads to be encountered during normal plant operation and shutdown, include:

D Dead loads or their related internal moments and forces including any permanent equipment loads; all hydrostatic loads; and earth loads applied to horizontal surfaces.

L Live loads or their related internal moments and forces including any movable equipment loads and other loads which vary with intensity and occurrence, such as lateral soil pressure.

T_o Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

R_o Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

Severe environmental loads include:

E Loads generated by the operating basis earthquake (OBE). The term "operating basis earthquake" has the same meaning as "one-half safe shutdown earthquake."

W Load generated by the design wind specified for the plant.

Extreme environmental loads include:

E' Load generated by the safe shutdown earthquake (SSE). The term "safe shutdown earthquake" has the same meaning as the term "design basis earthquake" (DBE).

W_t Loads generated by the design tornado specified for the plant. Tornado loads include loads due to the tornado wind pressure, the tornado-created differential pressure, and to tornado-generated missiles.

Abnormal loads, which are those loads generated by a postulated high-energy pipe break accident, include:

P_a Pressure equivalent static load within or across a compartment generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

T_a Thermal loads under thermal conditions generated by the postulated break and including T_o.

R_a Pipe reactions under thermal conditions generated by the postulated break and including R_o

Y_r Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Other loads:

C Construction live loads

F' Hydrostatic load from the probable maximum flood

F_a Flood load generated by a postulated pipe break

Concrete capacity:

U Concrete section strength required to resist design loads based on the strength design methods described in ACI 318-77.

3.8E.3 Load Combinations - Concrete

a. For service load conditions, the strength design method is used, and the following load combinations are considered.

(1) U = 1.4 D + 1.7 L

(2) U = 1.4 D + 1.7 L + 1.9 E

(3) U = 1.4 D + 1.7 L + 1.7 W

If thermal stresses due to T_o and R_o are present, the following combinations are also considered.

(1a) $U = (0.75)\ (1.4\ D + 1.7\ L + 1.7\ T_o + 1.7\ R_o)$

(2a) $U = (0.75)\ (1.4\ D + 1.7\ L + 1.9\ E + 1.7\ T_o + 1.7\ R_o)$

(3a) $U = (0.75)\ (1.4\ D + 1.7\ L + 1.7\ W + 1.7\ T_o + 1.7\ R_o)$

Both cases of L having its full value or being completely absent are checked. In addition, the following combinations are considered:

(2a') $U = 1.2\ D + 1.9\ E$

(3a') $U = 1.2\ D + 1.7\ W$

b. For factored load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental and abnormal/extreme environmental conditions, the strength design method is used; and the following load combinations are considered:

(4) $U = D + L + T_o + R_o + E'$

(5) $U = D + L + T_o + R_o + W_t$

(6) $U = D + L + T_a + R_a + 1.5\ P_a$

(7) $U = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

(8) $U = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

For the Additional Diesel Generator Building and structures initiated after July 1979, the three individual tornado-generated loads are combined as follows:

$W_t = W_w$

$W_t = W_p$

$W_t = W_m$

$W_t = W_w + 0.5W_p$

$W_t = W_w + W_m$

$W_t = W_w + 0.5\ W_p + W_m$

where:

W_t is the total tornado load,

W_w is the tornado wind load,

W_p is the tornado-generated pressure differential load, and

W_m is the tornado missile load.

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, is used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8), are satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these concentrated loads, local section strength capacities may be exceeded provided there is no loss of function of any safety-related system.

c. Other load conditions:

(9) $U = 1.4\ D + 1.4\ C$

(10) $U = D + L + F'$

(11) $U = D + F_a$

3.8E.4 Load Combinations - Structural Steel

a. For service load conditions, the elastic working stress design methods of Part 1 of the AISC specifications is used and the following load combinations are considered.

Allowable Stress	Load Combinations
(1) AISC Allowable*	D + L
(2) AISC Allowable*	D + L + E
(3) AISC Allowable*	D + L + W

*See Table 3.8E-1 for limiting values

If thermal stresses due to T_o and R_o are present, the following combinations are also considered:

Allowable Stress	Load Combinations
(1a) 1.5 x AISC Allowable*	$D + L + T_o + R_o$
(2a) 1.5 x AISC Allowable*	$D + L + T_o + R_o + E$
(3a) 1.5 x AISC Allowable*	$D + L + T_o + R_o + W$

* The allowable stress shall be limited to the values given in Table 3.8E-1.

Both cases of L having its full value or being completely absent, are checked.

b. For factored load conditions, the following load combinations are considered.

Allowable Stress	Load Combinations
(4) 1.6 x AISC Allowable*	$D + L + T_o + R_o + E'$
(5) 1.6 X AISC Allowable*	$D + L + T_o + R_o + W_t$
(6) 1.6 x AISC Allowable*	$D + L + T_a + R_a + P_a$
(7) 1.6 x AISC Allowable*	$D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) + E$

Allowable Stress	Load Combinations
(8) 1.7 x AISC Allowable*	$D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) + E'$
(9) 1.6 x AISC Allowable*	$D + F_a$
(10) 1.6 x AISC Allowable*	$D + L + E'$
(11) 1.6 x AISC Allowable*	$D + L + W_t$

* If thermal loads are not present, the allowable stress shall be limited to the values given in Table 3.8E-1.

Evaluations of miscellaneous and structural steel designed prior to July 1979, may be performed using load combinations (2), (10), and (11) unless other specific loads of a significant nature exist, in which case, the appropriate load combinations of Section 3.8E.4 must be considered. The design of modifications must meet the load combinations in Section 3.8E.4.

Thermal analyses using linear elastic methods are performed for restrained Category I structures located in high temperature environments.

In combinations (6), (7), and (8), the maximum values of P_a, R_a, T_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, was used unless a time-history analysis was performed to justify otherwise. Combinations (5), (7), and (8) were first satisfied without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8).

TABLE 3.8E-1 (Sheet 1 of 2)

LIMITING VALUES OF ALLOWABLE STRESS

Loading Combinations	Tension on Net Section	Shear on Gross Section	Compression on Section	Bending
(1), (2), (3)	$0.60F_y$	$0.40F_y$	See Note 1	See Note 2
(1a), (2a), (3a) (4) through (9)	$0.90F_y$	$\frac{0.90F_y}{\sqrt{3}}$	See Note 3	$0.90F_y$

Note 1 - Varies with slenderness ratio, see AISC "Manual of Steel Construction," 7th Edition, Table 1-36, Page 5-84.

Note 2 - Varies, see Section 1.5.1.4, "Bending", of Item 3.8E.1.b

Note 3 - Varies with slenderness ratio. The allowable stress was obtained from AISC Specification Section 1.5, using formula 1.5-1 or 1.5-2 and 1.5-3 with modifications, as shown below:

Main and secondary

members where $K\ell / r \leq C_c: \quad F_a = 0.9F_y\left[1 - \frac{(\frac{K\ell}{r})^2}{2C_c^2}\right]$ (Formula A)

Main members where $C_c < K\ell / r < 200: \quad F_a = \frac{0.9\pi^2 E}{\left(\frac{K\ell}{r}\right)^2}$ (Formula B)

Secondary members where $120 < K\ell / r \leq 200: \quad F_{as} = \frac{F_a [\text{by Formula (A) or (B)}]}{1.6 - \frac{\ell}{200r}}$

Where:

$$C_c = \sqrt{\frac{2\pi^2 E}{F_y}}$$

E = Modulus of elasticity of steel

TABLE 3.8E-1 (Sheet 2 of 2)

LIMITING VALUES OF ALLOWABLE STRESS

F_a = Axial compressive stress permitted in the absence of bending moment (kips per square inch)

F_{as} = Axial compressive stress, permitted in the absence of bending moment, for bracing and other secondary members (kips per square inch)

F_y = Specified minimum yield stress of materials (kips per square inch)

K = Effective length factor

ℓ = Actual unbraced length (inches)

r = Governing radius of gyration (inches)

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 General Topic for Analysis of Seismic Category I ASME Code and Non-Code Items

3.9.1.1 Design Transients

Transients used in the design and fatigue analysis for Westinghouse supplied ASME Code Class 1 components and RCS components are discussed and presented in Section 5.2.1.5. Specifically, the transients are identified for Class 1 components in Tables 5.2-2 and 5.2-3. The transients used in the design and analysis of RCS components are identified in Table 5.2-2.

3.9.1.2 Computer Programs Used in Analysis and Design

3.9.1.2.1 Other Than NSSS Systems, Components, Equipment, and Supports

1. The following computer programs are used in piping analyses:

a. TPIPE Program - TPIPE is a special purpose computer program capable of performing static and dynamic linear elastic analyses of power-related piping systems. The dynamic analysis option includes: (1) frequency extraction, (2) response spectrum, (3) time history modal superposition, and (4) time history direct integration methods.

   In addition to these basic analysis capabilities, the program can perform an ASME Section III, Class 1, 2, or 3 stress evaluation and perform thermal transient heat analysis to provide the linear thermal gradient, ΔT1, nonlinear thermal gradient, ΔT2, and gross discontinuity expansion difference, $\alpha_a T_a - \alpha_b T_b$, required for a Class 1 stress evaluation.

   This program is owned and maintained by TVA. It has been fully verified and documented and was compared with PISOL, SAP IV, PIPSD, STARDYNE, and SUPERPIPE with excellent correlation. These programs are well recognized and utilized throughout the industry. It is maintained and updates are verified in accordance with the TVA Quality Assurance Program for Computer Software.

b. Post Processors - The post processors are used in performing the stress evaluations and support load calculations made in the analysis of piping systems.

   The programs use moment, force, and deflection data generated by TPIPE. A stress evaluation is made for each joint on the analysis model. The appropriate stress intensifications/stress indices according to the ASME Section III code are utilized in evaluating stresses for the Normal, Upset, Emergency, and Faulted Conditions. Pipe rupture limits and active valve limits are also evaluated. The allowed stress difference for pipe lug attachments and the lug load is calculated for each load condition.

Support and anchor design loads are calculated for each support to meet the requirements given in Section 3.9.3.4.2.

c. The following computer programs are also used for piping analysis:

Program	Source	Program Description
ME-101	BECHTEL	Linear elastic analysis of piping systems - Bechtel Western Power Corp San Francisco, CA.
ANSYS	SWANSON	General purpose finite element program - Swanson Analysis Systems, Inc. Houston, PA.

2. The following computer programs are used in support design and equipment/component analysis.

ACRONYM	PROGRAM DESCRIPTION
FAPPS (ME150)	Frame Analysis For Pipe Supports
SMAPPS (ME152)	Standard Frame Analysis For Small Bore Pipe Supports
MAPPS (ME153)	Miscellaneous Applications For Pipe Supports
IAP	Integral Welded Attachments
CONAN	Allowable Tensile Load For Anchor Bolt Group With Shear Cone Overlap
BASEPLATE II	Finite Element Analysis Of Base Plates And Anchor Bolts
GT STRUDL	Structural Analysis Program
CASD	TVA Computer Aided Support Design Program
SUPERSAP	Structural Finite Element Analysis Program
ANSYS	Structural Finite Element Analysis Program
STARDYNE	Structural Analysis Program

3.9.1.2.2 Programs Used for Category I Components of NSSS

Computer programs that Westinghouse uses in analysis to determine structural and functional integrity of Seismic Category I systems, components, equipment and supports are presented in WCAP-8252, Revision 1[1] and WCAP-8929 [10].

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis was used per se, for the reactor internals. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in the following Sections 3.9.2.4, 3.9.2.5, and 3.9.2.6.

3.9.1.4 Consideration for the Evaluation of the Faulted Condition

3.9.1.4.1 Subsystems and Components Analyzed by Westinghouse

The analytical methods used to evaluate stresses for ASME Class 1 systems and components are presented in Section 5.2.1.10. The results of the analyses are documented in the stress reports that describe the system or component.

For reactor internals the faulted condition was evaluated based on a non-linear elastic system analysis and conforms to the requirements of Appendix F of the ASME Code Section III. Analytical methods are described in Section 3.9.2.5.

3.9.1.4.2 Subsystems and Components Analyzed by TVA

1. Piping Systems - The methods employed in the analysis of ASME Class 1 and Class 2/3 piping systems are elastic analytical methods as described by the equations of Sections NB-3600 and NC-3600 of the ASME Code.

   The faulted condition stress limits specified for Class 1 and Class 2/3 systems are in compliance with the elastic method limits set forth in Appendix F subsection F-1360 of the ASME Section III Code.

2. Piping System Supports - The methods employed in the analysis of ASME Code Classes 1, 2, and 3 piping system supports are as follows:

   a. Linear Type - Elastic methods as described by Part I of the AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," February 12, 1969. (Supplements 1, 2, & 3) (Later edition of the AISC code may be utilized when design safety is not compromised.)

   b. Standard Components - Elastic or load-rated methods as described by Manufacturers' Standardization Society (MSS) SP-58, 1967 edition, "Pipe Hangers and Supports."

The faulted condition stress limits for Class 1, 2, and 3 pipe supports are specified in Section 3.9.3.4.2. For linear supports these faulted condition limits meet the intent and requirements of the elastic method limits set forth in Appendix F, subsection F-1320 or F-1370, of the ASME Section III Code. See Section 3.9.3.4.2. For standard components, the allowable stresses or load ratings of MSS-SP-58 are based on a factor of safety of five based on normal operating conditions. Upset, emergency, faulted, and test conditions were evaluated using Table 3.9-21. This low allowable stress is adequate to assure that active components are properly supported for faulted conditions.

3. Mechanical Equipment

No plastic instability allowable limits given in ASME Section III have been used when dynamic analysis is performed. The limit analysis methods have the limits established by ASME Section III for Normal, Upset and Emergency Conditions. For these cases, the limits are sufficiently low to assure that the elastic system analysis is not invalidated. For ASME Code Class 1 mechanical equipment, the stress limits for faulted loading conditions are specified in Sections 3.9.3.1.2 and 5.2. For ASME Code Class 2 and 3 mechanical equipment the stress limits for faulted loading conditions are specified in Section 3.9.3.1.2. These faulted condition limits are established in such a manner that there is equivalence with the adopted elastic system analysis. Particular cases of concern are checked by readjusting the elastic system analysis.

4. Mechanical Equipment Supports

The stress limits for the faulted loading condition of mechanical equipment supports are given in Section 3.9.3.4.1 of Westinghouse's scope of supply, and Section 3.9.3.4.2 for TVA's scope of supply.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

ASME Code Section III, Subparagraph NB-3622.3, "Vibration," requires that vibration effects in piping systems shall be visually observed and where questionable shall be measured and corrected as necessary.

The preoperational piping dynamic effects test program at this plant is as follows:

a. The dynamic (steady state and transient) behavior of safety related piping systems designated as ASME Class 1, 2, and 3 is observed during the preoperational testing program. Sample and instrument lines beyond the root valves are normally not included. Also included in the program are those portions of ANSI B31.1 piping which has a potential to exhibit excessive vibrations.

b. Preoperational tests involving critical piping systems will be in compliance with Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

c. For the piping systems discussed in Item a., visual observation of the piping will be performed by trained personnel during predetermined, steady-state and transient modes of operation. The maximum point(s) of representative vibration, as determined by the visual observation, will be instrumented and measurement will be taken to determine actual magnitudes, if it is judged to be excessive.

d. The allowable criteria for measurements shall be either a maximum half-amplitude displacement or velocity value based on an endurance limit stress as defined in the ASME B&PV Code (refer to Section 3.7.3.8.1).

e. Should the measured magnitudes actually exceed the allowable, corrective measures will be performed for the piping system. Any new restraints, as required by corrective measures, will be incorporated into the piping system analysis.

f. The flow mode which produced the excessive vibrations will be repeated to assure that vibrations have been reduced to an acceptable level.

g. The flow modes to which the system components will be subjected are defined, in general terms, in the preoperational test program.

h. Vibration measurements will also be taken on the vital pumps at baseline and on a periodic basis so that excessive vibration can be corrected early in the program and/or detected if it gradually becomes a problem.

i. Vibrations of the affected portions of the main steam system during MS isolation valve trip will be tested and the results will be evaluated.

j. Thermal expansion tests will be conducted on the following piping systems:

Reactor Coolant System (RCS)
Main Steam
Steam Supply to Auxiliary Feedwater Pump Turbine
Main Feedwater
Pressurizer Relief Line
RHR in Shutdown Cooling Mode
Steam Generator Blowdown
Safety Injection System (those lines adjoining RCS which experience temperature > 200°F)
Auxiliary Feedwater
CVCS (Charging line from Regen. Hx to RCS, Letdown Line from RCS to Letdown Hx)

During the thermal expansion test, pipe deflections will be measured or observed at various locations based on the location of snubbers and hangers and expected large displacement. One complete thermal cycle (i.e., cold position to hot position to cold position) will be monitored. For most systems, the thermal expansion will be monitored at cold conditions and at normal operating temperature. Intermediate temperatures are generally not practical due to the short time during which the normal operating temperature is reached. For the RCS and the main steam system, measurements will be made at cold, 250°F, 350°F, 450°F and normal operating temperatures.

Acceptance criteria for the thermal expansion test verify that the piping system is free to expand thermally (i.e., piping does not bind or lock at spring hangers and snubbers nor interfere with structures or other piping), and to confirm that piping displacements do not exceed design limits, as described by ASME Section III (i.e., the induced stresses do not exceed the sum of the basic material allowable stress at design temperature and the allowable stress range for expansion stresses).

If thermal motion is not as predicted, the support system will be examined to verify correct function or to locate points of binding of restraints. If binding is found, the restraints will be adjusted to eliminate the unacceptable condition or reanalyzed to verify that the existing condition is acceptable.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

Design of Category I mechanical equipment to withstand seismic, accident, and operational vibratory loadings is provided either by analysis or dynamic testing.

Generally tests are run with either of the following two objectives:

1. To obtain information on parts or systems necessary to perform the required analysis, or

2. To prove the design (stress or operability) adequacy of a given equipment or structure without performing any analysis of this particular equipment or structure.

The need for the first type of tests is dictated by lack of information on some of the inputs vital to the performance of an analysis. These tests can be either static (to obtain spring constants) or dynamic (to obtain impedance characteristics).

The need for the second type of test is mainly dictated by the complexity of the structure/equipment under design. This vibration testing is usually performed in a laboratory or shop on a prototype basis, using various sources of energy.

For general seismic qualification requirements for mechanical and electrical equipment, see Section 3.7.3.16.

Laboratory vibration testing can be conducted by employing various forms of shakers, the variation depending on the source of the driving force. Generally, the primary source of motion may be electromagnetic, mechanical, or hydraulic-pneumatic. Each is subject to inherent limitations which usually dictate the choice.

To properly simulate the seismic disturbance, the waveform must be carefully defined. The waveform seen by a given piece of equipment depends on:

1. The earthquake motion specified for a given site.

2. The soil-structure interaction.

3. The building in which the component is housed.

4. The floor on which the equipment is located.

5. The support and attachments to the equipment.

Components located on rocks or on stiff lower floors of buildings founded on rock are subjected to random-type vibrations. Components located on the upper floors of flexible buildings, in flexible subsystems, or in buildings on soft foundations are roughly subjected to sine beats with a frequency close to fundamental frequency of the building or subsystem.

In cases where random vibration inputs are used, extreme care is paid to the selection of random forcing functions having frequency content and energy conservatively approaching those of the ground or buildings motion caused by the specified earthquake(s).

The most common and readily available vibration testing facilities could only carry simple harmonic motion. By analytical comparison with time history response obtained with a number of real earthquake motions, it has been found that these time histories can be approximately simulated with wave forms having the shape of sine beats with 5 or 10 cycles per beat, a frequency equal to the component natural frequencies, and maximum amplitude equal to the maximum seismic acceleration to which the component needs to be qualified. For equipment located on building floors, the maximum seismic input acceleration is the maximum floor acceleration. This is obtained from the dynamic analysis of the building or from the appropriate floor response spectrum at the zero period of the equipment.

The above procedure adheres closely to the IEEE 344-1971 "IEEE Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations." This standard was specified for equipment for the Watts Bar Nuclear Plant contracted for up to September 1, 1974. New contracts after this date specified IEEE 344-1975 "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The first test to the IEEE 344-1975 was run in March 1974 on 6.9 kV switch gear. On local panels, test qualification to both standards was used because some instruments and controls mounted there on were procured to each version. This one test revealed that the 1971 version of IEEE 344 was the more severe.

As an example, seismic qualification and the demonstration of operability of active Class 2 and 3 pumps, active Class 1, 2, or 3 valves, and their respective drives, operators and vital auxiliary equipment is shown by satisfying the criteria given in Section 3.9.3.2. Other active mechanical equipment will be shown operable by either testing, analysis or a combination of testing and analysis. The operability programs implemented on this other active equipment will be similar to the program described in Section 3.9.3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10 for electrical equipment will be used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive Seismic Category I equipment will be shown to have structural integrity during all plant conditions in one of the following manners: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment, or 2) by test showing that the equipment retains its structural integrity under the simulated seismic environment.

A list of Category I mechanical equipment and the original method of qualification is provided in the Table 3.7-25.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. Therefore, the determination of the forcing functions as a direct correlation of pressure oscillations can not be practically performed independently of the dynamic characteristics of the structure. The main objective, then, is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence, and pump-related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions.

The effects of these forcing functions have been studied from test runs on models, prototype plants and in component tests. [2,4,5]

The Indian Point Unit 2 plant has been established as the prototype for four-loop plant internals verification program and was fully instrumented and tested during initial startup.[4] In addition, the Sequoyah Unit 1 and Trojan Nuclear Plants have also been instrumented to provide prototype data applicable to Watts Bar.[5]

Although the Watts Bar plant is similar to Indian Point Unit 2, significant differences are the modifications resulting from the use of 17 x 17 fuel, the replacement of the annular thermal shield with neutron shielding panels, and reactor vessel barrel/baffle upflow flow design. These differences are addressed below.

1. 17 x 17 Fuel

   The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod drive line. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation is expected from the 15 x 15 fuel assembly vibration characteristics.

2. Neutron Shielding Pads Lower Internals

   The primary cause of core barrel excitation is flow turbulence, which is not affected by the upper internals [3]. The vibration levels due to core barrel excitation for Trojan and Watts Bar both having neutron shielding pads, are expected to be similar. Since Watts Bar has greater velocities than Trojan, vibration levels due to the core barrel excitation is expected to be somewhat greater than that for Trojan (proportional to flow velocity raised to a small power). However, scale model test results and preliminary results from Trojan show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel flange stresses with large safety margins were measured at Indian Point Unit 2 (thermal shield configuration) lead to the conclusion that stresses approximately equal to those of Indian Point Unit 2 will result on the Watts Bar internals with the attendant large safety margins.

3. Reactor Vessel Barrel/Baffle Upflow Conversion

The upflow conversion consists of changes to the reactor vessel components, which are to plug the core barrel inlet flow holes and to provide holes in the top former plate. These modifications change the flow path from being down flow to upflow between the core barrel and the baffle plate and increase core bypass flow by 1.5%. Changing the flow path reduces the pressure differential across the baffle plate, eliminating the jetting of coolant between the joints between the baffle plates. Although defined as a difference between Indian Point 2 and Watts Bar internals, the conversion of the Watts Bar internals to the upflow configuration has no direct impact on the reactor core system under earthquake conditions. Therefore, the fuel assembly structural integrity during a seismic event is not affected by the modification. The potential effects due to the LOCA contribution, as a result of the upflow modification, has been demonstrated by evaluation that the impact of the change in forces from the initial down flow design to upflow are insignificant. Therefore, the modifications associated with the upflow conversion do not increase the seismic or LOCA induced loads significantly compared to that of the downflow design, and the fuel assembly structural integrity and coolable geometry are maintained. This issue has been reviewed and approved by the NRC.[11 & 12]

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the Watts Bar reactor internals design configuration is well characterized, as was discussed in Section 3.9.2.3, it is not considered necessary to conduct instrumented tests of the Watts Bar plant hardware. The requirements of Regulatory Guide 1.20 will be met by conducting the confirmatory preoperational testing examination for integrity per Paragraph D, of Regulatory Guide 1.20, "Regulation for Reactor Internals Similar to the Prototype Design." This examination will include some 34 points for Unit 1 (Figure 3.9-1a and 29 points for Unit 2 (Figure 3.9-1b) with special emphasis on the following areas.

1. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

2. The lateral, vertical and torsional restraints provided within the vessel.

3. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

4. Those other locations on the reactor internal components which are similar to those which were examined on the prototype Indian Point Unit 2 design.

5. The inside of the vessel will be inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material are in evidence.

A particularly close inspection will be made on the following items or areas using a 5X or 1OX magnifying glass or penetrant testing where applicable.

1. Lower Internals

   a. Upper barrel to flange girth weld.

   b. Upper barrel to lower barrel girth weld.

   c. Upper core plate aligning pin. Examine bearing surfaces for any shadow marks, burnishing, buffing or scoring. Inspect welds for integrity.

   d. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

   e. Baffle assembly locking devices. Check for lock weld integrity.

   f. Lower barrel to core support girth weld.

   g. Neutron shield panel screw locking devices and dowel pin cover plate welds. Examine the interface surfaces for evidence of tightness and for lock weld integrity.

   h. Radial support key welds.

   i. Insert screw locking devices. Examine soundness of lock welds.

   j. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.

   k. Secondary core support assembly screw locking devices for lock weld integrity.

   l. Lower radial support keys and inserts. Examine for any shadow marks, burnishing, buffing or scoring. Check the integrity of the lock welds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing or shadow marks which would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

m. Gaps at baffle joints. Check for gaps between baffle and top former and at baffle to baffle joints.

2. Upper Internals

a. Thermocouple conduits, clamps and couplings. (Unit 1 Only)

b. Guide tube, support column, orifice plate, and thermocouple assembly locking devices. (Unit 1 Only)

c. Support column (Units 1 and 2) and thermocouple conduit assembly clamp (Unit 1 Only) welds.

d. Upper core plate alignment inserts. Examine for any shadow marks, burnishing, buffing or scoring. Check the locking devices for integrity of lock welds.

e. Thermocouple conduit gusset and clamp welds (where applicable). (Unit 1 Only)

f. Thermocouple conduit end-plugs. Check for tightness. (Unit 1 Only)

g. Guide tube enclosure welds, tube-transition plate welds and card welds.

h. Guide tube, support column, orifice plate, and flow restrictor locking devices. (Unit 2 Only)

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals will be subjected to a total operating time at greater than normal full-flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition there will be some operating time with only one, two and three pumps operating.

When no signs of abnormal wear, no harmful vibrations are detected and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operation.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

3.9.2.5.1 Design Criteria

The basic requirement of any LOCA including the double-ended severance of a reactor coolant pipe, is that sufficient integrity be maintained to permit the safe and orderly shutdown of the reactor. This implies that the core must remain essentially intact and the deformations of the internals must be sufficiently small so that primary loop flow, and particularly, adequate safety injection flow, is not impeded. The ability to insert control rods, to the extent necessary, to provide shutdown following the accident must be maintained. Maximum allowable deflection limitations are established for those regions of the internals that are critical for plant shutdown. The allowable and no loss of function deflection limits under dead weight loads plus the maximum potential earthquake and/or blowdown excitation loads are presented in Table 3.9-5. These limits have been established by correlating experimental and analytical results.

With the acceptance of Leak-Before Break by NRC, References [6][7][8][9][10][11] and [12] of Section 3.6B.1, the dynamic effects of main coolant loop piping are no longer considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of auxiliary lines need to be considered; and consequently the components will experience considerably less loads and deformations than those from the main loop line breaks.

3.9.2.5.2 Internals Evaluation

The horizontal and vertical forces exerted on reactor internals and the core, following a LOCA, are computed by employing the MULTIFLEX 3.0, which is an enhancement and extension of MULTIFLEX 1.0,[14] NRC reviewed and approved computer code developed for the space-time dependent analysis of nuclear power plants. MULTIFLEX 3.0 has been accepted by NRC for several other applications[15], [16], [17], [18] and also has been extensively used for the LOCA analyses of reactor internals components of numerous other 2, 3, and 4 loop nuclear power plants.

3.9.2.5.3 LOCA Forces Analysis

MULTIFLEX[14] is a digital computer program for calculation of pressure, velocity, and force transients in reactor primary coolant systems during the subcooled, transition, and early saturation portion of blowdown caused by a LOCA. During this phase of the accident, large amplitude rarefaction waves are propagated through the system with the velocity of sound causing large differences in local pressures. As local pressures drop below saturation, causing formation of steam, the amplitudes and velocities of these waves drastically decrease. Therefore, the largest forces across the reactor internals due to wave propagation occur during the subcooled portions of the blowdown transient. MULTIFLEX includes mechanical structure models and their interaction with the thermal-hydraulic system.

MULTIFLEX Code

The thermal-hydraulic portion of MULTIFLEX is based on the one dimensional homogeneous flow model which is expressed as a set of mass, momentum, and energy conservation equations. These equations are quasi-linear first order partial-differential equations that are solved by the method of characteristics. The numerical method employed is the explicit scheme; consequently time steps for stable numerical integration are restricted by sonic propagation.

In MULTIFLEX, the structural walls surrounding a hydraulic path may deviate from their neutral positions depending on the force differential on the wall. The wall displacements are represented by those of one-dimensional mass points which are described by the mechanical equations of vibration.

MULTIFLEX is a generalized program for analyzing and evaluating thermal-hydraulic-structure system dynamics. The thermal-hydraulic system is modeled with an equivalent pipe network consisting of one-dimensional hydraulic legs which define the actual system geometry. The actual system parameters of length, area, and volume are represented with the pipe network.

MULTIFLEX computes the pressure response of a system during a decompression transient. The asymmetric pressure field in the down-comer annulus region of a PWR can be calculated. This pressure field is integrated over the core support barrel area to obtain total dynamic load on the core support barrel. The pressure distributions computed by MULTIFLEX can also be used to evaluate the reactor core assembly and other primary coolant loop component support integrity.

MULTIFLEX evaluates the pressure and velocity transients for locations throughout the system. The pressure and velocity transients are made available to the programs LATFORC and FORCE-2 (described in Reference [14], Appendix A and B), which used detailed geometric descriptions to evaluate hydraulic loading on reactor internals.

Horizontal/Lateral Forces - LATFORC

LATFORC, described in Reference [14], Appendix A, calculates the lateral hydraulic loads on the reactor vessel wall, core barrel, and thermal shield, resulting from a postulated LOCA in the primary RCS. A variation of the fluid pressure distribution in the down-comer annulus region during the blowdown transient produces significant asymmetrical loading on the reactor vessel internals. The LATFORC computer code is used in conjunction with MULTIFLEX, which provides the transient pressures, mass velocities, and other thermodynamic properties as a function of time.

Vertical Forces – FORCE-2

FORCE-2, described in Reference [14], Appendix B, determines the vertical hydraulic loads on the reactor vessel internals. Each reactor component for which force calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.
2. Flow stagnation on, and unrecovered orifice losses across, the element.
3. Friction losses along the element.

Input to the code, in addition to the MULTIFLEX pressure and velocity transients, includes the effective area of each element on which acts the force due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

3.9.2.5.4 Structural Response of Reactor Internals During LOCA and Seismic Conditions

Structural Model and Methods of Analysis

The response of reactor vessel internals due to an excitation produced by a complete severance of auxiliary loop piping is analyzed. With the acceptance of Leak-Before-Break (LBB) by NRC, Reference [12] of Section 3.6B.1, the dynamic effects of main coolant loop piping no longer have to be considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of auxiliary lines need to be considered; and consequently the components will experience considerably less loads than those from the main loop line breaks.

Assuming that such a pipe break in cold leg occurs in a very short period of time (1 ms), the rapid drop of pressure at the break produces a disturbance that propagates through the reactor vessel nozzle into the down-comer (vessel and barrel annulus) and excites the reactor vessel and the reactor internals. The characteristics of the hydraulic excitation combined with those of the structures affected present a unique dynamic problem. Because of the inherent gaps that exist at various interfaces of the reactor vessel/reactor internals/fuel, the problem becomes that of nonlinear dynamic analysis of the reactor pressure vessel system. Therefore, nonlinear dynamic analyses (LOCA and Seismic) of the reactor pressure vessel system include the development of LOCA and seismic forcing functions.

Structural Model

Figure 3.9-1 is schematic representation of the reactor pressure vessel system. In this figure, the major components of the system are identified. The reactor pressure vessel system finite element model for the nonlinear time history dynamic analysis consists of three concentric structural sub-models connected by nonlinear impact elements and linear stiffness matrices. The first sub-model, shown in Figure 3.9-2a and 3.9-2b represents the reactor vessel shell and its associated components. The reactor vessel is restrained by four reactor vessel supports (situated beneath alternate nozzles) and by the attached primary coolant piping. Also shown in Figure 3.9-2a is a typical reactor pressure vessel support mechanism.

The second sub-model, shown in Figure 3.9-3a represents the reactor core barrel, lower support plate, tie plates, and the secondary support components for Watts Bar Unit 1. These sub-models are physically located inside the first, and are connected by stiffness matrices at the vessel/internals interfaces. Core barrel to reactor vessel shell impact is represented by nonlinear elements at the core barrel flange, upper support plate flange, core barrel outlet nozzles, and the lower radial restraints.

The third and innermost sub-model, shown in Figure 3.9-3b represents the upper support plate assembly consisting of guide tubes, upper support columns, upper and lower core plates, and the fuel. The fuel assembly simplified structural model incorporated into the reactor pressure vessel system model preserves the dynamic characteristics of the entire core. For each type of fuel design the corresponding simplified fuel assembly model is incorporated into the system model. The third sub-model is connected to the first and second by stiffness matrices and nonlinear elements. Finally, Figure 3.9-3c shows the reactor pressure vessel system model representation.

Analysis Technique

The WECAN Computer Code[10] which is used to determine the response of the reactor vessel

and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of discrete members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformations, the element global matrices and arrays are assembled into global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure.

$$[M]\{U\} + [D]\{U\} + [K]\{U\} = \{F\} \qquad (1)$$

The WECAN Code solves equation of motions (1) using the nonlinear modal superposition theory. Initial computer runs such as dead weight analysis and the vibration (modal) analyses are made to set the initial vertical interface gaps and to calculate eigenvalues and eigenvectors. The modal analysis information is stored on magnetic tapes, and is used in a subsequent computer runs which solves equation of motions. The first time step performs the static solution of equations to determine steady state solution under normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of equations of motions and nodal displacements and impact forces are stored on tape for post-processing.

The fluid-solid interactions in the LOCA analysis are accounted through the hydraulic forcing functions generated by MULTIFLEX Code. Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wave front with low pressure on one side and high pressure on the other.

The LOCA loads applied to the reactor vessel system for the auxiliary line breaks consist of:

(a) reactor internals hydraulic loads (vertical and horizontal, and
(b) reactor coolant loop mechanical loads.

These loads are calculated individually and combined in a time history manner.

Reactor Pressure Vessel Internal Hydraulic Loads

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle. After a postulated cold leg break the depressurization path for waves entering the reactor vessel is through the nozzle which contains the broken pipe and into the region between the core barrel and the reactor vessel (i.e., down-comer region). The initial waves propagate up, around, and down the down-comer annulus, then up through the region circumferentially enclosed by the core barrel, that is, the fuel region.

In the case of cold leg break, the region of the down-comer annulus close to the break depressurizes rapidly but, because of the restricted flow areas and finite wave speed (approximately 3000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and the reactor vessel. As the depressurization wave propagates around the down-comer annulus and up through the core, the core barrel differential pressure reduces and, similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the hot leg, the wave follows a similar depressurization path, passing through the outlet nozzle and directly into the upper internals region depressurizing the core and entering the down-comer annulus from the bottom exit of the core barrel. Thus, after a reactor pressure vessel outlet nozzle break, the down-comer annulus would be depressurized with very little difference in pressure forces across the outside diameter of the core barrel. A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the down-comer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg break of the same size. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code, calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The MULTIFLEX code employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture. The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator system. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into equally spaced segments and the pressures as well as the wall motions are projected onto the plane parallel to the broken nozzle. Horizontally, the barrel is divided into 10 segments; each segment consists of three separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible wall is determined by solving the global equations of motions for the masses representing the forced vibration of an undamped beam.

<u>Reactor Coolant Loop Mechanical Loads</u>

The loop mechanical loads result from the release of normal operating forces present in the pipe prior to the separation as well as transient hydraulic forces in the RCS. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The load existing in the pipe at the postulated break location are calculated and are “released” at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with a ramp time of one millisecond because of the assumed instantaneous break opening time.

In order to obtain the response of reactor pressure vessel system (vessel/internals/fuel), the LOCA horizontal and vertical forces obtained from the LATFORC and FORCE-2 Codes, which were described earlier, together with the loop mechanical loads are applied to the finite element system model shown in Figure 3.9-3c. The transient response of the reactor internals consists of time history nodal displacements and time history impact forces.

3.9.2.5.5 Seismic Analysis

The basic mathematical model for seismic analysis is essentially similar to the LOCA model except for some minor differences. In LOCA model, as mentioned earlier, the fluid-structure interactions are accounted though the MULTIFLEX Code; whereas in the seismic model the fluid-structure interactions are included through the hydrodynamic mass matrices in the down-comer region. Another difference between the LOCA and seismic models is the difference in loop stiffness matrices. The seismic model uses the unbroken loop stiffness matrix, whereas the LOCA model uses the broken loop stiffness matrix. Except for these two differences, the reactor pressure vessel system seismic model is identical to that of LOCA model.

The horizontal fluid-structure or hydroelastic interaction is significant in the cylindrical fluid flow region between the core barrel and the reactor vessel annulus. Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel, thermal shield and the reactor vessel. The mass matrices for the hydroelastic interactions of two concentric cylinders are developed using the work of Reference [20]. The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell, thermal shield, and core barrel. The off-diagonal terms reflect the fact that all the water mass does not participate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight forward, quantitative manner.

The matrices are a function of the properties of two cylinders with the fluid in the cylindrical annulus, specifically, inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the reactor vessel and the core barrel allows inclusion of radii variations along their heights and approximates the effects of beam mode deformation. These mass matrices were inserted between the selected nodes on the core barrel, thermal shield, and the reactor vessel. The seismic evaluations are performed by including the effects of simultaneous application of time history accelerations in three orthogonal directions. The WECAN computer code, which is described earlier, is also used to obtain the response for the reactor pressure vessel system under seismic excitations.

3.9.2.5.6 Results and Acceptance Criteria

The reactor internals behave as a highly nonlinear system during horizontal and vertical oscillations of the LOCA forces. The nonlinearities are due to the coulomb friction at the sliding surfaces and due to gaps between components causing discontinuities in force transmission. The frequency response is consequently a function not only of the exciting frequencies in the system but also of the amplitude. Different break conditions excite different frequencies in the system. This situation can be seen clearly when the response under LOCA forces is compared with the seismic response. Under seismic excitations, the system response is not as nonlinear as LOCA response because various gaps do not close during the seismic excitations.

The results of the nonlinear LOCA and seismic dynamic analyses include the transient displacements and impact loads for various elements of the mathematical model. These displacements and impact loads, and the linear component loads (forces and moments) are then used for detailed component evaluations to assess the structural adequacy of the reactor vessel, reactor internals, and the fuel.

The stresses due to the safe shutdown earthquake (SSE) are combined in the most unfavorable manner with the LOCA stresses in order to obtain the largest principal stresses and deflections. These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For transverse excitation of the core barrel, it is shown that the upper core barrel does not buckle during hot leg break.

The results also show that the guide tubes will deform well within the limits established experimentally to assure control rod insertion. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9.2.5.7 Structural Adequacy of Reactor Internals Components

The reactor internal components of Watts Bar Unit 1 are not ASME Code components. This is due to the fact that Sub-section NG of the ASME Boiler and Pressure Code edition applicable to Watts Bar reactor internals did not include design criteria for the reactor internals since its design preceded Subsection NG of the ASME Code. However, these components were originally designed to meet the intent of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through the Winter 1971. As noted previously, that with the acceptance of LBB by NRC, the dynamic effects of the main reactor coolant loop piping are no longer considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of the auxiliary lines (accumulator line, pressurizer surge line, and RHR lines) are to be considered.

It should be noted that LBB discussed in Section 3.6A.2.1.5 also refers to the elimination of pressurizer surge line break from the design basis of Watts Bar Unit 1. Therefore, LOCA response of Watts Bar Unit 1 was determined for the auxiliary line breaks consisting of accumulator line and an RHR line. Consequently, the components experience considerably less loads and deformations than those from the main loop breaks which were considered in the original design of the reactor internals.

Allowable Deflection and Stability Criteria

The criterion for acceptability with regard to mechanical integrity analyses is that adequate core cooling and core shutdown must be ensured. This implies that the deformation of reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the reactor internals are concerned principally with the maximum allowable deflections and stability of the components.

For faulted conditions, deflections of critical reactor internal components are limited to the values given in Table 3.9-5. In a hypothesized vertical displacement of internals, energy-absorbing devices limit the displacement to 1.25 inches by contacting the vessel bottom head.

Core Barrel Response Under Transverse Excitations

In general, there are two possible modes of dynamic response of the core barrel during LOCA conditions:

a) During a cold leg break the inside pressure of the core barrel is much higher than the outside pressures, thus subjecting the core barrel to outward deflections.

b) During a hot leg break the pressure outside the core barrel is greater than the inside pressure thereby subjecting the core barrel to compressive loading.

Therefore, this condition requires the dynamic stability check of the core barrel during a hot leg break.

(1) To ensure shutdown and cooldown of the core during cold leg blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to unbroken lines. A large outward deflection of the upper barrel in front of the inlet nozzles, accompanied with permanent strains, could close the inlet area and restrict the cooling water coming from the accumulators. Consequently, a permanent barrel deflection in front of the unbroken inlet nozzles larger than a certain limit, called "no loss of function" limit, could impair the efficiency of the ECCS.

(2) During the hot leg break, the rarefaction wave enters through the outlet nozzle into the upper internals region and thus depressurizes the core and then enters the down-comer annulus from the bottom exit of the core barrel. This depressurization of the annulus region subjects the core barrel to external pressures and this condition requires a stability check of the core barrel during hot leg break. Therefore, to ensure rod insertion and to avoid disturbing the control rod cluster guide structure, the barrel should not interfere with the guide tubes.

Table 3.9-5 summarizes the allowable and no loss of function displacement limits of the core barrel for both the cold leg and hot leg breaks postulated in the main line loop piping. With the acceptance of LBB, the reactor internal components such as core barrel will experience much less loads and deformations than those obtained from main loop piping.

Control Rod Cluster Guide Tubes

The deflection limits of the guide tubes which were established from the test data, and for fuel assembly thimbles, cross-section distortion (to avoid interference between the control rood and the guides) are given in Table 3.9-5

Upper Package

The local vertical deformation of the upper core plate, where a guide tube is located, shall be below 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between the plate and the guide tube is 0.100 inch. This limit will prevent the guide tubes from undergoing compression. For a plate local deformation of 0.150 inch, the guide tube will be compressed and deformed transversely to the upper limit previously established. Consequently, the value of 0.150 inch is adopted as the no loss function local deformation with an allowable limit of 0.100 inch.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

The dynamic behavior of reactor components has been studied using experimental data obtained from operating reactors along with results of model tests and static and dynamic tests in the fabricators shops and at plant site. Extensive instrumentation programs to measure vibration of reactor internals (including prototype units of various reactors) have been carried out during preoperational flow tests, and reactor operation.

From scale model tests, information on stresses, displacements, flow distribution, and fluctuating differential pressures is obtained. Studies have been performed to verify the validity and determine the prediction accuracy of models for determining reactor internals vibration due to flow excitation. Similarity laws were satisfied to assure that the model response can be correlated to the real prototype behavior.

Vibration of structural parts during prototype plants preoperational tests is measured using displacement gages, accelerometers, and strain transducers. The signals are recorded with F.M. magnetic tape records. On site and offsite signal analysis is done using both hybrid real time and digital techniques to determine the (approximate) frequency and phase content. In some structural components the spectral content of the signals include nearly discrete frequency or very narrow-band, usually due to excitation by the main coolant pumps and other components that reflect the response of the structure at a natural frequency to broad bands, mechanically and/or flow-induced excitation. Damping factors are also obtained from wave analyses.

It is known from the theory of shells that the normal modes of a cylindrical shell can be expressed as sine and cosine combinations with indices m and n indicating the number of axial half waves and circumferential waves, respectively. The shape of each mode and the corresponding natural frequencies are functions of the numbers m and n. The general expression for the radial displacement of a simply supported shell is:

$$w(x,\psi,t) = \ddot{\Sigma}_{n=0}\ddot{\Sigma}_{m=1}[A_{nm}(t)\ \cos n\psi\ +\ B_{nm}(t)\sin n\psi]\ \sin\ \frac{M\pi x}{L}$$

The shell vibration at a natural frequency depends on the boundary conditions at the ends. The effect of the ends is negligible for long shells or for higher order m modes, and long shells have the lowest frequency for n = 2 (elliptical mode). For short shells, the effects of the ends are more important, and the shell will tend to vibrate in modes corresponding to values of n > 2.

In general, studies of the dynamic behavior of components follow two parallel procedures: 1) obtain frequencies and spring constants analytically, and 2) confirm these values from the results of the tests. Damping coefficients are established experimentally. Once these factors are established, the response can be computed analytically. In parallel, the responses of important reactor structures are measured during preoperational reactor tests and the frequencies and mode shapes of the structures are obtained.

Theoretical and experimental studies have provided information on the added apparent mass of the water, which has the effect of decreasing the natural frequency of the component. For both cases, cross and parallel, the response is obtained after the forcing function and the damping of the system is determined.

Pre- and post-hot functional inspection results, in the case of plants similar to prototypes, serve to confirm predictions that the internals are well behaved. Any gross motion or undue wear would be evident following the application of approximately 10^7 cycles of vibration expected during the test period.

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

3.9.3.1.1 Subsystems and Components Supplied by Westinghouse

Design transients are presented in Section 5.2.1.5.

For ASME Code Class 1 components, systems, and supports, loading conditions are presented in Section 5.2.1.10.1, and stress criteria are provided in Section 5.2.1.10.7. Additional information concerning methods of analysis is presented throughout Section 5.2.1.10. Results of analyses are documented in the stress reports that describe the system or components.

For core support structures, design loading conditions are given in Section 4.2.2.3. Loading conditions are discussed in Section 4.2.2.4.

In general, for reactor internals components and for core support structures, the criteria for acceptability, with regard to mechanical integrity analyses, are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and stability of the parts, in addition to a stress criterion to assure integrity of the components.

For the LOCA plus the SSE condition, deflections of critical internal structures are limited. In a hypothesized downward vertical displacement of the internals, energy absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

The following mechanical functional performance criteria apply:

1. Following the design basis accident, the functional criterion to be met for the reactor internals was that the plant shall be shutdown and cooled in an orderly fashion so that fuel cladding temperature was kept within specified limits. This criterion implies that the deformation of critical components must be kept sufficiently small to allow core cooling.

2. For large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the emergency core cooling system uses borated water to maintain the core in a subcritical state. Therefore, the main requirement was to assure effectiveness of the emergency core cooling system. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

3. The inward upper barrel deflections are controlled to ensure no contacting of the nearest rod cluster control guide tube. The outward upper barrel deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

4. The rod cluster control guide tube deflections are limited to ensure operability of the control rods.

5. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited.

Methods of analysis and testing for core support structures are discussed in Sections 3.9.1.3, 3.9.1.4.1, 3.9.2.3, 3.9.2.5, and 3.9.2.6. Stress limits and deformation criteria are given in Sections 4.2.2.4 and 4.2.2.5.

3.9.3.1.1.1 Plant Conditions and Design Loading Combinations For ASME Code Class 2 and 3 Components Supplied by Westinghouse

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9.3.1.1.2 Design Loading Combinations by Westinghouse

The design loading combinations for ASME Code Class 2 and 3 equipment and supports are given in Table 3.9-1. The design loading combinations are categorized with respect to Normal, Upset, Emergency, and Faulted Conditions.

Stress limits for each of the loading combinations are equipment oriented and are presented in Tables 3.9-2, 3.9-3, 3.9-4, and 3.9-6 for tanks, inactive pumps, valves, and active pumps, respectively. The definition of the stress equations and limits are in accordance with the ASME Code as follows:

a.. For tanks and all other equipment, Section III of the ASME Code, 1971 Edition through Summer 1973 Addenda, and Code Cases 1607-1, 1635-1 and 1636-1 were utilized to establish stress limits for Normal, Upset, Emergency, and Faulted Conditions. In addition, Code Case 1657 was used for WBN Unit 2.

b. Some equipment was provided in accordance with the Code Edition and Addenda in effect on the date of the contract.

For the actual numerical values of the allowables for specific equipment, the ASME Code Edition applicable to the time period of equipment procurement as specified on the procurement documents was used for the qualification.

Active (Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories) pumps and valves are discussed in Section 3.9.3.2. The equipment supports are designed in accordance with the requirements specified in Section 3.9.3.4.

3.9.3.1.1.3 Design Stress Limits By Westinghouse

The design stress limits established for equipment are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are equipment oriented and are presented in Tables 3.9-2 through 3.9-4, and 3.9-6. See Section 3.9.3.1.1.2 for discussion of applicable code editions.

3.9.3.1.2 Subsystems and Components Analyzed or Specified by TVA

A. ASME Code Class 1, 2, and 3 Piping.

The analytic procedures and modeling of piping systems is discussed in Sections 3.7.3.8 and 3.7.3.3.* As discussed in Section 3.7.3.8.1 the TVA analysis effort has been categorized into two approaches: Rigorous and Alternate. The loading sources, conditions, and stress limits are described below for each category and the results are summarized for each.

* Generated reactor coolant loop response spectra curves and movements enveloping the Set B + Set C curves are used for the analysis or reanalysis of auxiliary piping systems attached to the reactor coolant loops. The ASME Code Case N-411 or Regulatory Guide 1.61 damping values can be used when Set B + Set C spectra are considered.

1. Loading Conditions, Stress Limits and Requirements for Rigorous Analysis

a. The loading sources considered in the rigorous analysis of a piping system are defined in Table 3.9-7.

b. The piping was analyzed to the requirements of applicable codes as defined in Section 3.7.3.8.1.

c. The design load combinations are categorized with respect to Normal, Upset, Emergency, and Faulted Conditions. Class 1 piping was analyzed using the limits established in Table 3.9-8 for all applicable loading conditions. The pressurizer surge line was also evaluated for the thermal stratification and thermal striping in response to the NRC Bulletin 88-11. Other rigorously analyzed piping meets the limits established in Table 3.9-9 for all applicable loading conditions.

d. Consideration was given to the sequence of events in establishing which load sources are taken as acting concurrently.

e. Equipment nozzle loads are within vendor and/or TVA allowable values. This ensures that functionality and 'Active' equipment operability requirements are met.

f. All equipment (i.e., valves, pumps, bellows, flanges, strainers, etc.) was checked to ensure compliance with vendor limitations.

g. The pipe/valve interface at each active valve was evaluated and the pipe stresses are limited to the levels indicated in Table 3.9-10 unless higher limits are technically justified on a case-by-case basis.

h. Documentation of rupture stress was provided for the locations in the system being analyzed where the stress exceeds the limits for which pipe rupture postulation was required (See Section 3.6). The tabulation identifies the point and tabulates the stress for each point exceeding the limits.

i. Valves with extended operators or structures (including handwheels) meet the dynamic plus gravitational acceleration limits of 3g along the stem axis and 3g (vectorial summation) in the plane perpendicular to the valve stem axis. For 1-inch and smaller valves with handwheels, the dynamic plus gravitational acceleration limit is 3g in each of the three global (or local) directions. These limits apply to any valve orientation and must be maintained during piping analysis.

   For steel body check valves (which have no external operators or structures) the limit for dynamic plus gravitational acceleration was 10g (vectorial summation of all three orthogonal directions).

   The valves as a minimum are qualified to the acceleration limits specified above. Higher accelerations are approved based on case-by-case technical justification.

j. Excessive pipe deformation was avoided.

k. Welded attachment loads and stresses for TVA Class 1 piping were evaluated in accordance with ASME Code Cases N-122 and N-391.

For Class 2 and 3 piping, loads and stresses from welded attachments were evaluated in accordance with ASME Code Cases N-318 and N-392.

Special cases of other welded attachments were evaluated by detailed finite element analysis or other applicable methods to assure that ASME Code stress allowables were met.

The attachment welds are full penetration, partial penetration, or fillet welded as detailed on the support drawings. Attachments are used generally on all piping systems, and locations can be determined from the support drawings.

2. <u>Loading Conditions and Stress Limits for Alternate Analysis</u>

   a. The scope of the alternate analysis application is generally limited to systems having the following load sources: self-weight, internal pressure, seismic event, end point displacement, and limited thermal expansion. (Other load sources may be considered for special cases.)

b. The design load combinations are categorized with respect to Normal, Upset, Emergency, and Faulted Conditions. The criteria are developed to meet the stress limits given in Table 3.9-9 considering the applicable load sources.

c. The general limitations imposed on the piping by the application of the Alternate Analysis method are discussed in Section 3.7.3.8.3. For ASME Category I piping designed by alternate analysis, the same levels of valve acceleration and interface/nozzle load requirements of Section 3.9.3.1.2.A shall be maintained. Non-ASME, Category I(L) piping designed by alternate analysis is described in Sections 3.7.3.8.3 and 3.2.1.

3. <u>Considerations for the Faulted Condition</u>

Tables 3.9-8 through 3.9-10 identify the load sources and allowed stresses associated with the faulted condition. The stress limits used are those limits established in ASME Section III for the faulted condition.

The feedwater system inside containment, from the check valves to the steam generators including the piping components are evaluated for pressure boundary integrity to withstand the postulated water hammer event due to the feedwater check valve slam following pipe rupture at the main header (Turbine Building) using the ASME Section III Appendix F (1980 Edition through Winter 1982 Addenda) rules and limits.

The four main feedwater check valves were evaluated for structural integrity following the feedwater pipe rupture. Energy equivalence methods, in conjunction with nonlinear finite element and linear hand analyses, were used. The evaluations demonstrated that deformations in three of the four valves are within acceptable strain levels following the slam. With the assumption that the fourth valve is not functional, the transient effects of the resulting one steam generator blowdown are bounded by the "Major Rupture of a Main Feedwater Pipe inside containment" per Section 15.4.2.2.

Note that during the rigorous analysis phase of most piping systems, the postulated break locations are unknown and the jet impingement loads are unavailable and thus not included in the evaluation of the faulted condition. However, where it was determined by the guidelines of Section 3.6 that jet impingement must be evaluated, the effect of the loads on pipe stress was evaluated during the pipe rupture analysis.

4. Summary of Results - Rigorous Analysis of Class 1 and Class 2/3 Piping Performed by TVA

The results of the piping system analyses performed in accordance with the above paragraphs are presented and consolidated in calculations with the following documentation:

a. Certification Report for ASME Code Class 1 Analyses.

b. Owner's review for ASME Code Class 1 Analyses.

c. Statement of Compliance with code requirements for ASME Code Class 2/3 Analyses.

d. Problem revision status form - for maintaining the traceability of revision performed on analysis, and correlating various forms affected by each revision.

e. Piping input data for recording all physical data used in the analysis.

f. Table of system operating modes - for identifying the various thermal conditions required and included in analysis.

g. Stress summary - for summarizing the maximum stresses for various loading combinations.

h. Equipment nozzle load qualification to demonstrate satisfaction of limits.

i. Valve acceleration qualification to demonstrate satisfaction of limits.

j. Summary of loads and movements at pipe supports.

A record copy of these problem calculations is maintained at TVA and is available for review upon request.

B. Category I ASME Code Class 2 and 3 Mechanical Equipment

1. Plant Conditions and Design Loading Combinations

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

2. Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 equipment and supports are given in Table 3.9-13B. The design loading combinations are categorized with respect to Normal, Upset, Emergency and Faulted Conditions.

Stress limits for each of the loading combinations are equipment oriented and are presented in Tables 3.9-14, 3.9-15 and 3.9-16 for tanks, inactive* pumps, and inactive* valves respectively. The definition of the stress equations and limits are in accordance with the ASME Code as follows:

* Inactive components are those whose operability are not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

a. For tanks and all other equipment, Section III of the ASME Code, 1971 Edition through Summer 1973 Addenda, and Code Cases 1607-1, 1635-1 and 1636-1 were utilized to establish stress limits for Normal, Upset, Emergency, and Faulted Conditions. In addition, Code Case 1657 was used for WBN Unit 2.

b. Some equipment was provided in accordance with the Code Edition and Addenda in effect on the date of the award of contract.

   For the actual numerical values of the allowables for specific equipment, the ASME Code Edition applicable to the time period of equipment procurement as specified on the procurement documents was used for the qualification.

Active pumps and valves are discussed in Section 3.9.3.2.1. The vendor supplied equipment/component supports stress levels are limited to the allowable stress of AISC or ASME Section III subsection NF or other comparable stress limits as delineated in the applicable design specification. Section 3.8.4 describes the allowable stresses used for TVA-designed equipment/component supports.

The design stress limits established for the components are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are component oriented and are presented in Tables 3.9-14 through 3.9-16.

3.9.3.2 Pumps and Valve Operability Assurance

3.9.3.2.1 Active ASME Class 1, 2, & 3 Pumps and Valves

Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

The list of active valves for primary fluid (i.e., water and steam containing components) systems in the Westinghouse scope of supply is presented in Table 3.9-17. The list of active pumps supplied by Westinghouse is presented in Table 3.9-28. The list of pumps and valves for fluid systems within TVA scope of supply are presented in Tables 3.9-25 and 3.9-27. Only ASME Section III pumps and valves that were purchased after September 1, 1974, were considered to be within the scope of WBN compliance with Regulatory Guide 1.48. These pumps and valves meet the special design requirements verifying operability as specified in Regulatory Guide 1.48. The remaining components in Tables 3.9-17, 3.9-25, 3.9-27, and 3.9-28 meet the appropriate qualification requirements in accordance with the guidelines of IEEE 344-1971 and consistent with the ASME Code applicable at the time of the contract date for procuring the component. These qualifications provide an adequate level of operability assurance for all active pumps and valves.

The following rules were used to identify active pumps and valves:

1. Only UFSAR Chapter 15 Design Basis Events (DBE's) were assumed. These DBE's were studied to identify the active pumps and valves required to mitigate the DBE and place the plant in a safe shutdown condition.

2. Reactor Coolant Pressure Boundary (RCPB) - Valves that are a part of the RCPB (defined by 10 CFR Section 50.2) and require movement to isolate the RCS were identified as active.

3. Containment Isolation - Containment isolation valves that require movement to isolate the containment were identified as active.

4. Check Valves - Any check valve required to close or cycle when performing its system safety function was identified as active.

   Any check valve that was only required to open in the performance of its system safety function was not identified as active. This position was justified by: (a) the free-swinging nature of the valves and (b) the normal stress over-design of the valve body.

5. Achieve and Maintain a Safe Shutdown Condition - The minimum redundant complement of equipment required to achieve and maintain safe shutdown was selected.

3.9.3.2.2 Operability Assurance

3.9.3.2.2.1 Westinghouse Scope of Supply

Mechanical equipment classified as safety-related must be shown capable of performing its function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include 'active' pumps and valves in fluid systems such as the residual heat removal system, safety injection system, and the containment spray system. Seismic analysis is presented in Section 3.7 and covers all safety-related mechanical equipment.

Operability is assured by satisfying the requirements of the programs specified below. Additionally, equipment specifications include requirements for operability under the specified plant conditions and define appropriate acceptance criteria to ensure that the program requirements defined below are satisfied.

Pump and Valve Qualification for Operability Program

Active pumps are qualified for operability by first, being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include: 1) hydrostatic tests of pressure retaining parts to 150 percent times the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, 2) seal leakage tests, and 3) performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements and other pump parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer, based on the bearing material, clearances, oil type, and rotational speed. These limits are approved by Westinghouse. Vibration limits are also determined by the manufacturer and are approved by Westinghouse. After the pump is installed in the plant, it undergoes the cold hydro-tests, hot functional test, and the required periodic inservice inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant. In addition to these tests, the safety-related active pumps, are qualified for operability by assuring that the pump will start up, continue operating, and not be damaged during the faulted condition.

The pump manufacturer was required to show by analysis correlated by test, prototype tests or existing documented data that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum faulted nozzle loads. It was required that test or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor was performed with the conservative SSE accelerations of 3g horizontal and 2g vertical acting simultaneously. The deflections determined from the static shaft analysis were compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact will be of short duration. If rubbing or impact is predicted, it is required that it be shown by prototype tests or existing documented data that the pump will not be damaged or cease to perform its design function. The effect of impacting on the operation of the pump was evaluated by analysis or by comparison of the impacting surfaces of the pump to similar surfaces of pumps which had been tested.

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, dynamic system loads are limited to the limits indicated in Table 3.9-6. In addition, the pump casing stresses caused by the maximum faulted nozzle loads are limited to the stresses outlined in Table 3.9-6.

The changes in operating rotor clearances caused by casing distortions due to these nozzle loads were considered. The maximum seismic nozzle loads combined with the loads imposed by the seismic accelerations were considered in an analysis of the pump supports. Furthermore, the calculated misalignment was shown to be less than that misalignment which could cause pump misoperation. The stresses in the supports are below those in Table 3.9-6; therefore, the support distortion is elastic and of short duration (equal to the duration of the seismic event).

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-6 as allowables, assure that critical parts of the pump are not damaged during the short duration of the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation is not impaired by the seismic event.

If the natural frequency was found to be below 33 Hz, an analysis was performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations were determined using the same conservatisms contained in the 3g horizontal and 2g vertical accelerations used for 'rigid' structures. The static analysis was performed using the adjusted accelerations; the stress limits stated in Table 3.9-6 were satisfied.

To complete the seismic qualification procedures, the pump motor was qualified for operation during the maximum seismic event. Any auxiliary equipment identified to be vital to the operation of the pump or pump motor, and which is not proven adequate for operation by the pump or motor qualifications, was separately qualified by meeting the requirements of IEEE Standard 344-1971 or -1975, as applicable, with the additional requirements and justifications outlined in this section.

The program above gives the required assurance that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings, and, therefore, will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured since only normal operating loads and steady state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Safety-related active valves must perform their mechanical motion at times of an accident. Assurance is supplied that these valves will operate during a seismic event. Tests and analyses were conducted to qualify active valves.

The safety-related active valves were subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests were performed: shell hydrostatic test to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valve will open and close.

For the active valves qualification of electric motor operators for the environmental conditions (i.e., aging, radiation, accident environment simulation, etc.) refer to Section 3.11 and Regulatory Guide 1.73. Cold hydro tests, hot functional qualifications tests, periodic inservice inspections, and periodic inservice operations are performed in-situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

On all active valves, an analysis of the extended structure was performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The stress limits allowed in these analyses show structural integrity. The limits used for active Class 2 and 3 valves are shown in Table 3.9-4.

In addition to these tests and analyses, a representative electric motor operated valve was tested for verification of operability during a simulated plant faulted condition event by demonstrating operational capabilities within the specified limits. The testing procedures are described below.

The valve was mounted in a manner which represents typical valve installations. The valve included operator and limit switches if such are normally attached to the valve in service. The faulted condition nozzle loads were considered in the test in either of two ways: 1) loads equivalent to the faulted condition nozzle loads were limited such that the operability of the valve was not affected.

The operability of the valve during a faulted condition was demonstrated by satisfying the following criteria:

1. All the active valves were designed to have a first natural frequency which is greater than 33 Hz, if it was practical to do so. If the lowest natural frequency of an active valve was less than 33 Hz, then the valve's mathematical model was included in the piping dynamic analysis, so as to assure the calculated valve acceleration does not exceed the values used in the static tests of the manufacturer's qualification program and to reflect the proper valve dynamic behavior.

2. The actuator and yoke of the representative motor operated valve system was statically deflected using an equivalent static load that simulates those conditions applied to the valve under faulted condition accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve was simultaneously applied to the valve during the static deflection tests.

3. The valve was cycled while in the deflected position. The time required to open or close the valve in the defected position was compared to similar data taken in the undeflected condition to evaluate the significance of any change.

The accelerations used for the static valve qualification are 3g horizontal and 2g vertical with the valve yoke axis vertical. The piping designer maintained the operator accelerations to these levels unless higher limits were technically justified on a case-by-case basis.

The testing was conducted on the valves with extended structures which are most affected by acceleration, according to mass, length and cross-section of extended structures. Valve sizes which cover the range of sizes in service were qualified by the tests and the results are used to qualify all valves within the intermediate range of sizes.

Valves which are safety-related but can be classified as not having an extended structure, such as check valves and safety valves, were considered separately. Check valves are characteristically simple in design and their operation will not be affected by seismic accelerations or the applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to seismic excitation will not affect the functional ability of the valve since the valve disc is typically designed to be isolated from the body wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valve was subjected to the following tests and analysis: 1) in-shop hydrostatic test, 2) in-shop seat leakage test, and 3) periodic in-situ valve exercising and inspection to assure the functional ability of the valve.

The pressurizer safety valves were qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): stress and deformation analyses of critical items which may affect operability for faulted condition loads, in-shop hydrostatic and seat leakage tests, and periodic in-situ valve inspection. In addition to these tests, a static load equivalent to that applied by the faulted condition was applied at the top of the bonnet and the pressure will be increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assured its overpressurization safety capabilities during a seismic event.

Using these methods, active valves were qualified for operability during a faulted event. These methods conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary. The above testing program for valves is conservative. Alternate valve operability testing, such as dynamic vibration testing, was allowed if it was shown to adequately assure the faulted condition functional ability of the valve system.

Pump Motor and Valve Operator Qualification

Active pump motors (and vital pump appurtenances) and active valve electric motor operators (and limit switches and pilot solenoid valves), were seismically qualified by meeting the requirements of IEEE Standard 344-1971 or 1975, as applicable. If the testing option was chosen, sine-beat testing was justified. This justification was provided by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

1. The equipment response is basically due to one mode.

2. The sine-beat response spectra envelopes the floor response spectra in the region of significant response.

3. The floor response spectra consists of one dominant mode and has a peak at this frequency.

If the degree of coupling in the equipment is small, then single axis testing may have been justified. Multi-axis testing was required if there is considerable cross coupling; however, if the degree of coupling can be determined, then single axis testing will be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

Seismic qualification by analysis alone, or by a combination of analysis and testing, has been used when justified. The analysis program can be justified by: 1) demonstrating that equipment being qualified is amendable to analysis, and 2) that the analysis be correlated with test or be performed using standard analysis techniques.

3.9.3.2.2.2 TVA Scope of Supply

TVA used the following criteria to prescribe a suitable program to assure the functional adequacy of active Category I fluid system components (pumps and valves) under combined loading conditions. These criteria supplement or amend previously stated requirements for fluid system components in those cases where fluid system components are judged to be active (i.e., if they perform a required mechanical motion during the course of accomplishing a safety function). These criteria assure that all active seismic Category I fluid system components will maintain structural integrity and perform their safety functions under loadings, including seismic, associated with normal, upset, and faulted conditions. These criteria are similar to the accepted response to NRC Position for the TVA's Bellefonte Nuclear Plant units 1 and 2 concerning compliance with the requirements of Regulatory Guide 1.48. The exception is that the seismic qualification for Watts Bar is for a 2-dimensional earthquake, while for Bellefonte it is for a 3-dimensional earthquake.

3.9.3.2.3 Criteria For Assuring Functional Adequacy of Active Seismic Category I Fluid System Components (Pumps and Valves) and Associated Essential Auxiliary Equipment

1. The seismic design adequacy of Category I electrical power and control equipment and instrumentation directly associated with the active Category I pumps and valves is assured by seismically qualifying the components by analysis and/or testing in accordance with the requirements of IEEE Standard 344 (for applicable edition, refer to Section 3.9.2.2).

2. When either analysis or testing is used to demonstrate the seismic design adequacy of Category I components, the characteristics of the required input motion is specified by either response spectra, power spectral density function or time history data derived from the structure or system seismic analysis. When the testing method is used, random vibration input motion shall be used, but single frequency input, such as sine beats, may be used provided that:

    a. The characteristics of the required input motion indicate that the motion is dominated by one frequency.

    b. The anticipated response of equipment is adequately represented by one mode.

    c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelop the corresponding response spectra of the individual modes.

For equipment with more than one dominant frequency and for equipment supported near the base of the structure where some random components of the earthquake may remain, single frequency testing may still be applicable provided that the input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelop the corresponding response spectra of the individual modes. When equipment responses along one direction are sensitive to the vibration frequencies along another perpendicular direction, in the case of single frequency testing, the time phasing of the inputs in the vertical and horizontal directions is such that a purely rectilinear resultant output is avoided.

In both the testing and analysis procedure, the possible amplified design loads for vendor supplied equipment is considered as follows:

a. If supports are tested, they were tested with the actual components mounted and operating or if the components are inoperative during the support test, the response at the equipment mounting locations were monitored and components were tested separately and the actual input to the equipment was more conservative in amplitude and frequency content than the monitored responses.

b. The support analysis includes the component loads. Seismic restraints were used as applicable with their adequacy verified by either testing or analysis as described above.

3. Active Category I pumps were subjected to tests both prior to installation in the plant and after installation in the plant. The in-shop tests include (a) hydrostatic tests of pressure-retaining parts, (b) seal leakage tests, and (c) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements and other pump/motor parameters. Bearing temperatures and vibration levels were monitored during these operating tests. Both were shown to be below appropriate limits specified to the manufacturer for design of the pump. After the pump was installed in the plant, it was subject to cold hydro tests, or operational tests, hot functional tests, and the required periodic in-service inspection and operation.

4. Active Category I pumps were analyzed to show that the pump will operate normally when subjected to the maximum seismic accelerations and maximum seismic nozzle loads. Tests or dynamic analysis show that the lowest natural frequency of the pump is above 33 Hz, and thus considered essentially rigid. A static shaft deflection analysis of the rotor was performed with the conservative seismic accelerations of 1.5 times the applicable plant floor response spectra. The deflections determined from the static shaft analysis were compared to the allowable rotor clearances. Stresses caused by the combination of normal operating loads, seismic, and dynamic system loads were limited to the material elastic limit, as indicated in Table 3.9-18. The primary membrane stress (P_m) for the faulted condition loads were limited to $1.2S_h$, or approximately 0.75 S (S = yield stress). The primary membrane stress plus the primary bending stress (P_b) was limited to $1.8S_h$, or approximately 1.l S. In addition, the pump nozzle stresses caused by the maximum seismic nozzle loads were limited to stresses outlined in Table 3.9-18. The maximum seismic nozzle loads were considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

   If the natural frequency is found to be below 33 Hz, then analyses are performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations were determined using the same conservatism contained in the accelerations used for "rigid" structures. The static analysis was performed using the adjusted accelerations; the stress limits stated in Table 3.9-18 were satisfied.

5. Each type of active Category I pump motor is independently qualified for operating during the maximum seismic event. Any appurtenances which are identified to be vital to the operation of the pump or pump motor and which are not qualified for operation during the pump analysis or motor qualifications, shall also be separately qualified for operation at the accelerations each would see at its mounting. The pump motor and vital appurtenances are qualified by meeting the requirements of IEEE Standard 344-1971 or 1975 edition, depending on the procurement date (see Section 3.9.2.2.). If the testing option was chosen, sine-beat testing was justified by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

   a. The equipment response is basically due to one mode.

   b. The sine-beat response spectra envelops the floor response spectra in the region of significant response.

   c. The floor response spectra consist of one dominant mode and have a peak at this frequency. The degree of mass or stiffness coupling in the equipment will, in general, determine if a single or multi-axis test is required. Multi-axis testing was required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified; or, if the degree of coupling can be determined, then single axis testing was used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

6. The post-faulted condition operating loads for active Category I pumps was considered identical to the normal plant operating loads. This was assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions be limited by the magnitudes of the normal condition nozzle loads. Thus, the post-faulted condition ability of the pumps to function under these applied loads was proven during the normal operating plant conditions.

7. Active Category I valves, except check valves, were subjected to a series of stringent tests prior to installation and after installation in the plant. Prior to installation, the following tests were performed: (a) shell hydrostatic test, (b) backseat and main seat leakage tests, (c) disc hydrostatic test, (d) functional tests to verify that the valve will operate within the specified time limits when subjected to the design differential pressure prior to operating, and (e) operability qualification of motor and air operator control valves for the conditions over their installed life (i.e., aging, radiation, accident environment simulation, etc.) in accordance with the requirements of IEEE Standard 382 (see Section 3.11). Cold hydro qualification tests, preoperational tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation were performed after installation to verify and assure functional ability of the valves. To the extent practicable, functional tests are performed after installation to verify that the valve will open and/or close in a time consistent with required safety functions.

8. Active Category I valves are designed using either stress analyses or the pressure containing minimum wall thickness requirements. An analysis of the extended structure is performed for static equivalent seismic loads applied at the center of gravity of the extended structure. The maximum stress limits allowed in these analyses confirms structural integrity and were the limits developed and accepted by the ASME for the particular ASME class of valve analyzed. The stress limits used for active Class 2 and 3 valves are given in Table 3.9-19. Class 1 valves were designed according to the rules of the ASME Boiler and Pressure Vessel Code, Section III, NB-3500.

9. Representative active Category I valves of each design type with overhanging structures (i.e., motor or pneumatic operator) were tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within specified limits. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types (e.g., motor- operated gate valves) were tested. Valve sizes which cover the range of sizes in service were qualified by the tests and the results were used to qualify all valves within the intermediate range of sizes. Stress and deformation analyses are used to support the interpolation.

   The valve was mounted in a manner which is conservatively representative of a typical plant installation. The valve includes the operator and all appurtenances normally attached to the valve in service. The operability of the valve during a seismic event was verified by satisfying the following requirements:

   a. All active valves are designed to have a first natural frequency which is greater than 33 Hz if practical to do so. This may be shown by suitable test or analysis.

      If the lowest natural frequency of an active valve is less than 33 Hz, the valve's mathematical model is included in the piping dynamic analysis. This assures the calculated valve acceleration does not exceed the values used in the static tests of the manufacturers qualification program and reflects the proper valve dynamic behavior.

   b. The actuator and yoke of the valve system were statically loaded an amount greater than that determined by an analysis as representing the applicable seismic accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

   c. The valve was operated while in the deflected position, i.e., from the normal operating mode to the faulted operating mode. The valve performed its safety-related function within the specified operating time limits.

   d. Motor operators and other electrical appurtenances necessary for operation were qualified as operable during the seismic event by analysis and/or testing in accordance with the requirements of IEEE Standard 344 (refer to Section 3.9.2.2 for the applicable edition).

The accelerations used for the static valve qualification are 3.0 g horizontal and 2.0 g vertical with the valve yoke axis vertical. The piping designer shall maintain the motor operator accelerations to equivalent levels. If the valve accelerations exceed these levels, an evaluation of the valve is performed to document acceptability on a case-by-case basis.

If the frequency of the valve, by test or analysis, was less than 33 Hz, a dynamic analysis of the valve was performed to determine the equivalent acceleration to be applied during the static test. The analysis provided the amplification of the input acceleration considering the natural frequency of the valve and piping along with the frequency content of the applicable plant floor response spectra. The adjusted accelerations were determined using the same conservatism contained in the accelerations used for "rigid" valves. The adjusted accelerations were then used in the static analysis and the valve operability was assured by the methods outlined in steps (b), (c), and (d) above using the modified acceleration input.

10. The design of each active Category I check valve was such that once the structural integrity of the valve was assured using standard design or analysis methods, the ability of the valve to operate was assured by the design features. In addition to design considerations, each active check valve undergoes:

    a. Stress analysis including the applicable seismic loads

    b. In-shop hydrostatic tests for parts that could affect the operability of the valve,

    c. In-shop seat leakage tests, and

    d. Preoperational and periodic in-situ testing and inspection to assure functional ability of the valves.

11. The design of the pressurizer safety valve (Category I) was such that once the structural integrity of the valves was assured using standard design or analysis methods, the ability of the valve to operate was assured by the design features. In addition to design considerations, the pressurizer safety valve was subjected to:

    a. Stress and/or deformation analyses for parts that could affect the operability of the valve for the applicable seismic loads,

    b. In-shop hydrostatic and seal leakage tests, and

    c. Periodic in-situ valve inspection.

    In addition to these tests, a static load equivalent to the seismic load was applied to the top of the bonnet and the pressure was increased until the valve mechanism actuated. Successful actuation within the design requirements of the valve has been demonstrated.

12. Wherever practicable, prototype test and analytical results are utilized to assure functional adequacy of active Category I pumps and valves and their appurtenances.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The design and installation of pressure relieving devices are consistent with the requirements established by Regulatory Guide 1.67, "Installation of Overpressure Protective Devices."

Each main steam line is provided with one power operated atmospheric relief valve and five safety valves sized in accordance with ASME, B&PV, Section III.

The safety valves are set for progressive relief in intermediate steps of pressure within the allowed range (100% to 105% of the design pressure) of pressure settings to prevent more than one valve actuating simultaneously. The valve pressure settings at which the individual valves open are tabulated in Table 10.1-1 in the column identified as "Set Pressure." The valves are designed to reseat at the pressure values identified in the column "Blowdown Pressure."

Valves are connected to a rigidly supported common header that is in turn connected to the main steam piping through branch piping equal in size to the main steam piping. The header and valves are located immediately outside containment in the main steam valve building.

The safety valves are mounted on the header such that the valves produce torsion, bending, and thrust loads in the header during valve operation. The header has been designed to accommodate both dynamic and static loading effects of all valves blowing down simultaneously.

The stress produced by the following loading effects assumed to act concurrently are within the code allowable.

1. Deadweight effects
2. Thermal loads and movements
3. Seismic loads and movements
4. Safety valve thrust, moments, torque loading[1]
5. Internal pressure

Note 1. The safety valve thrust loads are assumed to occur in the upset plant condition, and do not occur concurrently with an OBE.

The nozzles connecting each valve to the header are analyzed to assure that for both dynamic and static loading situations, the stresses produced in the nozzle wall are within the code allowable for the same loading consideration as the header.

The safety valves and power-operated atmospheric relief valves are Seismic Category I components. They have been seismically qualified by analyses per criteria presented in Section 3.7.3.16 and Table 3.9-19.

Pressure relief valves in auxiliary safety related systems have been installed considering loads carried in the support members produced by:

1. Deadweight of valve and appurtenances,

2. Thermal effects,

3. Seismic effects,

4. Maximum valve thrust, moment, and torque loading effects, and

5. Internal pressure.

Relief valves that discharge to the atmosphere are either rigidly supported by their own individual support, or the nozzle and component to which the valve is attached (vessel, tank, or pipe) has been designed to carry the valve static and dynamic loads. Individual supports have been designed to stress levels in accordance with Section 3.9.3.4.2. Stresses in nozzles and components produced by the valve loads considered above are determined per the method delineated in Welding Research Council Bulletin No. 107 or equivalent and are combined with normal loading operational loads for the component. Relief valves blowing down are considered as an upset loading condition for the plant. Therefore, the allowable stress intensity for the component supporting the valve loads is in accordance with those tabulated in Tables 3.9-2, 3.9-8, 3.9-9, or 3.9-14, as applicable.

Loading associated with relief valves discharging through piping components to a collector tank are analyzed considering the surge effects of the initial discharge through the pipe. This condition is considered as an upset loading condition for the piping components connecting to the valve and the allowable stress intensity is in accordance with those for piping components tabulated in Tables 3.9-8 and 3.9-9.

Pressure relief valves and pertinent operating information for the valves that have been considered in the installation requirements of the valve are tabulated in Table 3.9-20.

As related to the design and installation criteria of pressure relieving devices, Westinghouse interfaces with TVA by providing the following:

1. Overpressure Protection Report[13]. This report documents the compliance for overpressure protection requirements as per the ASME Boiler and Pressure Vessel Code, Section III, NB-7300 and NC-7300, and provides the maximum relieving requirements.

2. Mounting brackets on the pressurizer. These brackets can be used as structural supports, if needed.

3. Criteria and guidelines. Supplementary criteria specifically applicable to the design and fabrication of nuclear plant safety and relief valve installations are provided.

4. Review of system layout and resultant loads for acceptability, where applicable.

3.9.3.4 Component Supports

3.9.3.4.1 Subsystem and Component Supports Analyzed or Specified by Westinghouse

1&2. The criteria for Westinghouse supplied supports for ASME Code Class 1 Mechanical Equipment is presented in Section 5.2.1.10.7.

3. ASME Code - Class 2 and 3 supports are designed as follows:

   A. Linear

      a. Normal - The allowable stresses of AISC-69 Part 1, a reference basis for Subsection NF of ASME Section III, are employed for normal condition limits.

      b. Upset - Stress limits for upset conditions are the same as normal condition stress limits. This is consistent with Subsection NF of ASME Section III (see NF-3320).

c. Emergency - For emergency conditions, the allowable stresses or load ratings are 33% higher than those specified for normal conditions. This is consistent with Subsection NF of ASME Section III in which (see NF-3231) limits for emergency conditions are 33% greater than the normal condition limits.

d. Faulted - Section 5.2.1.10 specifies limits which assure that no large plastic deformations will occur (stress $\leq 1.2S_y$). If any inelastic behavior is considered in the design, detailed justification is provided for this limit. Otherwise, the supports for active components are designed so that stresses are less than or equal to S_y. Thus the operability of active components is not endangered by the supports during faulted conditions.

Welding was in accordance with the American Welding Society, (AWS) "Structural Welding Code," AWS D1.1, with revisions 1-73 and 1-74, except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders. Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS. Visual inspection of structural welds will meet the minimum requirements of NCIG-01 and NCIG-02 as specified on the design drawings or other design output. Inspectors performing visual examination to the criteria of NCIG-01 are trained in the subject criteria.

B. Plates and Shells

The stress limits used for ASME Class 2 and 3 plate and shell component supports are identical to those used for the supported component. These allowed stresses are such that the design requirements for the components and the system structural integrity are maintained.

For active Class 2 or 3 pumps, support adequacy was proven by satisfying the criteria in Section 3.9.3.2.1. The requirements consist of both stress analysis and an evaluation of pump/motor support misalignment.

Active valves are, in general, supported only by the pipe attached to the body. Exterior supports on the valve are not used.

3.9.3.4.2 Subsystem and Component Supports Analyzed or Specified by TVA

1. ASME Code Class 1, 2, and 3 Piping Supports

a. Loading Conditions

The following conditions have been assigned for support load evaluation for Watts Bar Nuclear Plant support design (not including pipe whip restraints): normal, upset, emergency, faulted, and test condition. The piping support design loads and combinations are given in Table 3.9-13A.

b. Support Types, Loading Combinations, Stress Limits, and Applicable Codes

1) Linear Supports

The allowed stresses are defined in Table 3.9-21. The load combinations and allowable stresses are based on and exceed the requirements of NRC Regulatory Standard Review Plan, Section 3.9.3. The design load is determined by the condition yielding the most conservative support design.

Welding was in accordance with the American Welding Society, (AWS) "Structural Welding Code," AWS D1.1 with revisions 1-73 and 1-74, except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders. Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS. Visual inspection of structural welds will meet the minimum requirements of NCIG-01 and NCIG-02 as specified on the design drawings or other design output. Inspectors performing visual examination to the criteria of NCIG-01 are trained in the subject criteria.

2) Standard Support

The allowable stresses are defined in Table 3.9-21. The load combinations consider all applicable load sources which induce load into the appropriate type support. The design conforms to the requirements of MSS-SP-58, 1967 edition or ASME Boiler and Pressure Vessel Code, Section III, subsection NF.

3) Pre-engineered Support Element

Pre-engineered support elements are defined as standard hardware items such as rods, clamps, clevises, and struts used in the installation of either a linear support or a standard support component.

The design load is determined from the tabulated loads described above for the linear or standard support component. The allowable stresses are given in Table 3.9-21.

c. General Design Requirements

1) The gravitational or actual loads are considered to consist of pipe, fittings, pipe covering, contents of pipe systems, and valves.

2) All thermal modes of operation are considered in load evaluation. Thermal loads are not considered to relieve primary loads induced by gravity, other sustained loads, or seismic events.

3) Installation tolerances are not considered a source of load reduction unless special installation requirements are required.

4) The required movement in unrestrained directions for the line being supported is tabulated in the table of support loads. The support design is arranged to accommodate this required movement of the piping. Hangers are designed in such a manner that they cannot become disengaged by any movement of the supported pipe.

5) If ASME Code Case N-318-3 is used in the design of integral welded attachments to the piping pressure boundary, the requirements of Regulatory Guide 1.84 are documented in TVA calculations.

d. Deformation Limits

Pipe support stiffness/deflection limitations are required for seismic Category I.

The following criteria are used for support stiffness requirements:

1) All pipe support structural steel, except as described below, was designed to limit the maximum deflection to 0.0625" (based on the greater of the seismic/dynamic load components of the upset or faulted loading conditions, or based on the minimum design load). In addition, the maximum deflection is limited to 0.125" (based on the total design load). These analyses were performed independently for each restrained direction (axis) at the point of load application.

2) The first dynamic support in each lateral direction adjacent to strain sensitive equipment (i.e., pump, compressor or turbine nozzle) is designed to limit the maximum deflection to 0.0625" (based on the total design load). This analysis was performed independently for each restrained direction (axis) at the point of load application.

3) Except for the unbraced cantilevers, baseplate rotation or deflection due to baseplate flexibility are considered insignificant and, therefore, are not considered. Anchor bolt stiffness is not considered for this evaluation.

4) For supports with a common member (i.e., gang supports) the deflection at the point under consideration due to the simultaneous application of each pipe's dead weight and thermal loads added algebraically are evaluated to determine the maximum deflection for both the hot and cold pipe conditions. The deflection at the point under consideration resulting from the simultaneous application of each pipe's dynamic loads is determined by SRSS method. The total deflection due to dead weight plus thermal, and dynamic loads is evaluated based on absolute summation of the two deflections calculated above.

5) Support components carrying load primarily in axial tension or compression meet the requirements for stiffness without further evaluation. Also, the stiffness/deflection limitations do not apply in the unrestrained support direction (i.e., due to friction loads).

6) Component standard support elements are considered rigid and therefore, no stiffness/deflection evaluation is necessary except as provided in approved design standards.

7) Higher deflection limits may be used if justified on a case-by-case basis.

e. <u>Considerations for the Faulted Condition</u>

Table 3.9-21 identifies the allowed stresses associated with the faulted condition. The faulted load conditions, represented by postulated pipe whip and jet impingement, were evaluated as described in Section 3.6 for all systems, piping, equipment, and structures shown on issued TVA drawings. Piping, systems, pipe supports and other structures that are the targets of postulated pipe whip or jet impingement may require protection to assure safe plant shutdown. This protection is provided by evaluation on a case-by-case basis.

For evaluation of field routed piping or other field located equipment a different method was used. For these cases, a minimum allowable separation method was used for screening piping systems, conduit, and instrument lines to assure that adequate separation exists between these systems and postulated breaks or through-wall leakage cracks in fluid piping. Where adequate separation is not available, piping is relocated, supports were strengthened, supports added, or mitigative devices are provided to prevent unacceptable loads.

Piping and supports subjected to a jet force from a break in piping of equal or less nominal size and wall thickness are assumed to receive no unacceptable damage, provided the target piping and supports were designed in accordance with accepted codes; i.e. ASME Section III, ANSI B31.1, AISC, etc.

f. Results

The design information for pipe supports of systems analyzed by TVA was tabulated on tables of support design loads and included in the problem file as indicated in Section 3.9.3.1.2.

2. Mechanical Equipment and Component Supports

TVA-designed supports for Category I equipment and components satisfy the AISC allowable stress limits described in Section 3.8.4.5.2 and the stiffness requirements described in Section 3.7.3.16.5. Valve actuator supports were designed as pipe supports. Valve actuator tiebacks have special stiffness requirements to limit valve extended, structure stresses under load.

Vendor-supplied equipment and component supporting structures that were provided as part of the equipment assembly, were seismically qualified as part of the equipment package. This qualification is described in Section 3.9.3.1.2.

3.9.4 Control Rod System

3.9.4.1 Descriptive Information of CRDS

Refer to Section 4.2.3.

3.9.4.2 Applicable CRDS Design Specifications

Refer to Sections 4.2.3.1.4 and 4.2.3.2.2.

3.9.4.3 Design Loadings, Stress Limits, and Allowable Deformations

Refer to Section 4.2.3.3.1.

3.9.4.4 CRDS Performance Assurance Program

Refer to Section 4.2.3.4.2.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Design Arrangements

For verification that changes in design from those in previously licensed plants of similar design do not affect the flow-induced vibration behavior, refer to Section 3.9.2.3.

3.9.5.2 Design Loading Conditions

Refer to Section 4.2.2.3.

3.9.5.3 Design Loading Categories

Refer to Section 4.2.2.4.

3.9.5.4 Design Basis

Refer to Section 4.2.2.5.

3.9.6 Inservice Testing of Pumps and Valves

For Unit 1, inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be conducted to the extent practical in accordance with the 2001 Edition of the ASME OM Code with Addenda through 2003, as required by 10 CFR 50.55a(f). For Unit 2, inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be conducted to the extent practical in accordance with the 2004 Edition through 2006 addenda of the ASME OM Code[25], as required by 10 CFR 50.55a(f). Since the Watts Bar piping systems were designed before the Code was issued, some valves and pump parameters cannot be tested in accordance with the ASME OM Code. These exceptions have been noted in the Inservice Testing Program submittal made to NRC.

The following safety-related pumps will be tested:

1. Centrifugal Charging Pumps
2. Safety Injection Pumps
3. Residual Heat Removal Pumps
4. Containment Spray Pumps
5. Component Cooling System Circulation Pumps
6. Auxiliary Feedwater Pumps
7. Essential Raw Cooling Water Pumps
8. Boric Acid Transfer Pumps
9. ERCW Screenwash Pumps
10. Main Control Room Chilled Water Pumps
11. Electrical Board Room Chilled Water Pumps
12. Shutdown Board Room Chilled Water Pumps

Table 3.9-26 is a tabulation of the various category valves in each of the systems.

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19. DELETED AMENDMENT 11

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TABLE 3.9-1

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS ANALYZED BY WESTINGHOUSE, (EXCLUDING PIPE SUPPORTS) (A)

Condition Classification	Loading Combination (B, C)
Design and Normal	Design pressure Design temperature, Dead weight, nozzle loads
Upset	Upset condition pressure, Upset condition metal temperature, deadweight, OBE, nozzle loads
Emergency	Emergency condition pressure, emergency condition metal temperature, deadweight, nozzle loads
Faulted	Faulted condition pressure, faulted condition metal temperature, deadweight, SSE, nozzle loads

(A) The responses for each loading combination are combine using the absolute sum method. On a case-by-case basis, algebraic summation may be used when signs are known for final design evaluations.

(B) Temperature is used to determine allowable stress only.

(C) Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

WBN

TABLE 3.9-2

STRESS CRITERIA FOR SAFETY RELATED ASME
CLASS 2 AND 3 TANKS ANALYZED BY WESTINGHOUSE

Condition	Stress Limits
Design and Normal	$[_m \le 1.0\ S$ $([_m \text{ or } [_L) + [_b \le 1.5\ S$
Upset	$[_m \le 1.1\ S$ $([_m \text{ or } [_L) + [_b \le 1.65\ S$
Emergency	$[_m \le 1.5\ S$ $([_m \text{ or } [_L) + [_b \le 1.8\ S$
Faulted	$[_m \le 2.0\ S$ $([_m \text{ or } [_L) + [_b \le 2.4\ S$

TABLE 3.9-3

STRESS CRITERIA FOR CATEGORY I ASME CODE CLASS 2 AND CLASS 3 NONACTIVE PUMPS AND PUMP SUPPORTS ANALYZED BY WESTINGHOUSE

Condition	Stress Limits*	Pmax**
Design and Normal	&m ≤ 1.0 S (&m or &L) + &b ≤ 1.5 S	1.0
Upset	&m ≤ 1.1 S (&m or &L) + &b ≤ 1.65 S	1.1
Emergency	&m ≤ 1.5 S (&m or &L) + &b ≤1.80 S	1.2
Faulted	&m ≤ 2.0 S (&m or &L) + &b ≤ 2.4 S	1.5

* Stress limits are taken from ASME III, Subsections NC and ND, or, for pumps procured prior to the incorporation of these limits into ASME III, from Code Case 1636-1.

** The maximum pressure shall not exceed the tabulated factors listed under Pmax times the design pressure.

TABLE 3.9-4 (SHEET 1 of 2)

STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2
AND CLASS 3 VALVES ANALYZED BY WESTINGHOUSE

Condition	Stress Limits (Notes 1-6)	P_{max} (Note 7)
Design & Normal	Valve bodies shall conform to the requirements of ASME Section III, NC-3500 (or ND-3500)	
Upset	$[_m \le 1.1\ S$ $([_m$ or $[_L) + [_b \le 1.65\ S$	1.1
Emergency	$[_m \le 1.5\ S$ $([_m$ or $[_L) + [_b \le 1.80\ S$	1.2
Faulted	$[_m \le 2.0\ S$ $([_m$ or $[_L) + [_b \le 2.4\ S$	1.5

Notes:

1. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation: (1) section modulus and area of a plane, normal to the flow, through the region of valve body crotch is at least 10% greater than the piping connected (or joined) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above shall be multiplied by the ratio of S pipe/S valve. If unable to comply with this requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

TABLE 3.9-4 (SHEET 2 of 2)

STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2
AND CLASS 3 VALVES ANALYZED BY WESTINGHOUSE (Cont'd)

Notes (continued):

5. These rules do not apply to Class 2 and 3 safety relief valves. Safety relief valves are designed in accordance with ASME Section III requirements.

6. Stress limits are taken from ASME III, Subsections NC and ND, or, for valves procured prior to the incorporation of these limits into ASME III, from Code Case 1635-1.

7. The maximum pressure resulting from upset, emergency or faulted conditions shall not exceed the tabulated factors listed under Pmax times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in Table 3.9-4 are considered to be satisfied.

TABLE 3.9-5

INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATION (in) [1.2,3.4]

		Allowable Limit	No Loss-of-Function Limit
Upper Barrel, Expansion/Compression (to ensure sufficient inlet flow area/ and to prevent the barrel from touching any guide tube to avoid disturbing the rod cluster control guide structure)	Inward	4.1	8.2
	Outward	1.0	1.0
Upper Package, Axial Deflection (to maintain the control rod guide structure geometry)[3,4]		0.100	0.150
Rod Cluster Control Guide Tube Deflection As a Beam (to be consistent with conditions under which ability to trip has been tested)[4]		1.0	1.75
Fuel Assembly Thimbles Cross-Section Distortion (to avoid interference between the Control Rods and the guides)[4]		0.036	0.072

Notes:

1. The allowable limit deflection values given above correspond to stress levels for internals structure well below the limiting criteria given by the collapse curves in WCAP-5890.[21] Consequently, for the internals the geometric limitations established to ensure safe shutdown capability are more restrictive than those given by the failure stress criteria.

2. All dimensions in inches.

3. See Reference [22].

4. See Reference [23].

*Only to assure that the plate will not touch a guide tube. (Unit 2 only)

TABLE 3.9-6

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS
ANALYZED BY WESTINGHOUSE

Condition	Design Criteria[1]
Design, Normal and Upset	$\sigma_m \leq 1.0$ S $\sigma_m + \sigma_b \leq 1.5$ S
Emergency	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.65$ S
Faulted	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.8$ S

Note:

(1) The stress limits specified for active pumps are more restrictive than the ASME III limits, to provide assurance that operability will not be impaired for any operating condition.

TABLE 3.9-7

LOAD SOURCES

Load	Description
BUILDING SETTLEMENT	Predicted or Measured Settlement of the Building
DBA	Design Basis Accident Loading
DEADWEIGHT	Weight of Pipe, Insulation, and Fluid
FLUID TRANSIENTS	Transient Loads Due to Valve Operation, Water Hammer
LOCA	Reactor Coolant Loop Movements Due to Loss of Coolant Accident
OBE	Operating Basis Earthquake
PRESSURE	Internal (or External) Pressure in Pipe
SAM	Seismic Anchor Motions
SSE	Safe Shutdown Earthquake
THERMAL	Operating Temperature, Environmental Temperature, Thermal Anchor Movements
VALVE THRUST	Relief Valve Discharge Thrust Loads
PIPE RUPTURE	Jet Impingement and Pipe Whip Loads

TABLE 3.9-8 (SHEET 1 of 2)

LOADING CONSTITUENTS AND STRESS LIMITS
FOR ASME CLASS 1 PIPING

CONDITION	LOADING CONSTITUENTS[1]	STRESS LIMIT	NB-3650 EQUATION[2]	NOTES
PRIMARY STRESS				
DESIGN (Normal & Upset)	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, OBE INERTIA, VALVE THRUST, FLUID TRANSIENTS	$1.5S_m$	9	
EMERGENCY	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, OBE INERTIA, VALVE THRUST, FLUID TRANSIENTS	$2.25S_m$	9	
FAULTED	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS SSE INERTIA, VALVE THRUST, FLUID TRANSIENTS, DBA INERTIA, LOCA, PIPE RUPTURE	$3.0S_m$	9	5
PRIMARY AND SECONDARY STRESS				
NORMAL & UPSET	PRESSURE, THERMAL, THERMAL ANCHOR MOVEMENT, THERMAL LINEAR GRADIENT, THERMAL DISCONTINUITY, VALVE THRUST, FLUID TRANSIENT, OBE, OBE SAM	$3.0S_m$	0	3
	PRESSURE, THERMAL, THERMAL ANCHOR MOVEMENT, THERMAL NONLINEAR GRADIENT, THERMAL LINEAR GRADIENT, THERMAL DISCONTINUITY, VALVE THRUST, FLUID TRANSIENT, OBE, OBE SAM		11	4,8
	THERMAL, THERMAL ANCHOR MOVEMENTS	$3.0S_m$	12	3

TABLE 3.9-8 (SHEET 2 of 2)

LOADING CONSTITUENTS AND STRESS LIMITS
FOR ASME CLASS 1 PIPING

CONDITION	LOADING CONSTITUENTS[1]	STRESS LIMIT	NB-3650 EQUATION[2]	NOTES
PRIMARY AND SECONDARY STRESS (continued)				
NORMAL & UPSET	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, OBE INERTIA, VALVE THRUST, FLUID TRANSIENTS, THERMAL DISCONTINUITY	$3.0S_m$	13	3
TESTING	PRESSURE	$0.9S_y$		7,8
	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS	$1.35S_y$		
PRESSURE DESIGN		PRESSURE LIMITS		
DESIGN	DESIGN PRESSURE	P_a		6
UPSET	Max. Service PRESSURE	P_a		6
EMERGENCY	Max. Service PRESSURE	$1.5P_a$		6
FAULTED	Max. Service PRESSURE	$2.0P_a$		6

NOTES

1. Loads which are not concurrent need not be combined.
2. All references are for ASME Code, Subsection NB for Class 1 piping.
3. If the requirements of equation 10 are not met, then the requirements of equations 12 & 13 must be met.
4. Salt for all load sets are calculated per NB-3653.3 and then, the cumulative usage factor per NB-3653.4 and NB-3653.5. The cumulative usage factor shall not exceed 1.0.
5. The design measures taken to protect against pipe rupture loads and the evaluation of these loads is described in Section 3.6.
6. The maximum allowable internal pressure value Pa is calculated per NB-3640, 3655 and 3656.
7. The testing limits are per NB-3226.
8. If there are more than 10 hydrostatic, pneumatic or other tests, then such extra tests shall be included in the fatigue evaluation of the component.

TABLE 3.9-9 (SHEET 1 of 2)

LOADING CONSTITUENTS AND STRESS LIMITS
FOR CATEGORY I ASME CLASS 2 AND 3 PIPING

CONDITION	LOADING CONSTITUENTS[1]	STRESS LIMIT	NC-3652 EQUATION[2]	NOTES
NORMAL	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS	1.0 S_h	8	
UPSET	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, OBE INERTIA, VALVE THRUST, FLUID TRANSIENTS	$1.2S_h$	9	4
EMERGENCY	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, OBE INERTIA, VALVE THRUST, FLUID TRANSIENTS	$1.8S_h$	9	
FAULTED	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, SSE INERTIA, VALVE THRUST, FLUID TRANSIENTS, DBA INERTIA, LOCA, PIPE RUPTURE	$2.4S_h$	9	5
SECONDARY				
EXPANSION	THERMAL, OBE SAM	S_A	10	3,4
PRESSURE + SUSTAINED + EXPANSION	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, THERMAL, OBE SAM	S_A+S_h	11	3,4
ONE TIME SECONDARY	BUILDING SETTLEMENT, DBA SCV MOVEMENT COLD SPRING	$3.0S_c$ $0.5(S_A + S_h)$	10A NA	

TABLE 3.9-9 (SHEET 2 of 2)

LOADING CONSTITUENTS AND STRESS LIMITS
FOR CATEGORY I ASME CLASS 2 AND 3 PIPING

CONDITION	LOADING CONSTITUENTS(1)	STRESS LIMIT	NC-3652 EQUATION2	NOTES
TEST	PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS	$1.2S_h$	NA	
TEST	PRESSURE, DEADWEIGHT, THERMAL, OTHER SUSTAINED LOADS	S_A+S_h	NA	

NOTES

1. Loads which are not concurrent need not be combined.

2. All references are for ASME Code, Subsection NC for Class 2 piping. The corresponding equations in ASME Code Subsection ND for Class 3 should be used as applicable.

3. The requirements of either Equation 10 or Equation 11 must be met.

4. The effects of OBE Seismic Anchor Movements may be excluded from Equations 10 and 11 if they are included in Equation 9 Upset.

5. The design measures taken to protect against pipe rupture loads and the evaluation of these loads is described in Section 3.6.

TABLE 3.9-10

LOADING CONSTITUENTS AND STRESS LIMITS FOR ACTIVE VALVE EVALUATION

LOADING CONSTITUENTS[1]	STRESS LIMIT[3]
PRESSURE, DEADWEIGHT, OTHER SUSTAINED LOADS, THERMAL, SSE INERTIA, SSE SAM, VALVE THRUST, FLUID TRANSIENTS, DBA INERTIA, LOCA, PIPE RUPTURE[2]	$0.76S_y$[4] $1.0S_y$[4] (for swing check valves)

NOTES

(1) Loads which are not concurrent need not be combined.

(2) The design measures taken to protect against pipe rupture loads and the evaluation of these loads are described in Section 3.6.

(3) The stress is calculated using the section modulus of the attached pipe.

(4) The value of pipe yield stress, S_y, in this table is determined from the code of record for piping analysis (reference Section 3.7.3.8.1).

TABLE 3.9-11
TABLE 3.9-12

DELETED

TABLE 3.9-13A

DESIGN LOADS FOR CATEGORY I PIPING SUPPORTS

CONDITION	LOADING CONSTITUENTS[1]	NOTES
NORMAL	DEADWEIGHT, OTHER SUSTAINED, LOADS, THERMAL, COLD SPRING BUILDING SETTLEMENT	
UPSET	DEADWEIGHT, OTHER SUSTAINED LOADS, THERMAL, OBE INERTIA, OBE SAM, VALVE THRUST, FLUID TRANSIENTS, COLD SPRING	
EMERGENCY	DEADWEIGHT, OTHER SUSTAINED LOADS, THERMAL, OBE INERTIA, OBE SAM, VALVE THRUST, FLUID TRANSIENTS, COLD SPRING	
FAULTED	DEADWEIGHT, OTHER SUSTAINED LOADS, THERMAL, SEE INERTIA, SSE SAM, VALVE THRUST, FLUID TRANSIENTS, DBA INERTIA, LOCA, PIPE RUPTURE,COLD SPRING	2
TEST	DEADWEIGHT, OTHER SUSTAINED LOADS, THERMAL, COLD SPRING	

NOTES:

1. Loads which are not concurrent need not be combined.

2. The design measures taken to protect against pipe rupture loads and the evaluation of these loads is described in Section 3.6.

TABLE 3.9-13B

DESIGN LOADING COMBINATIONS FOR CATEGORY I ASME CODE CLASS 2 AND 3 LINE MOUNTED*** COMPONENTS AND COMPONENT SUPPORTS ANALYZED BY TVA

Condition Classification	Loading Combination
Design and Normal	Design pressure Design temperature* Dead weight, nozzle loads,** Operating Loads
Upset	Upset condition pressure, Upset condition metal temperature*, deadweight, OBE, nozzle loads,** Operating Loads
Emergency	Emergency condition pressure, emergency condition metal temperature*, deadweight, OBE, nozzle loads,** Operating Loads
Faulted	Faulted condition pressure, faulted condition metal temperature*, deadweight, SSE, nozzle loads,** Operating Loads, DBA

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* Temperature is used to determine allowable stress only.

** Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

*** Line-mounted component (valve) design load combinations correspond to the attached piping load combinations for normal, upset, emergency, and faulted conditions.

TABLE 3.9-14

STRESS CRITERIA FOR CATEGORY I
ASME CLASS 2 AND CLASS 3 TANKS ANALYZED BY TVA

Condition	Stress Limits[1]	P_{max}[2]
Design and Normal	$P_m \leq 1.0\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.5\ S_h$	1.0
Upset	$P_m \leq 1.1\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.65\ S_h$	1.1
Emergency	$P_m \leq 1.5\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.8\ S_h$	1.2
Faulted	$P_m \leq 2.0\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 2.4\ S_h$	1.5

Pm, PL, Pb, and Sh are as defined in Table 3.9-18.

NOTES

(1) The stress allowable values given above were permitted for design, evaluation, and modification activities. As an alternative, a simplified approach was also permitted. By the alternative approach, in addition to meeting the applicable design condition requirements, the tank was analyzed for the faulted condition and tank stresses were limited to 1.2 times the applicable ASME code design/normal condition primary stress allowables.

(2) The maximum pressure was not permitted to exceed the tabulated factors listed under P_{max} times the design pressure.

TABLE 3.9-15

STRESS CRITERIA FOR CATEGORY I ASME CODE CLASS 2 AND CLASS 3 NONACTIVE PUMPS ANALYZED BY TVA

Condition	Stress Limits[(2)]	P_{max}[(1)]
Design and Normal	$P_m \leq 1.0\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.5\ S_h$	1.0
Upset	$P_m \leq 1.1\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.65\ S_h$	1.1
Emergency	$P_m \leq 1.5\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.8\ S_h$	1.2
Faulted	$P_m \leq 2.0\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 2.4\ S_h$	1.5

P_m, P_L, P_b, and S_h are as defined in Table 3.9-18.

Notes:

(1) The maximum pressure was not permitted to exceed the tabulated factors listed under P_{max} times the design pressure.

(2) The stress allowable values given above were permitted for design, evaluation, and modification activities. As an alternative, a simplified approach was permitted for pumps procured prior to September 1, 1974. By this alternative approach, in addition to meeting applicable ASME code design condition requirements, the pump was analyzed for the faulted condition and pump stresses were limited to 1.2 times the applicable ASME code design/normal condition primary stress allowables.

TABLE 3.9-16 (SHEET 1 of 2)

STRESS CRITERIA FOR CATEGORY I ASME CODE CLASS 2
AND CLASS 3 NONACTIVE VALVES ANALYZED BY TVA

Condition	Stress Limits (Notes 1-6)	Pmax(note 7)
Design and Normal	Pm = Sh (Pm or PL) + Pb = 1.5 Sh	1.0
Upset	Pm=1.1 Sh (Pm or PL) + Pb = 1.65 Sh	1.1
Emergency	Pm = 1.5 Sh (Pm or PL) + Pb = 1.8 Sh	1.2
Faulted	Pm =2.0 Sh (Pm or PL) + Pb = 2.4 Sh	1.5

Pm, PL, Pb, and Sh are as defined in Table 3.9-18.

Notes:

1. Valve nozzle (Piping load) stress analysis is not required when both the following conditions are satisfied by calculation: (1) section modulus and area of a plane, normal to the flow, through the region of valve body crotch is at least 10% greater than the piping connected (or joined) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, to connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above shall be multiplied by the ratio of Spipe/Svalve. If unable to comply with this requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable. Non-pressure boundary components are evaluated against the stress limits of AISC or other justifiable reference. The 33% increase permitted by AISC for abnormal loads is applicable for upset, emergency and faulted conditions.

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

5. These rules do not apply to Class 2 and 3 safety relief valves. Safety relief valves are designed in accordance with ASME Section III requirements.

TABLE 3.9-16 (SHEET 2 of 2)

STRESS CRITERIA FOR CATEGORY I ASME CODE CLASS 2
AND CLASS 3 NONACTIVE VALVES ANALYZED BY TVA

6. The stress allowable values given above were permitted for design, evaluation, and modification activities. As an alternative, a simplified approach was permitted for valves procured prior to September 1, 1974. By this alternative approach, in addition to meeting the applicable ASME code design condition requirements, the valve was analyzed or tested per IEEE 344-1971 requirements for the faulted condition. When qualified by analysis using this alternative, the valve stresses were limited to 1.2 times the applicable ASME code design/normal condition primary stress allowables and the valve extended structure stresses were limited to 1.33 times the AISC code normal stress allowables.

7. The maximum pressure resulting from upset or faulted conditions shall not exceed the tabulated factors listed under Pmax times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

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TABLE 3.9-17 (Sheet 1 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Chemical and Volume Control System (62)	FCV-62-61	8112	4	Motor	Gate	Containment Isolation
	FCV-62-63	8100	4	Motor	Gate	Containment Isolation
	FCV-62-69	LCV-460	3	Air	Globe	Letdown Isolation
	FCV-62-70	LCV-459	3	Air	Globe	Letdown Isolation
	FCV-62-72	8149A	2	Air	Globe	Containment Isolation
	FCV-62-73	8149B	2	Air	Globe	Containment Isolation
	FCV-62-74	8149C	2	Air	Globe	Containment Isolation
	FCV-62-77	8152	2	Air	Globe	Containment Isolation
	FCV-62-84	8145	3	Air	Globe	Aux Spray Isolation
	FCV-62-90	8105	3	Motor	Gate	ECCS Flowpath Integrity
	FCV-62-91	8106	3	Motor	Gate	ECCS Flowpath Integrity
	FCV-62-76	8306A	2	Air	Globe	Containment Isolation
	LCV-62-132	LCV-112B	4	Motor	Gate	CVCS Charging Pump Suction
	LCV-62-133	LCV-112C	4	Motor	Gate	CVCS Charging Pump Suction
	LCV-62-135	LCV-112D	8	Motor	Gate	CVCS Charging Pump Suction
	LCV-62-136	LCV-112E	8	Motor	Gate	CVCS Charging Pump Suction
	CKV-62-504	8546	8	Self Actuated	Check	CVCS Charging Pump Suction
	RFV-62-505	8124	3/4	Self Actuated	Relief	CCP Suction Relief Valve
	CKV-62-543[1]	8381	3	Self Actuated	Check	Containment Isolation
	CKV-62-560[1]	8368A	2	Self Actuated	Check	Containment Isolation
	CKV-62-561[1]	8368B	2	Self Actuated	Check	Containment Isolation
	CKV-62-562[1]	8368C	2	Self Actuated	Check	Containment Isolation
	CKV-62-563[1]	8368D	2	Self Actuated	Check	Containment Isolation
	CKV-62-576	8367A	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-577	8367B	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-578	8367C	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-579	8367D	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-638	8557	3	Self Actuated	Check	Normal Charging Isolation
	1-CKV-62-639	---	¾	Self Actuated	Check	Seal Water 1-FCV-62-61 Bypass
	CKV-62-640	8556	3	Self Actuated	Check	Alternate Charging Isolation
	1-RFV-62-649	---	2	Self Actuated	Relief	Seal Water Hx Relief Valve

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TABLE 3.9-17 (Sheet 2 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
	CKV-62-659	8378	3	Self Actuated	Check	Normal Charging Isolation
	CKV-62-660	8379	3	Self Actuated	Check	Alternate Charging Isolation
	CKV-62-661	8377	3	Self Actuated	Check	Auxiliary Spray Isolation
	CKV-62-525	8481A	4	Self Actuated	Check	Prevent Backflow thru Centrifugal Charging Pump
	CKV-62-532	8481B	4	Self Actuated	Check	Prevent Backflow thru Centrifugal Charging Pump
	FCV-62-1228	8870B	1	Air	Globe	Hydrogen vent header isolation valve
	FCV-62-1229	8870A	1	Air	Globe	Hydrogen vent header isolation Valve
	1-RFV-62-662	8117	2	Self Actuated	Relief	Cntmt Isolation Thermal Relief
	1-RFV-62-1221		3/4	Self Actuated	Relief	CCP 1A-A Over Pressure
	1-RFV-62-1222		3/4	Self Actuated	Relief	CCP 1B-B Over Pressure
SafetyInjection (63)	FCV-63-1	8812	14	Motor	Gate	Prevent Radioactive Release in Recirc Mode
	FCV-63-3	8813	2	Motor	Globe	Prevent Radioactive Release in Recirc Mode
	FCV-63-4	8814	2	Motor	Globe	Prevent Radioactive Release in Recirc Mode
	FCV-63-5	8806	8	Motor	Gate	Prevent Radioactive Release in Recirc Mode
	FCV-63-6	8807B	4	Motor	Gate	Sis Recirc Flowback Integrity
	FCV-63-7	8807A	4	Motor	Gate	Sis Recirc Flowback Integrity
	FCV-63-8	8804A	8	Motor	Gate	Sis Recirc Flowpath Integrity
	FCV-63-11	8804B	8	Motor	Gate	Sis Recirc Flowpath Integrity
	FCV-63-23	8888	1	Air	Globe	Containment Isolation
	FCV-63-25	8801B	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-26	8801A	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-47	8923A	6	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-48	8923B	6	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-64	8880	1	Air	Globe	Containment Isolation
	FCV-63-71	8871	3/4	Air	Globe	Containment Isolation
	FCV-63-72	8811A	18	Motor	Gate	CNTMT Sump Isolation
	FCV-63-73	8811B	18	Motor	Gate	CNTMT Sump Isolation
	FCV-63-84	8964	3/4	Air	Globe	Containment Isolation
	FCV-63-93	8809A	8	Motor	Gate	RHR Cold Leg Injection Line Isolation

WBN

TABLE 3.9-17 (Sheet 3 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
	FCV-63-94	8809B	8	Motor	Gate	RHR Cold Leg Injection Line Isolation
	FCV-63-152	8821A	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-153	8821B	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-156	8802A	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-157	8802B	4	Motor	Gate	ECCS Flowpath Intgrity
	FCV-63-172	8840	12	Motor	Gate	RHR Hot Leg Recirc
	FCV-63-175	8920	3/4	Motor	Globe	Prevent Radioactive Release in Recirc Mode
	FCV-63-185	---	3/4	Air	Globe	Leak Test Line Isolation
	CKV-63-524	8922A	4	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-63-526	8922B	4	Self Actuated	Check	Prevent Backflow thru Nonoperational Train
	CKV-63-543	8905A	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-545	8905C	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-547	8905B	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-549	8905D	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-551	8819A	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-553	8819B	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-555	8819C	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-557	8819D	2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-558	8949D	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-559	8949B	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-560	8948A	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-561	8948B	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-562	8948C	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-563	8948D	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-581	8805	3	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-586	8900A	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-587	8900B	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-588	8900C	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-589	8900D	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-622	8956A	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot

WBN

TABLE 3.9-17 (Sheet 4 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
	CKV-63-623	8956B	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-624	8956C	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-625	8956D	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-632	8818B	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-633	8818A	6	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-634	8818C	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-640	8841A	8	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press Bound. Prot
	CKV-63-635	8818D	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press Bound. Prot
	CKV-63-641	8949A	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press Bound. Prot
	CKV-63-643	8841B	8	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press Bound. Prot
	CKV-63-644	8949C	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press Bound. Prot
	CKV-63-502[3]	8958	12	Self Actuated	Check	ECCS Flowpath Integrity
	CKV-63-510[3]	8926	8	Self Actuated	Check	ECCS Flowpath Integrity
	RFV-63-511[3]	8858	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-534[3]	8853A	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-535[3]	8851	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-536[3]	8853B	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-577[3,4]	8852	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-602[3]	8855A	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-603[3]	8855B	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-604[3]	8855C	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-605[3]	8855D	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-626[3]	8856A	2	Self Actuated	Angle	SI Pressure Release
	RFV-63-627[3]	8856B	3	Self Actuated	Angle	SI Pressure Release
	RFV-63-637[3]	8842	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-835	----	¾	Self Actuated	Angle	SI Pressure Release
	CKV-63-868	---	1	Self Actuated	Check	Containment Isolation

TABLE 3.9-17 (Sheet 5 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Reactor Coolant System (68)	FCV-68-305	8033	3/4	Air	Daphragm	Containment Isolation
	FCV-68-307	8025	3/8	Air	Globe	Containment Isolation
	FCV-68-308	8026	3/8	Air	Globe	Containment Isolation
	FCV-68-332	8000A	3	Motor	Gate	PORV Isolation
	FCV-68-333	8000B	3	Motor	Gate	PORV Isolation
	PCV-68-334	PCV-456	3	Solenoid	Globe	PORV
	PCV-68-340A	PCV-455	3	Solenoid	Globe	PORV
	CKV-68-559	8097	4	Self Actuated	Check	Containment Isolation
	RFV-68-563	8010C	6	Self Actuated	Angle	RCS Pressure Release
	RFV-68-564	8010B	6	Self Actuated	Angle	RCS Pressure Release
	RFV-68-565	8010A	6	Self Actuated	Angle	RCS Pressure Release
	FSV-68-394	8012A	1	Solenoid	Globe	Head Vent
	FSV-68-395	8012B	1	Solenoid	Globe	Head Vent
	FSV-68-396	8014B	1	Solenoid	Globe	Head Vent
	FSV-68-397	8014A	1	Solenoid	Globe	Head Vent
	CKV-68-849	---	3/4	Self Actuated	Check	Containment Isolation
Containment Spray System (72)	FCV-72-2	9001B	10	Motor	Gate	Containment Isolation
	FCV-72-21	9017B	12	Motor	Gate	Containment Sump Recirculation
	FCV-72-22	9017A	12	Motor	Gate	Containment Sump Recirculation
	FCV-72-39	9001A	10	Motor	Gate	Containment Isolation
	FCV-72-40	9026A	8	Motor	Gate	Containment Isolation
	FCV-72-41	9026B	8	Motor	Gate	Containment Isolation
	FCV-72-44	9020A	12	Motor	Gate	Containment Sump Recirculation
	FCV-72-45	9020B	12	Motor	Gate	Containment Sump Recirculation
	CKV-72-506	9018A	12	Self Actuated	Check	Containment Sump Recirculation
	CKV-72-507	9018B	12	Self Actuated	Check	Containment Sump Recirculation
	RFV-72-508	9019A	0.75	Self Actuated	Relief	Containment Spray PMP A Suction Pressure Relief
	RFV-72-509	9019B	0.75	Self Actuated	Relief	Containment Spray PMP B Suction Pressure Relief
	CKV-72-548	9011A	10	Self Actuated	Check	Containment Isolation
	CKV-72-549	9011B	10	Self Actuated	Check	Containment Isolation
	CKV-72-562	9022A	8	Self Actuated	Check	RHR Isolation
	CKV-72-563	9022B	8	Self Actuated	Check	RHR Isolation

TABLE 3.9-17 (Sheet 6 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Residual Heat Removal System (74)	FCV-74-1	8702	14	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-2	8701	14	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-3	8700A	14	Motor	Gate	ECCS Flowpath Integrity
	FCV-74-8	8703	10	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-9	8704	10	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-12	FCV-610	3	Motor	Globe	RHR Mini-Flow
	FCV-74-21	8700B	14	Motor	Gate	ECCS Flowpath Integrity
	FCV-74-24	FCV-611	3	Motor	Globe	RHR Mini-Flow
	FCV-74-33	8716A	8	Motor	Gate	ECCS Flowpath Integrity
	FCV-74-35	8716B	8	Motor	Gate	ECCS Flowpath Integrity
	RFV-74-505	8708	3	Self Actuated	Relief	RHR Pump Suction
	CKV-74-514	8730A	8	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-74-515	8730B	8	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-74-544		8	Self Actuated	Check	Prevents pump-to-pump Interaction
	CKV-74-545		8	Self Actuated	Check	Prevents pump-to-pump Interaction
Waste Disposal System (77)	FCV-77-9	9170	3	Air	Diaphragm	Containment Isolation
	FCV-77-10	FCV-1003	3	Air	Diaphragm	Containment Isolation
	FCV-77-16	9159A	3/4	Air	Globe	Containment Isolation
	FCV-77-17	9159B	3/4	Air	Globe	Containment Isolation
	FCV-77-18	9160A	1	Air	Globe	Containment Isolation
	FCV-77-19	9160B	1	Air	Globe	Containment Isolation
	FCV-77-20	9157	1	Air	Globe	Containment Isolation
	FCV-77-127		2	Air	Plug	Containment Isolation
	FCV-77-128		2	Air	Plug	Containment Isolation
	RFV-77-2875		1/2	Self Actuated	Relief	Thermal Relief Valve Penetration X41

TABLE 3.9-17 (Sheet 7 of 7)
UNIT 1

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Spent Fuel Pool Cooling (78)	0-CKV-78-509		8	Self Actuated	Check	Pump A-A Discharge Check Valve
	0-CKV-78-510		8	Self Actuated	Check	Pump B-B Discharge Check Valve
	0-ISV-78-581		10	Manual	Gate	Standby Pump Train A Suction Isolation Valve
	0-ISV-78-582		10	Manual	Gate	Standby Pump Train B Suction Isolation Valve
	0-CKV-78-586		8	Self Actuated	Check	Pump C-S Discharge Check Valve
	0-ISV-78-587		8	Manual	Gate	Standby Pump Train B Discharge Isolation Valve
	0-ISV-78-588		8	Manual	Gate	Standby Pump Train A Discharge Isolation Valve

[1] Testing not required to meet 10 CFR 50 Appendix J.
[2] Testing not required as part of Inservice Testing Program. Testing not required to meet 10 CFR 50 Appendix J. See Table 6.2.4-1.
[3] These components were not committed to meet the requirements of RG 1.48 since these were procured under contracts issued prior to September 1, 1974.
[4] Valve 1-RFV-63-577 is not required to open to mitigate the consequences of an accident. However, the valve must be in the closed position to mitigate the consequences of an accident.

TABLE 3.9-17 (Sheet 1 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Chemical and Volume Control System (62)	FCV-62-61	8112	4	Motor	Gate	Containment Isolation
	FCV-62-63	8100	4	Motor	Gate	Containment Isolation
	FCV-62-69	LCV-460	3	Air	Globe	Letdown Isolation
	FCV-62-70	LCV-459	3	Air	Globe	Letdown Isolation
	FCV-62-72	8149A	2	Air	Globe	Containment Isolation
	FCV-62-73	8149B	2	Air	Globe	Containment Isolation
	FCV-62-74	8149C	2	Air	Globe	Containment Isolation
	FCV-62-77	8152	2	Air	Globe	Containment Isolation
	FCV-62-84	8145	3	Air	Globe	Aux Spray Isolation
	FCV-62-90	8105	3	Motor	Gate	ECCS Flowpath Integrity
	FCV-62-91	8106	3	Motor	Gate	ECCS Flowpath Integrity
	FCV-62-76	8306A	2	Air	Globe	Containment Isolation
	LCV-62-132	LCV-112B	4	Motor	Gate	CVCS Charging Pump Suction
	LCV-62-133	LCV-112C	4	Motor	Gate	CVCS Charging Pump Suction
	LCV-62-135	LCV-112D	8	Motor	Gate	CVCS Charging Pump Suction
	LCV-62-136	LCV-112E	8	Motor	Gate	CVCS Charging Pump Suction
	CKV-62-504	8546	8	Self Actuated	Check	CVCS Charging Pump Suction
	RFV-62-505	8124	-	Self Actuated	Relief	CCP Suction Relief Valve
	CKV-62-507	---	1	Self Actuated	Check	CCP Suction Chem Feed Check Valve
	CKV-62-543	8381	3	Self Actuated	Check	Containment Isolation
	CKV-62-560[1]	8368A	2	Self Actuated	Check	Containment Isolation
	CKV-62-561[1]	8368B	2	Self Actuated	Check	Containment Isolation
	CKV-62-562[1]	8368C	2	Self Actuated	Check	Containment Isolation
	CKV-62-563[1]	8368D	2	Self Actuated	Check	Containment Isolation
	CKV-62-576	8367A	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-577	8367B	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-578	8367C	2	Self Actuated	Check	CCP Discharge Header Integrity
	CKV-62-579	8367D	2	Self Actuated	Check	CCP Discharge Header Integrity

TABLE 3.9-17 (Sheet 2 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
	CKV-62-638	8557	3	Self Actuated	Check	Normal Charging Isolation
	CKV-62-639		:	Self Actuated	Check	Seal Water 1-FCV-62-61 Bypass
	CKV-62-640	8556	3	Self Actuated	Check	Alternate Charging Isolation
	RFV-62-649		2	Self Actuated	Relief	Seal Water Hx Relief Valve
	CKV-62-659	8378	3	Self Actuated	Check	Normal Charging Isolation
	CKV-62-660	8379	3	Self Actuated	Check	Alternate Charging Isolation
	CKV-62-661	8377	3	Self Actuated	Check	Auxiliary Spray Isolation
	CKV-62-525	8481A	4	Self Actuated	Check	Prevent Backflow thru Centrifugal Charging Pump
	CKV-62-532	8481B	4	Self Actuated	Check	Prevent Backflow thru Centrifugal Charging Pump
	FCV-62-1228	8870B	1	Air	Globe	Hydrogen vent header Isolation Valve
	FCV-62-1229	8870A	1	Air	Globe	Hydrogen vent header Isolation Valve
	RFV-62-662	8117	2	Self Actuated	Relief	Cntmt Isolation Thermal Relief
	RFV-62-1221		:	Self Actuated	Relief	CCP 1A-A Over Pressure
	RFV-62-1222		:	Self Actuated	Relief	CCP 1B-B Over Pressure
Safety Injection (63)	FCV-63-1	8812	14	Motor	Gate	Prevent Radioactive Release in Recirc Mode
	FCV-63-3	8813	2	Motor	Globe	Prevent Radioactive Release in Recirc Mode
	FCV-63-4	8814	2	Motor	Globe	Prevent Radioactive Release in Recirc Mode
	FCV-63-5	8806	8	Motor	Gate	Prevent Radioactive Release in Recirc Mode
	FCV-63-6	8807B	4	Motor	Gate	Sis Recirc Flowback Integrity
	FCV-63-7	8807A	4	Motor	Gate	Sis Recirc Flowback Integrity
	FCV-63-8	8804A	8	Motor	Gate	Sis Recirc Flowpath Integrity
	FCV-63-11	8804B	8	Motor	Gate	Sis Recirc Flowpath Integrity
	FCV-63-23	8888	1	Air	Globe	Containment Isolation
	FCV-63-25	8801B	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-26	8801A	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-47	8923A	6	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-48	8923B	6	Motor	Gate	ECCS Flowpath Integrity

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TABLE 3.9-17 (Sheet 3 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Safety Injection (63)	FCV-63-64	8880	1	Air	Globe	Containment Isolation
	FCV-63-71	8871	3/4	Air	Globe	Containment Isolation
	FCV-63-72	8811A	18	Motor	Gate	CNTMT Sump Isolation
	FCV-63-73	8811B	18	Motor	Gate	CNTMT Sump Isolation
	FCV-63-84	8964	3/4	Air	Globe	Containment Isolation
	FCV-63-93	8809A	8	Motor	Gate	RHR Cold Leg Injection Line Isolation
	FCV-63-94	8809B	8	Motor	Gate	RHR Cold Leg Injection Line Isolation
	FCV-63-152	8821A	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-153	8821B	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-156	8802A	4	Motor	Gate	ECCS Flowpath Integrity
	FCV-63-157	8802B	4	Motor	Gate	ECCS Flowpath Intgrity
	FCV-63-172	8840	12	Motor	Gate	RHR Hot Leg Recirc
	FCV-63-175	8920	:	Motor	Globe	Prevent Radioactive Release in Recirc Mode
	FCV-63-185	---	:	Air	Globe	Leak Test Line Isolation
	CKV-63-524	8922A	4	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-63-526	8922B	4	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-63-543	8905A	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-545	8905C	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-547	8905B	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-549	8905D	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-551	8819A	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-553	8819B	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-555	8819C	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-557	8819D	2	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-558	8949D	6	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot
	CKV-63-559	8949B	6	Self Actuated	Check	ECCS Flowpath Integrity/ RCS Press. Bound. Prot

TABLE 3.9-17 (Sheet 4 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Safety Injection (63)	CKV-63-560	8948A	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-561	8948B	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-562	8948C	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-563	8948D	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-581	8805	3	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-586	8900A	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-587	8900B	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-588	8900C	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-589	8900D	1-1/2	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-622	8956A	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-623	8956B	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-624	8956C	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-625	8956D	10	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-632	8818B	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-633	8818A	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-634	8818C	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-640	8841A	8	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-635	8818D	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-641	8949A	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-643	8841B	8	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-644	8949C	6	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-5023	8958	12	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot
	CKV-63-5103	8926	8	Self Actuated	Check	ECCS Flowpath Integrity/RCS Press. Bound. Prot

WBN

TABLE 3.9-17 (Sheet 5 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Safety Injection (63)	RFV-63-511[3]	8858	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-534[3]	8853A	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-535[3]	8851	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-536[3]	8853B	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-577[3,4]	8852	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-602[3]	8855A	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-603[3]	8855B	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-604[3]	8855C	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-605[3]	8855D	1	Self Actuated	Angle	SI Pressure Release
	RFV-63-626[3]	8856A	2	Self Actuated	Angle	SI Pressure Release
	RFV-63-627[3]	8856B	3	Self Actuated	Angle	SI Pressure Release
	RFV-63-637[3]	8842	3/4	Self Actuated	Angle	SI Pressure Release
	RFV-63-835	----	3/4	Self Actuated	Angle	SI Pressure Release
	CKV-63-868	---	1	Self Actuated	Check	Containment Isolation
	RFV-63-28	---	3/4	Self Actuated	Relief	Penetration X-30 pressure relief containment isolation
Reactor Coolant System (68)	FCV-68-305	8033	3/4	Air	Diaphragm	Containment Isolation
	FCV-68-307	8025	3/8	Air	Globe	Containment Isolation
	FCV-68-308	8026	3/8	Air	Globe	Containment Isolation
	FCV-68-332	8000A	3	Motor	Gate	PORV Isolation
	FCV-68-333	8000B	3	Motor	Gate	PORV Isolation
	*PCV-68-334	8097	3	Solenoid	Globe	PORV
	*PCV-68-340A	8010C	3	Solenoid	Globe	PORV
	CKV-68-559	8010B	4	Self Actuated	Check	Containment Isolation
	RFV-68-563	8010A	6	Self Actuated	Angle	RCS Pressure Release
	RFV-68-564	8012A	6	Self Actuated	Angle	RCS Pressure Release
	RFV-68-565	8012B	6	Self Actuated	Angle	RCS Pressure Release
	FSV-68-394	8014B	1	Solenoid	Globe	Head Vent
	FSV-68-395	8014A	1	Solenoid	Globe	Head Vent
	FSV-68-396	---	1	Solenoid	Globe	Head Vent
	FSV-68-397		1	Solenoid	Globe	Head Vent
	CKV-68-849		3/4	Self Actuated	Check	Containment Isolation

* Valves PCV-68-334 and PCV-68-340A were originally supplied by Westinghouse as valve numbers PCV-456 and PCV-455, respectively. Although these valves were in the original Westinghouse scope of supply, replacement valves have been supplied by TVA.

TABLE 3.9-17 (Sheet 6 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Containment Spray System (72)	FCV-72-2	9001B	10	Motor	Gate	Containment Isolation
	FCV-72-21	9017B	12	Motor	Gate	Containment Sump Recirculation
	FCV-72-22	9017A	12	Motor	Gate	Containment Sump Recirculation
	FCV-72-39	9001A	10	Motor	Gate	Containment Isolation
	FCV-72-40	9026A	8	Motor	Gate	Containment Isolation
	FCV-72-41	9026B	8	Motor	Gate	Containment Isolation
	FCV-72-44	9020A	12	Motor	Gate	Containment Sump Recirculation
	FCV-72-45	9020B	12	Motor	Gate	Containment Sump Recirculation
	CKV-72-506	9018A	12	Self Actuated	Check	Containment Sump Recirculation
	CKV-72-507	9018B	12	Self Actuated	Check	Containment Sump Recirculation
	RFV-72-508	9019A	0.75	Self Actuated	Relief	Containment Spray PMP A Suction Pressure Relief
	RFV-72-509	9019B	0.75	Self Actuated	Relief	Containment Spray PMP B Suction Pressure Relief
	CKV-72-548	9011A	10	Self Actuated	Check	Containment Isolation
	CKV-72-549	9011B	10	Self Actuated	Check	Containment Isolation
	CKV-72-562	9022A	8	Self Actuated	Check	RHR Isolation
	CKV-72-563	9022B	8	Self Actuated	Check	RHR Isolation
Heat Removal System (74)	FCV-74-1	8702	14	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-2	8701	14	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-3	8700A	14	Motor	Gate	ECCS Flowpath Integrity
	FCV-74-8	8703	10	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-9	8704	10	Motor	Gate	RCS Pressure Boundary Protection
	FCV-74-12	FCV-610	3	Motor	Globe	RHR Mini-Flow
	FCV-74-21	8700B	14	Motor	Gate	ECCS Flowpath Integrity
	FCV-74-24	FCV-611	3	Motor	Globe	RHR Mini-Flow
	FCV-74-33	8716A	8	Motor	Gate	ECCS Flowpath Integrity
	FCV-74-35	8716B	8	Motor	Gate	ECCS Flowpath Integrity
	RFV-74-505	8708	3	Self Actuated	Relief	RHR Pump Suction
	CKV-74-514	8730A	8	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-74-515	8730B	8	Self Actuated	Check	Prevent Backflow thru Nonoperating Train
	CKV-74-544		8	Self Actuated	Check	Prevents pump-to-pump Interaction
	CKV-74-545		8	Self Actuated	Check	Prevents pump-to-pump Interaction

TABLE 3.9-17 (Sheet 7 of 7)
UNIT 2

ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS

System Name	TVA Valve No.	W Valve No.	Size Inches	Actuation	Type	Function/Description
Waste Disposal System (77)	FCV-77-9	9170	3	Air	Diaphragm	Containment Isolation
	FCV-77-10	FCV-1003	3	Air	Diaphragm	Containment Isolation
	FCV-77-16	9159A	3/4	Air	Globe	Containment Isolation
	FCV-77-17	9159B	3/4	Air	Globe	Containment Isolation
	FCV-77-18	9160A	1	Air	Globe	Containment Isolation
	FCV-77-19	9160B	1	Air	Globe	Containment Isolation
	FCV-77-20	9157	1	Air	Globe	Containment Isolation
	FCV-77-127		2	Air	Plug	Containment Isolation
	FCV-77-128		2	Air	Plug	Containment Isolation
	RFV-77-2875		1/2	Self Actuated	Relief	Thermal Relief Valve Penetration X41
Spent Fuel Pool Cooling (78)	CKV-78-509		8	Self Actuated	Check	Pump A-A Discharge Check Valve
	CKV-78-510		8	Self Actuated	Check	Pump B-B Discharge Check Valve
	ISV-78-581		10	Manual	Gate	Standby Pump Train A Suction Isolation Valve
	ISV-78-582		10	Manual	Gate	Standby Pump Train B Suction Isolation Valve
	CKV-78-586		8	Self Actuated	Check	Pump C-S Discharge Check Valve
	ISV-78-587		8	Manual	Gate	Standby Pump Train B Discharge Isolation Valve
	ISV-78-588		8	Manual	Gate	Standby Pump Train A Discharge Isolation Valve

[1] Testing not required to meet 10 CFR 50 Appendix J.

[2] Footnote 2 deleted by Amendment 113.

[3] These components were not committed to meet the requirements of RG 1.48 since these were procured under contracts issued prior to September 1, 1974.

[4] Valve 1-RFV-63-577 is not required to open to mitigate the consequences of an accident. However, the valve must be in the closed position to mitigate the consequences of an accident.

TABLE 3.9-18

STRESS LIMITS FOR ACTIVE CATEGORY I ASME CLASS 2 AND 3 PUMPS

Condition	Stress Limits[(1) (2)]
Design Normal and Upset	$P_m \leq 1.0\ S_h$ $(P_m \text{ or } P_L) + Pb \leq 1.5\ S_h$
Emergency	$P_m \leq 1.1\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.65\ S_h$
Faulted	$P_m \leq 1.2\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.8\ S_h$

S_h = Material allowable stress at maximum operating temperature from ASME Section III, 1971 Edition, through Summer 1973 Addenda or the applicable code edition specified at the time of procurement.

P_m = Primary general membrane stress, the average primary stress across the solid section under consideration. Excludes effects of discontinuities and concentrations. Produced by pressure, mechanical loads.

P_b = Primary bending stress. This stress is produced by pressure and mechanical loads including inertia earthquake effects but excluding effects of discontinuities and concentrations.

P_L = Primary local membrane stress, the average stress across any solid section under consideration. Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads including inertia earthquake effects.

Notes:

1. The stress allowables given above were permitted for design, evaluation, and modification activities. As an alternative, a simplified approach was permitted for pumps procured prior to September 1, 1974. By this alternative approach, in addition to meeting applicable ASME Code design condition requirements, the pump was analyzed for the faulted condition and pump stresses were limited to 1.2 times the applicable ASME Code design/normal condition primary stress allowables.

2. Active pumps procured after September 1, 1974 also complied with operability tests and analysis requirements described in Section 3.9.3.2.2.2.

TABLE 3.9-19

STRESS LIMITS FOR ACTIVE CATEGORY I ASME CLASS 2 AND 3 VALVES

Condition	Stress Limits (Notes 1-6)
Design & Normal	$P_m \leq 1.0\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.5\ S_h$
Upset	$P_m \leq 1.1\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.65\ S_h$
Emergency	$P_m \leq 1.5\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 1.8\ S_h$
Faulted	$P_m \leq 2.0\ S_h$ $(P_m \text{ or } P_L) + P_b \leq 2.4\ S_h$

Notes:

1. S_h, P_m, P_b, and P_L are defined in Table 3.9-18.
2. Valve nozzle (piping load) stress analysis is not required when section modulus and area of a plane, normal to the flow, through the region of valve body crotch is at least 10% greater than the piping connected (or joined) to the valve body inlet and outlet nozzles.
3. Stress in the valve nozzles resulting from connecting pipe does not exceed the limits listed in this table where S is based on the valve material. To ensure this, the attached pipe stress is limited in accordance with Table 3.9-10 unless justified on a case-by-case basis, or otherwise not part of the pressure boundary.
4. Design requirements listed in this table are not applicable to valve discs, stems, cast rings, or other parts of valves which are contained within the confines of the body and bonnet.
5. The stress allowables given above were permitted for design, evaluation, and modification activities. As an alternative, a simplified approach was permitted for valves procured prior to September 1, 1974. By this alternative approach, in addition to meeting the applicable ASME code design condition requirements, the valve was analyzed or tested on IEEE 344-1971 requirements for the faulted condition. When qualified by analysis using this alternative, the valve stresses were limited to 1.2 times the applicable ASME code design/normal condition primary stress allowables and the valve extended structure stresses were limited to 1.33 times the AISC Code normal stress allowable.
6. Active valves procured after September 1, 1974 also complied with operability test and analysis requirements of R.G. 1.48.

TABLE 3.9-20 (Sheet 1 of 2)
UNIT 1

RELIEF VALVES IN CLASS 2 AUXILIARY SYSTEMS

System	No.	Location	Fluid	Relieving Temperature	Valve Set Pressure	Max. Required Flow Rate	Discharge	Associated Relief Valve
SIS	1	Accumulator Tanks/ N_2 Supply	Nitrogen or Borated Water	120°F	700/750 psig	1500 cfm	Atmosphere	1-RFV-63-602, -603, -604, -605
SIS	2	S. I. Pump Suction	Borated Water	100°F	220 psig	25 gpm	to PRT	1-RFV-63-511; -835
SIS	3	S. I. Pump Discharge	Borated Water	100°F	1750 psig	20 gpm	to PRT	1-RFV-63-534, -535, -536
SIS	4	Boron Injection Tank	Borated Water	180°F	2735 psig	20 gpm	to HT	1-RFV-63-577
RHR	5	Residual Pump Suction	Borated Water	350°F ** 200°F	450 psig 450 psig	480 gpm 690 gpm	to PRT to PRT	1-RFV-74-505
RHR	6	Residual Pump Discharge	Borated Water	120°F	600 psig	20 gpm	to PRT	1-RFV-63-626, -627, -637
CVCS	7	Letdown Line Orifice	Borated Water	347°F **	600 psig	227 gpm	to PRT	1-RFV-62-662
CVCS	8	Seal Water Return Line	Borated Water	250°F **	150 psig	225 gpm	to PRT	1-RFV-62-636
CVCS	9	Letdown Line	Borated Water	200°F	200 psig	200 gpm	to VCT***	1-RFV-62-675
CVCS	10	Seal Water Return Line	Borated Water	150°F	200 psig	180 gpm	to VCT***	1-RFV-62-649
CVCS	11	Volume Control Tank	Hydrogen Nitrogen or Borated Water	130°F	70 psig	350 gpm	to HT***	1-RFV-62-688
CVCS	12	Charging Pump Suction	Borated Water	100°F	220 psig	25 gpm	to PRT	1-RFV-62-1221, -1222
WDS	14	Downstream of 1-FCV-77-127	Liquid Radwaste	140°F	50 psig	1 gpm	upstream of 1-FCV-77-127	1-RFV-77-2875
SIS	15	Penetration X-30	Borated Water	235°F	2480 psig	20 gpm	to PRT	1-RFV-63-28

**Water-steam Mixture Downstream of Valve
***Safety Class B Discharge Piping

PRT - Pressurizer Relief Tank
VCT - Volume Control Tank
HT - Holdup Tanks

Reference:
SIS = Figure 6.3-1, Sh 1 (TVA Dwg. 1-47W811-1)
RHR = Figure 2.4-107 and 5.5-4-1 (TVA Dwg. 1-47W810-1)
CVCS = Figure 9.3-15, Sh 1 (TVA Dwg. 1-47W809-1)
WDS = Figure 9.3-7 (TVA Dwg. 1-47W851-1)

TABLE 3.9-20 (Sheet 2 of 2)
UNIT 2

RELIEF VALVES IN CLASS 2 AUXILIARY SYSTEMS

System	No.	Location	Fluid	Relieving Temperature	Valve Set Pressure	Max. Required Flow Rate	Discharge	Associated Relief Valve
SIS	1	Accumulator Tanks/ N_2 Supply	Nitrogen or Borated Water	120°F	700/750 psig	1500 cfm	Atmosphere	1-RFV-63-602, -603, -604, -605
SIS	2	S. I. Pump Suction	Borated Water	100°F	220 psig	25 gpm	to PRT	1-RFV-63-511; -835
SIS	3	S. I. Pump Discharge	Borated Water	100°F	1750 psig	20 gpm	to PRT	1-RFV-63-534, -535, -536
SIS	4	Boron Injection Tank	Borated Water	180°F	2735 psig	20 gpm	to HT	1-RFV-63-577
RHR	5	Residual Pump Suction	Borated Water	350°F ** 200°F	450 psig 450 psig	480 gpm 690 gpm	to PRT to PRT	1-RFV-74-505
RHR	6	Residual Pump Discharge	Borated Water	120°F	600 psig	20 gpm	to PRT	1-RFV-63-626, -627, -637
CVCS	7	Letdown Line Orifice	Borated Water	347°F **	600 psig	227 gpm	to PRT	1-RFV-62-662
CVCS	8	Seal Water Return Line	Borated Water	250°F **	150 psig	225 gpm	to PRT	1-RFV-62-636
CVCS	9	Letdown Line	Borated Water	200°F	200 psig	200 gpm	to VCT***	1-RFV-62-675
CVCS	10	Seal Water Return Line	Borated Water	150°F	200 psig	180 gpm	to VCT***	1-RFV-62-649
CVCS	11	Volume Control Tank	Hydrogen Nitrogen or Borated Water	130°F	70 psig	350 gpm	to HT***	1-RFV-62-688
CVCS	12	Charging Pump Suction	Borated Water	100°F	220 psig	25 gpm	to PRT	1-RFV-62-1221, -1222
WDS	14	Downstream of 1-FCV-77-127	Liquid Radwaste	140°F	50 psig	1 gpm	upstream of 1-FCV-77-127	1-RFV-77-2875
SIS	15	Penetration X-30	Borated Water	235°F	2480 psig	20 gpm	to PRT	1-RFV-63-28

**Water-steam Mixture Downstream of Valve
***Safety Class B Discharge Piping

PRT - Pressurizer Relief Tank
VCT - Volume Control Tank
HT - Holdup Tanks

Reference:
SIS = Figure 6.3-1, Sh 1 (TVA Dwg. 1-47W811-1, 2-47W811-1)
RHR = Figure 2.4-107 and 5.5-4-1 (TVA Dwg. 1-47W810-1, 2-47W810-1))
CVCS = Figure 9.3-15, Sh 1 (TVA Dwg. 1-47W809-1, 2-47W809-1)
WDS = Figure 9.3-7 (TVA Dwg. 1-47W851-1, 2-47W851-1)

TABLE 3.9-21

SUPPORT DESIGN ALLOWABLE STRESSES FOR CATEGORY I PIPING SUPPORTS

Load Condition	Supplemental Structural Steel, Welds & Structural Bolts[(2)]	Component Standard Supports W/ LCDS' Except Unistrut Clamps	Component Standard Supports W/O LCDS' Except Unistrut Clamps[(1)]
Normal & Friction	Normal AISC Allowable	Manufacturer's LCDS for Level A	Manufacturer's Allowable Catalog Value
Upset	Normal AISC Allowable X 1.33	Manufacturer's LCDS for Level B	Manufacturer's Allowable Catalog Value X 1.2
Emergency	Normal AISC Allowable X 1.5	Manufacturer's LCDS for Level C	Manufacturer's Allowable Catalog Value X 1.5
Faulted	Normal AISC Allowable X 1.5	Manufacturer's LCDS for Level D	Manufacturer's Allowable Catalog Value X 1.5
Test	Normal AISC Allowable	Manufacturer's LCDS for Level A X 1.33	Manufacturer's Allowable Catalog Value X 1.33

Notes:

(1) The allowable loads for both U-bolts and unistrut clamps were developed based on the load testing per the requirements of ASME Section III 1974, Subsection NF including Winter 1974, Addenda.

(2) Tensile stresses do not exceed $0.9F_y$ and shear stresses do not exceed $0.9F_y/\sqrt{3}$. For compressive loads, the stress does not exceed 2/3 critical buckling.

TABLE 3.9-22
TABLE 3.9-23
TABLE 3.9-24

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TABLE 3.9-25 (Sheet 1 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Main Steam (1)	FCV-1-4	32	Air2	Globe	Main steam and containment isolation
	PCV-1-5	6	Air	Globe	SG pressure control and relief
	FCV-1-7	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-11	32	Air	Globe	Main steam and containment isolation
	PCV-1-12	6	Air	Globe	SG pressure control and relief
	FCV-1-14	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-22	32	Air	Globe	Main steam and containment isolation
	PCV-1-23	6	Air	Globe	SG pressure control and relief
	FCV-1-25	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-29	32	Air	Globe	Main steam and containment isolation
	PCV-1-30	6	Air	Globe	SG pressure control and relief
	FCV-1-32	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-51	4	Motor	Globe	AFW pump turbine operation
	FCV-1-52	4	Hydraulic	Globe	AFW pump turbine operation
	FCV-1-147	2	Air	Globe	Containment isolation
	FCV-1-148	2	Air	Globe	Containment isolation
	FCV-1-149	2	Air	Globe	Containment isolation
	FCV-1-150	2	Air	Globe	Containment isolation
	FCV-1-181	4	Solenoid	Globe	SG blowdown isolation
	FCV-1-182	4	Solenoid	Globe	SG blowdown isolation
	FCV-1-183	4	Solenoid	Globe	SG blowdown isolation
	FCV-1-184	4	Solenoid	Globe	SG blowdown isolation
	SFV-1-512	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-513	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-514	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-515	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-516	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-517	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-518	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-519	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-520	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-521	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-522	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-523	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-524	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-525	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-526	6x10	Self-actuated	Angle-relief	Steam generator safety relief

TABLE 3.9-25 (Sheet 2 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	SFV-1-527	6x10	Self-actuated	Angle-relief	Steam generator safety relief
	SFV-1-528	6x10	Self-actuated	Angle-relief	Steam generator safety relief
	SFV-1-529	6x10	Self-actuated	Angle-relief	Steam generator safety relief
	SFV-1-530	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-531	6x10	Self-actuated	Angle relief	Steam generator safety relief
	FCV-1-15	4	Motor	Gate	AFW pump turbine operation
	FCV-1-16	4	Motor	Gate	AFW pump turbine operation
	FCV-1-17	4	Motor	Gate	AFW pump turbine operation
	FCV-1-18	4	Motor	Gate	AFW pump turbine operation
	CKV-1-891	4	Self-actuated	Check	Prvnt bkflw btwn stm gen
	CKV-1-892	4	Self-actuated	Check	Prvnt bkflw btwn stm gen
Feedwater (3)	FCV-3-33	16	Motor	Gate	Main feedwater isolation
	FCV-3-47	16	Motor	Gate	Main feedwater isolation
	FCV-3-87	16	Motor	Gate	Main feedwater isolation
	FCV-3-100	16	Motor	Gate	Main feedwater isolation
	FCV-3-116A	4	Motor	Gate	AFW flow path integrity
	FCV-3-116B	4	Motor	Gate	AFW flow path integrity
	PCV-3-122	4	Air	Globe	Maintain head on pump
	FCV-3-126A	4	Motor	Gate	AFW flow path integrity
	FCV-3-126B	4	Motor	Gate	AFW flow path integrity
	PCV-3-132	4	Air	Globe	Maintain head on pump
	FCV-3-136A	6	Motor	Gate	AFW flow path integrity
	FCV-3-136B	6	Motor	Gate	AFW flow path integrity
	LCV-3-148	4	Air	Globe	Steam generator level control
	LCV-3-148A	2	Air	Globe	Steam generator level control
	LCV-3-156	4	Air	Globe	Steam generator level control
	LCV-3-156A	2	Air	Globe	Steam generator level control
	LCV-3-164	4	Air	Globe	Steam generator level control
	LCV-3-164A	2	Air	Globe	Steam generator level control
	LCV-3-171	4	Air	Globe	Steam generator level control
	LCV-3-171A	2	Air	Globe	Steam generator level control
	LCV-3-172	3	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-173	3	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-174	3	Air/Nitrogen	Globe	Steam generator level control

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TABLE 3.9-25 (Sheet 3 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	LCV-3-175	3	Air/Nitrogen	Globe	Steam generator level control
	FCV-3-179A	6	Motor	Gate	AFW flow path integrity
	FCV-3-179B	6	Motor	Gate	AFW flow path integrity
	FCV-3-236	6	Air	Globe	Main feedwater isolation
	FCV-3-239	6	Air	Globe	Main feedwater isolation
	FCV-3-242	6	Air	Globe	Main feedwater isolation
	FCV-3-245	6	Air	Globe	Main feedwater isolation
	FCV-3-355	2	Air	Globe	Recirc. Isolation
	FCV-3-359	2	Air	Globe	Recirc. Isolation
	CKV-3-508	16	Self-actuated	Check	Pipe break protection
	CKV-3-509	16	Self-actuated	Check	Pipe break protection
	CKV-3-510	16	Self-actuated	Check	Pipe break protection
	CKV-3-511	16	Self-actuated	Check	Pipe break protection
	CKV-3-638	6	Self-actuated	Check	Pipe break protection
	CKV-3-644	6	Self-actuated	Check	Pipe break protection
	CKV-3-645	6	Self-actuated	Check	Pipe break protection
	CKV-3-652	6	Self-actuated	Check	Pipe break protection
	CKV-3-655	6	Self-actuated	Check	Pipe break protection
	CKV-3-656	6	Self-actuated	Check	Pipe break protection
	CKV-3-669	6	Self-actuated	Check	Pipe break protection
	CKV-3-670	6	Self-actuated	Check	Pipe break protection
	CKV-3-678	6	Self-actuated	Check	Pipe break protection
	CKV-3-679	6	Self-actuated	Check	Pipe break protection
	CKV-3-805	6	Self-actuated	Check	ERCW flowpath integrity
	CKV-3-806	6	Self-actuated	Check	ERCW flowpath integrity
	CKV-3-810	10	Self-actuated	Check	ERCW flowpath integrity
	CKV-3-830	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-831	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-832	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-833	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-861	4	Self-actuated	Check	Containment Isolation
	CKV-3-862	4	Self-actuated	Check	Containment Isolation
	CKV-3-871	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-872	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-873	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity

TABLE 3.9-25 (Sheet 4 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	CKV-3-874	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-921	4	Self-actuated	Check	Containment Isolation
	CKV-3-922	4	Self-actuated	Check	Containment Isolation
Fuel Oil (18)	CKV-18-556A	1.500	Self-actuated	Check	7-DAY MDP #1 ISOL
	CKV-18-556B	1.500	Self-actuated	Check	7-DAY MDP #1 ISOL
	CKV-18-557A	1.500	Self-actuated	Check	7-DAY MDP #2 ISOL
	CKV-18-557B	1.500	Self-actuated	Check	7-DAY MDP #2 ISOL
	CKV-18-558A	1.000	Self-actuated	Check	DAY TK #1 MDP ISOL
	CKV-18-558B	1.000	Self-actuated	Check	DAY TK #1 MDP ISOL
	CKV-18-559A	1.000	Self-actuated	Check	DAY TK #1 EDP ISOL
	CKV-18-559B	1.000	Self-actuated	Check	DAY TK #1 EDP ISOL
	RFV-18-560A	0.375	Self-actuated	Relief	DAY TK #1 MDP DISC
	RFV-18-560B	0.375	Self-actuated	Relief	DAY TK #1 MDP DISC
	CKV-18-563A	1.000	Self-actuated	Check	DAY TK #1 MDP DISC
	CKV-18-563B	1.000	Self-actuated	Check	DAY TK #1 MDP DISC
	CKV-18-565A	1.000	Self-actuated	Check	DAY TK #2 MDP ISOL
	CKV-18-565B	1.000	Self-actuated	Check	DAY TK #2 MDP ISOL
	CKV-18-566A	1.000	Self-actuated	Check	DAY TK #2 EDP ISOL
	CKV-18-566B	1.000	Self-actuated	Check	DAY TK #2 EDP ISOL
	RFV-18-567A	0.375	Self-actuated	Relief	DAY TK #2 MDP DISC
	RFV-18-567B	0.375	Self-actuated	Relief	DAY TK #2 MDP DISC
	CKV-18-570A	1.000	Self-actuated	Check	DAY TK #2 MDP DISC
	CKV-18-570B	1.000	Self-actuated	Check	DAY TK #2 MDP DISC
High-Pressure	FCV-26-240	4	Motor	Gate	Containment isolation
Fire Protection	FCV-26-243	4	Motor	Gate	Containment isolation
System (26)	CKV-26-1260	4	Self-actuated	Check	Containment isolation
	CKV-26-1296	4	Self-actuated	Check	Containment isolation

TABLE 3.9-25 (Sheet 5 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Ventilation (30)	FCV-30-2	24	Air	Butterfly	Isolation Valve
	FCV-30-5	24	Air	Butterfly	Isolation Valve
	FCV-30-7	24	Air	Butterfly	Containment Isolation
	FCV-30-8	24	Air	Butterfly	Containment isolation
	FCV-30-9	24	Air	Butterfly	Containment isolation
	FCV-30-10	24	Air	Butterfly	Containment isolation
	FCV-30-12	24	Air	Butterfly	Isolation Valve
	FCV-30-14	24	Air	Butterfly	Containment isolation
	FCV-30-15	24	Air	Butterfly	Containment isolation
	FCV-30-16	24	Air	Butterfly	Containment isolation
	FCV-30-17	24	Air	Butterfly	Containment isolation
	FCV-30-19	10	Air	Butterfly	Containment isolation
	FCV-30-20	10	Air	Butterfly	Containment isolation
	FCV-30-37	8	Air	Butterfly	Containment isolation
	FCV-30-40	8	Air	Butterfly	Containment isolation
	FCV-30-50	24	Air	Butterfly	Containment isolation
	FCV-30-51	24	Air	Butterfly	Containment isolation
	FCV-30-52	24	Air	Butterfly	Containment isolation
	FCV-30-53	24	Air	Butterfly	Containment isolation
	FCV-30-54	24	Air	Butterfly	Isolation Valve
	FCV-30-56	24	Air	Butterfly	Containment isolation
	FCV-30-57	24	Air	Butterfly	Containment isolation
	FCV-30-58	10	Air	Butterfly	Containment isolation
	FCV-30-59	10	Air	Butterfly	Containment isolation
	FCV-30-61	24	Air	Butterfly	Isolation Valve
	FCV-30-62	24	Air	Butterfly	Isolation Valve
	FSV-30-134*	1	Solenoid	Solenoid	Containment isolation
	FSV-30-135*	1	Solenoid	Solenoid	Containment isolation
Air-Conditioning System (31)	FCV-31-305	2	Air	Gate	Containment Isolation
	FCV-31-306	2	Air	Gate	Containment Isolation
	FCV-31-308	2	Air	Gate	Containment Isolation
	FCV-31-309	2	Air	Gate	Containment Isolation
	FCV-31-326	2	Air	Gate	Containment Isolation
	FCV-31-327	2	Air	Gate	Containment Isolation

TABLE 3.9-25 (Sheet 6 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	FCV-31-329	2	Air	Gate	Containment Isolation
	FCV-31-330	2	Air	Gate	Containment Isolation
	CKV-31-3378	½	Self Actuated	Check	Containment Isolation
	CKV-31-3392	½	Self Actuated	Check	Containment Isolation
	CKV-31-3407	½	Self Actuated	Check	Containment Isolation
	CKV-31-3421	½	Self Actuated	Check	Containment Isolation
Control Air	0-FCV-32-70	1	Motor	Ball	Aux. Dryer Purge Control
System (32)	0-CKV-32-70A	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-70B	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-70C	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-70D	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-FCV-32-71	1	Motor	Ball	Aux. Dryer Purge Control
	0-FCV-32-72	1	Motor	Ball	Aux. Dryer Purge Control
	0-FCV-32-73	1	Motor	Ball	Aux. Dryer Purge Control
	1-FCV-32-80	2	Air	Globe	Containment Isolation
	0-FCV-32-82	2	Air	Globe	Control Air Normal Flow Isol
	0-FCV-32-85	2	Air	Globe	Control Air Normal Flow Isol
	0-FCV-32-94	1	Motor	Ball	Aux. Dryer Purge Control
	0-CKV-32-94A	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-94B	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-94C	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-94D	1	Self Actuated	Check	Air Dryer Purge Check Valve
	0-FCV-32-95	1	Motor	Ball	Aux. Dryer Purge Control
	0-FCV-32-96	1	Motor	Ball	Aux. Dryer Purge Control
	0-FCV-32-97	1	Motor	Ball	Aux. Dryer Purge Control
	1-FCV-32-102	2	Air	Globe	Containment isolation
	1-FCV-32-110	2	Air	Globe	Containment isolation
	0-CKV-32-240	2	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-256	2	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-264	2	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-279	2	Self Actuated	Check	Air Dryer Purge Check Valve
	1-CKV-32-293	2	Self Actuated	Check	Containment isolation
	1-CKV-32-303	2	Self Actuated	Check	Containment isolation
	1-CKV-32-313	2	Self Actuated	Check	Containment isolation

TABLE 3.9-25 (Sheet 7 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Sampling and Water Quality (43)	FCV-43-2	3/8	Solenoid	Globe	Containment isolation
	FCV-43-3	3/8	Air	Globe	Containment isolation
	FCV-43-11	3/8	Solenoid	Globe	Containment isolation
	FCV-43-12	3/8	Air	Globe	Containment isolation
	FCV-43-22	3/8	Solenoid	Globe	Containment isolation
	FCV-43-23	1-1/3	Air	Gate	Containment isolation
	FCV-43-34	3/8	Solenoid	Globe	Containment isolation
	FCV-43-35	3/8	Air	Globe	Containment isolation
	FCV-43-54D	3/8	Air	Globe	Steam Gen 1 Drum/Bldn Sample Isol
	FCV-43-55	3/8	Air	Globe	Containment isolation
	FCV-43-56D	3/8	Air	Globe	Steam Gen 2 Drum/Bldn Sample Isol
	FCV-43-58	3/8	Air	Globe	Containment isolation
	FCV-43-59D	3/8	Air	Globe	Steam Gen 3 Drum/Bldn Sample Isol
	FCV-43-61	3/8	Air	Globe	Containment isolation
	FCV-43-63D	3/8	Air	Globe	Steam Gen 4 Drum/Bldn Sample Isol
	FCV-43-64	3/8	Air	Globe	Containment isolation
	FCV-43-75	3/8	Solenoid	Globe	Containment isolation
	FCV-43-77	3/8	Air	Globe	Containment isolation
	1-PCV-43-200A	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	1-PCV-43-200B	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	FCV-43-201	3/8	Solenoid	Globe	Containment isolation
	FCV-43-202	3/8	Solenoid	Globe	Containment isolation
	FCV-43-207	3/8	Solenoid	Globe	Containment isolation
	FCV-43-208	3/8	Solenoid	Globe	Containment isolation
	1-PCV-43-210A	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	1-PCV-43-210B	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	FCV-43-433	3/8	Solenoid	Globe	Containment isolation
	FCV-43-434	3/8	Solenoid	Globe	Containment isolation
	FCV-43-435	3/8	Solenoid	Globe	Containment isolation
	FCV-43-436	3/8	Solenoid	Globe	Containment isolation
	1-CKV-43-834	1/2	Self Actuated	Check	Cntmt Isolation PASS Waste Holdup Tank
	1-CKV-43-841	1/2	Self Actuated	Check	Cntmt Isolation PASS Waste Holdup Tank
	1-CKV-43-883	1/2	Self Actuated	Check	Cntmt Isolation PASS Cntmt Air Return
	1-CKV-43-884	1/2	Self Actuated	Check	Cntmt Isolation PASS Cntmt Air Return

TABLE 3.9-25 (Sheet 8 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	1-PREG-43-1470A-A	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
	1-PREG-43-1470B-A	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
	1-PREG-43-1471A-B	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
	1-PREG-43-1471B-B	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
Ice Condenser (61)	CKV-61-533	3/8	Self Actuated	Check	Containment isolation
	CKV-61-680	3/8	Self Actuated	Check	Containment isolation
	CKV-61-692	3/8	Self Actuated	Check	Containment isolation
	CKV-61-745	3/8	Self Actuated	Check	Containment isolation
	FCV-61-96	2	Air	Diaphragm	Containment isolation
	FCV-61-97	2	Air	Diaphragm	Containment isolation
	FCV-61-110	2	Air	Diaphragm	Containment isolation
	FCV-61-122	2	Air	Diaphragm	Containment isolation
	FCV-61-191	4	Air	Diaphragm	Containment isolation
	FCV-61-192	4	Air	Diaphragm	Containment isolation
	FCV-61-193	4	Air	Diaphragm	Containment isolation
	FCV-61-194	4	Air	Diaphragm	Containment isolation
Safety Injection System (63)	1-RFV-63-28	3/4	Self Actuated	Relief	Penetration X-30 pressure relief containment isolation
Emergency Gas Treatment	1-FCV-65-8	8	Air	Butterfly	EGTS Tr A Unit 1 Suction Damper
System (65)	1-FCV-65-10	24	Air	Butterfly	EGTS Tr A Unit 1 Suction Damper
	0-FCV-65-24	8	Air	Butterfly	EGTS Tr A Fan Isolation Damper
	0-FCV-65-28A	8	Air	Butterfly	EGTS Tr A Decay Cooling Damper
	0-FCV-65-28B	8	Air	Gate	EGTS Tr A Decay Cooling Damper
	1-FCV-65-30	24	Air	Butterfly	EGTS Tr B Unit 1 Suction Damper
	0-FCV-65-43	8	Air	Butterfly	EGTS Tr B Fan Isolation Damper
	0-FCV-65-47A	8	Air	Gate	EGTS Tr B Decay Cooling Damper
	0-FCV-65-47B	8	Air	Gate	EGTS Tr B Decay Cooling Damper
	1-FCV-65-51	8	Air	Butterfly	EGTS Tr B Unit 1 Suction Damper
	1-FCV-65-52	14	Air	Butterfly	Cntmt Annulus Vacuum Fan Isolation Damper
	1-FCV-65-53	14	Air	Butterfly	Cntmt Annulus Vacuum Fan Isolation

TABLE 3.9-25 (Sheet 9 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
					Damper
	1-PCV-65-81	16	Air	Butterfly	Shield Bldg & Cntmt Annulus Isol Damper
	1-PCV-65-83	16	Air	Butterfly	Shield Bldg & Cntmt Annulus Isol Damper
	1-PCV-65-86	16	Air	Butterfly	EGTS Cntmt Annulus Isolation Damper
	1-PCV-65-87	16	Air	Butterfly	EGTS Cntmt Annulus Isolation Damper
Essential Raw Cooling Water (67) (See Note **)	FCV-67-9A	4	Motor	Ball	ERCW strainer
	FCV-67-9B	4	Motor	Ball	ERCW strainer
	FCV-67-10A	4	Motor	Ball	ERCW strainer
	FCV-67-10B	4	Motor	Ball	ERCW strainer
	FCV-67-66	8	Motor	Butterfly	Diesel Generator Cooling
	FCV-67-67	8	Motor	Butterfly	Diesel Generator Cooling
	1-FCV-67-83	6	Motor	Butterfly	Containment isolation
	1-FCV-67-87	6	Motor	Butterfly	Containment isolation
	1-FCV-67-88	6	Motor	Butterfly	Containment isolation
	1-FCV-67-89	6	Motor	Butterfly	Containment isolation
	1-FCV-67-91	6	Motor	Butterfly	Containment isolation
	1-FCV-67-95	6	Motor	Butterfly	Containment isolation
	1-FCV-67-96	6	Motor	Butterfly	Containment isolation
	1-FCV-67-97	6	Motor	Butterfly	Containment isolation
	1-FCV-67-99	6	Motor	Butterfly	Containment isolation
	1-FCV-67-103	6	Motor	Butterfly	Containment isolation
	1-FCV-67-104	6	Motor	Butterfly	Containment isolation
	1-FCV-67-105	6	Motor	Butterfly	Containment isolation
	1-FCV-67-107	6	Motor	Butterfly	Containment isolation
	1-FCV-67-111	6	Motor	Butterfly	Containment isolation
	1-FCV-67-112	6	Motor	Butterfly	Containment isolation
	1-FCV-67-113	6	Motor	Butterfly	Containment isolation
	1-FCV-67-123	18	Motor	Butterfly	Containment spray cooling
	1-FCV-67-124	18	Motor	Butterfly	Containment spray cooling
	1-FCV-67-125	18	Motor	Butterfly	Containment spray cooling
	1-FCV-67-126	18	Motor	Butterfly	Containment spray cooling
	1-FCV-67-130	2	Motor	Plug	Containment isolation
	1-FCV-67-131	2	Motor	Plug	Containment isolation
	1-FCV-67-133	2	Motor	Plug	Containment isolation

TABLE 3.9-25 (Sheet 10 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	1-FCV-67-134	2	Motor	Plug	Containment isolation
	1-FCV-67-138	2	Motor	Plug	Containment isolation
	1-FCV-67-139	2	Motor	Plug	Containment isolation
	1-FCV-67-141	2	Motor	Plug	Containment isolation
	1-FCV-67-142	2	Motor	Plug	Containment isolation
	FCV-67-143	12	Motor	Globe	ERCW flow path integrity
	0-FCV-67-144	16	Motor	Globe	ERCW flow path integrity
	1-FCV-67-146	24	Motor	Butterfly	ERCW from CCS Heat Exchanger A
	2-FCV-67-146	24	Motor	Butterfly	ERCW from CCS Heat Exchanger B
	0-FCV-67-152	24	Motor	Butterfly	CCS flow path integrity
	TCV-67-158	4	Self-Actuated	Globe	SBR temperature control
	1-FCV-67-162	2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-164	2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-176	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-182	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-184	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-186	1-½	Air	Globe	ERCW to pump room coolers
	0-FCV-67-205	4	Motor	Butterfly	Nonessential equipment isolation
	0-FCV-67-208	4	Motor	Butterfly	Nonessential Equipment isolation
	1-FCV-67-213	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-215	1-½	Air	Globe	ERCW to pump room coolers
	2-FCV-67-217	2	Air	Globe	ERCW to pump room coolers
	2-FCV-67-219	2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-295	2	Motor	Plug	Containment isolation
	1-FCV-67-296	2	Motor	Plug	Containment isolation
	1-FCV-67-297	2	Motor	Plug	Containment isolation
	1-FCV-67-298	2	Motor	Plug	Containment isolation
	2-FCV-67-336	1	Air	Globe	ERCW to pump room coolers
	2-FCV-67-338	1	Air	Globe	ERCW to pump room coolers
	1-FCV-67-342	2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-344	2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-346	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-348	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-350	1-½	Air	Globe	ERCW to pump room coolers
	1-FCV-67-352	1-½	Air	Globe	ERCW to pump room coolers

TABLE 3.9-25 (Sheet 11 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	FCV-67-354	1-½	Air	Globe	ERCW to pump room coolers
	FCV-67-356	1-½	Air	Globe	ERCW to pump room coolers
	0-CKV-67-503A	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503B	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503C	20	Self-actuated	Check	ERCW flow path integrity
	0-TCV-67-1050-A	3	Self-actuated	Globe	EBR condenser temperature control
	0-TCV-67-1051-A	3	Self-actuated	Globe	MCR condenser temperature control
	0-TCV-67-1052-B	3	Self-actuated	Globe	EBR condenser temperature control
	0-TCV-67-1053-B	3	Self-actuated	Globe	MCR condenser temperature control
	CKV-67-508A	8	Self-actuated	Check	ERCW flow path integrity
	CKV-67-508B	8	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-512A	10	Self-Actuated	Check	ERCW flow path integrity
	0-CKV-67-517A	10	Self-Actuated	Check	ERCW flow path integrity
	1-CKV-67-940A	3	Self-Actuated	Check	ERCW flow path integrity
	2-CKV-67-935B	3	Self-Actuated	Check	ERCW flow path integrity
	0-CKV-67-503D	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503E	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503F	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503G	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503H	20	Self-actuated	Check	ERCW flow path integrity
	1-CKV-67-575A	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-575B	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-575C	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-575D	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-580A	2	Self-actuated	Check	Containment Isolation
	1-CKV-67-580B	2	Self-actuated	Check	Containment Isolation
	1-CKV-67-580C	2	Self-actuated	Check	Containment Isolation
	1-CKV-67-580D	2	Self-actuated	Check	Containment Isolation
	1-CKV-67-585A	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-585B	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-585C	1/2	Self-actuated	Check	Containment Isolation
	1-CKV-67-585D	1/2	Self-actuated	Check	Containment Isolation
	1-RFV-67-1060A	3/4 x 1	Self-actuated	Relief	Containment Isolation
	1-RFV-67-1060B	3/4 x 1	Self-actuated	Relief	Containment Isolation
	1-RFV-67-1060C	3/4 x 1	Self-actuated	Relief	Containment Isolation

TABLE 3.9-25 (Sheet 12 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	1-RFV-67-1060D	3/4 x 1	Self-actuated	Relief	Containment Isolation
	0-CKV-67-502A	2	Self-actuated	Check	Air release
	0-CKV-67-502B	2	Self-actuated	Check	Air release
	0-CKV-67-502C	2	Self-actuated	Check	Air release
	0-CKV-67-502D	2	Self-actuated	Check	Air release
	0-CKV-67-502E	2	Self-actuated	Check	Air release
	0-CKV-67-502F	2	Self-actuated	Check	Air release
	0-CKV-67-502G	2	Self-actuated	Check	Air release
	0-CKV-67-502H	2	Self-actuated	Check	Air release
	0-FSV-67-1221-A	1	Solenoid	Globe	ERCW Supply
	0-PCV-67-1222	1	Self	Regulating	Press Cntr for ERCW to Aux Air Comp A
	0-TCV-67-1222A	1/2	Self	Regulating	Throttles ERCW to Aux Air Comp A cyl jacket
	0-TCV-67-1222B	3/4	Self	Regulating	Throttles ERCW to Aux Air Comp A cyl jacket
	0-FSV-67-1223-B	1	Solenoid	Globe	ERCW Supply
	0-PCV-67-1224	1	Self	Regulating	Press Cntr for ERCW to Aux Air Comp B
	0-TCV-67-1224A	1/2	Self	Regulating	Throttles ERCW to Aux Air Comp B cyl jacket
	0-TCV-67-1224B	3/4	Self	Regulating	Throttles ERCW to Aux Air Comp B cyl jacket
Component Cooling Water (70)	1-FCV-70-85	6	Air	Butterfly	Containment Isolation
	1-FCV-70-87	3	Motor	Gate	Containment Isolation
	1-FCV-70-89	6	Motor	Butterfly	Containment Isolation
	1-FCV-70-90	3	Motor	Gate	Containment Isolation
	1-FCV-70-92	6	Motor	Butterfly	Containment Isolation
	1-FCV-70-100	6	Motor	Butterfly	Containment Isolation
	1-FCV-70-133	3	Motor	Gate	RCP Thermal Barrier Isolation
	1-FCV-70-134	3	Motor	Gate	Containment Isolation
	1-FCV-70-140	6	Motor	Butterfly	Containment Isolation
	1-FCV-70-143	6	Motor	Butterfly	Containment Isolation
	1, 2-FCV-70-153	18	Motor	Butterfly	RHR Hx Isolation
	1-FCV-70-156	18	Motor	Butterfly	RHR Hx Isolation
	0-FCV-70-194	20	Motor	Butterfly	Spent Fuel Pit Hx Isolation

TABLE 3.9-25 (Sheet 13 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	0-FCV-70-197	20	Motor	Butterfly	Spent fuel pit Hx isolation
	0-CKV-70-504	16	Self-actuated	Check	Prevent backflow through CCP
	1,2-CKV-70-504A	16	Self-actuated	Check	Prevent backflow through CCP
	1,2-CKV-70-504B	16	Self-actuated	Check	Prevent backflow through CCP
	1-CKV-70-679	3	Self-actuated	Check	Containment isolation CCP
	1-CKV-70-698	3/4 x 1	Self-actuated	Relief	Containment isolation CCP
	1-CKV-70-790	3/4 x 1	Self-actuated	Relief	Containment isolation
	1-FCV-70-183	3	Motor	Gate	Sample Hx Isolation
	1-FCV-70-215	3	Motor	Gate	Sample Hx Isolation
	1-FCV-70-66	2	Air	Angle	CCS surge tank isolation
	2-FCV-70-66	2	Air	Angle	CCS surge tank isolation
	1-RFV-70-538	3x4	Self-actuated	Relief	Surge tank 1A/1B relief
	1-CKV-70-681-A,B,C, D	1-1/2	Self -actuated	Check	Overpressurization protection
	1-CKV-70-682-A,B,C, D	1-1/2	Self -actuated	Check	Overpressurization protection
	1-CKV-70-687	3/4 x 1	Self -actuated	Relief	RCP thermal barrier isolation
	1-RFV-70-703	3x4	Self -actuated	Relief	Excess letdown HX relief
	1-RFV-70-835	3/4x1	Self -actuated	Relief	RCP thermal barrier supply relief
Containment Spray System (72)	1-FCV-72-13-B	2	Motor	Globe	Min-flow pump recirculation control valve
	1-FCV-72-34-A	2	Motor	Globe	Min-flow pump recirculation control valve
	1-RFV-72-40	3/4 x 1	Self-Actuated	Relief	Bonnet Overpressure Relief Valve for 1-FCV-72-40
	1-RFV-72-41	3/4 x 1	Self-Actuated	Relief	Bonnet Overpressure Relief Valve for 1-FCV-72-41
Primary Water System (81)	1-FCV-81-12	3	Air	Gate	Primary water to RCS PRT
	1-CKV-81-502	3	Self-actuated	Gate	Primary water to RCS PRT
Standby Diesel Generators (82)	1-FCV-82-160	1.500	Air	Diaph	Flow Cont 1A1 TK A
	1-FSV-82-160-A	0.370	Elec	Sol V	Air ST SOL 1A1 (A)
	1-FCV-82-161	1.500	Air	Diaph	Flow Cont 1A2 TK A
	1-FSV-82-161-A	0.370	Elec	Sol V	Air ST SOL 1A2 (A)
	1-PCV-82-162A	2.000	Air	Diaph	PR RED V 1A1 TK A

TABLE 3.9-25 (Sheet 14 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	1-PREG-82-162B	0.250	Self	Diaph	PR REG V 1A1 TK A
	1-PCV-82-163A	2.000	Air	Diaph	PR RED V 1A2 TK A
	1-PREG-82-163B	0.250	Self	Diaph	PR REG V 1A2 TK A
	1-FCV-82-170	1.500	Air	Diaph	FLOW CONT 1A1 TK B
	1-FSV-82-170-A	0.370	Elec	Sol V	AIR ST SOL 1A1 (B)
	1-FCV-82-171	1.500	Air	Diaph	FLOW CONT 1A2 TK B
	1-FSV-82-171-A	0.370	Elec	Sol V	AIR ST SOL 1A2 (B)
	1-PCV-82-172A	2.000	Air	Diaph	PR RED V 1A1 TK B
	1-PREG-82-172B	0.250	Self	Diaph	PR REG V 1A1 TK B
	1-PCV-82-173A	2.000	Air	Diaph	PR RED V 1A2 TK B
	1-PREG-82-173B	0.250	Self	Diaph	PR REG V 1A2 TK B
	1-FCV-82-190	1.500	Air	Diaph	FLOW CONT 1B1 TK A
	1-FSV-82-190-A	0.370	Elec	Sol V	AIR ST SOL 1B1(A)
	1-FCV-82-191	1.500	Air	Diaph	FLOW CONT 1B2 TK A
	1-FSV-82-191-A	0.370	Elec	Sol V	AIR ST SOL 1B2 (A)
	1-PCV-82-192A	2.000	Air	Diaph	PR REG V 1B1 TK A
	1-PREG-82-192B	0.250	Self	Diaph	PR REG V 1B1 TK A
	1-PCV-82-193A	2.000	Air	Diaph	PR RED V 1B2 TK A
	1-PREG-82-193B	0.250	Self	Diaph	PR REG V 1B2 TK A
	1-FCV-82-200	1.500	Air	Diaph	FLOW CONT 1B1 TK B
	1-FSV-82-200-A	0.370	Elec	Sol V	AIR ST SOL 1B1 (B)
	1-FCV-82-201	1.500	Air	Diaph	FLOW CONT 1B2 TK B
	1-FSV-82-201-A	0.370	Elec	Sol V	AIR ST SOL 1B2 (B)
	1-PCV-82-202A	2.000	Air	Diaph	PR RED V 1B1 TK B
	1-PREG-82-202B	0.250	Self	Diaph	PR REG V 1B1 TK B
	1-PCV-82-203A	2.000	Air	Diaph	PR RED V 1B2 TK B
	1-PREG-82-203B	0.250	Self	Diaph	PR REG V 1B2 TK B
	2-FCV-82-220	1.500	Air	Diaph	FLOW CONT 2A1 TK A
	2-FSV-82-220-A	0.370	Elec	Sol V	AIR ST SOL 2A1 (A)
	2-FCV-82-221	1.500	Air	Diaph	FLOW CONT 2A2 TK A
	2-FSV-82-221-A	0.370	Elec	Sol V	AIR ST SOL 2A2 (A)
	2-PCV-82-222A	2.000	Air	Diaph	PR RED V 2A1 TK A
	2-PREG-82-222B	0.250	Self	Diaph	PR REG V 2A1 TK A
	2-PCV-82-223A	2.000	Air	Diaph	PR RED V 2A2 TK A
	2-PREG-82-223B	0.250	Self	Diaph	PR REG V 2A2 TK A

TABLE 3.9-25 (Sheet 15 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	2-FCV-82-230	1.500	Air	Diaph	FLOW CONT 2A1 TK B
	2-FSV-82-230-A	0.370	Elec	Sol V	AIR ST SOL 2A1 (B)
	2-FCV-82-231	1.500	Air	Diaph	FLOW CONT 2A2 TK B
	2-FSV-82-231-A	0.370	Elec	Sol V	AIR ST SOL 2A2 (B)
	2-PCV-82-232A	2.000	Air	Diaph	PR RED V 2A1 TK B
	2-PREG-82-232B	0.250	Self	Diaph	PR REG V 2A1 TK B
	2-PCV-82-233A	2.000	Air	Diaph	PR RED V 2A2 TK B
	2-PREG-82-233B	0.250	Self	Diaph	PR REG V 2A2 TK B
	2-FCV-82-250	1.500	Air	Diaph	FLOW CONT 2B1 TK A
	2-FSV-82-250-A	0.370	Elec	Sol V	AIR ST SOL 2B1(A)
	2-FCV-82-251	1.500	Air	Diaph	FLOW CONT 2B2 TK A
	2-FSV-82-251-A	0.370	Elec	Sol V	AIR ST SOL 2B2 (A)
	2-PCV-82-252A	2.000	Air	Diaph	PR RED V 2B1 TK A
	2-PREG-82-252B	0.250	Self	Diaph	PR REG V 2B1 TK A
	2-PCV-82-253A	2.000	Air	Diaph	PR RED V 2B2 TK A
	2-PREG-82-253B	0.250	Self	Diaph	PR REG V 2B2 TK A
	2-FCV-82-260	1.500	Air	Diaph	FLOW CONT 2B1 TK B
	2-FSV-82-260-A	0.370	Elec	Sol V	AIR ST SOL 2B1 (B)
	2-FCV-82-261	1.500	Air	Diaph	FLOW CONT 2B2 TK B
	2-FSV-82-261-A	0.370	Elec	Sol V	AIR ST SOL 2B2 (B)
	2-PCV-82-262A	2.000	Air	Diaph	PR RED V 2B1 TK B
	2-PREG-82-262B	0.250	Self	Diaph	PR REG V 2B1 TK B
	2-PCV-82-263A	2.000	Air	Diaph	PR RED V 2B2 TK B
	2-PREG-82-263B	0.250	Self	Diaph	PR REG V 2B2 TK B
	1,2-CKV-82-502A1-A	0.750		Check	TK SUP A1 (A)
	1,2-CKV-82-502B1-B	0.750		Check	TK SUP B1 (A)
	1,2-CKV-82-505A1-A	0.750		Check	TK SUP A1 (B)
	1,2-CKV-82-505B1-B	0.750		Check	TK SUP B1 (B)
	1,2-CKV-82-509A1-A	0.750		Check	CROSS CONN CK A1
	1,2-CKV-82-509B1-B	0.750		Check	CROSS CONN CK B1
	1,2-CKV-82-523A1-A	0.750		Check	RELAY CK A1 (A)
	1,2-CKV-82-523B1-B	0.750		Check	RELAY CK B1 (A)
	1,2-SPV-82-524A1-A	0.375		Check	SLIDE VLV A1
	1,2-SPV-82-524B1-B	0.375		Check	SLIDE VLV B1
	1,2-CKV-82-531A1-A	0.750		Check	RELAY CK A1 (B)

TABLE 3.9-25 (Sheet 16 of 16)
UNIT 1

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	1,2-CKV-82-531B1-B	0.750		Check	RELAY CK B1 (B)
	1,2-CKV-82-536A2-A	0.750		Check	TK SUP A2 (A)
	1,2-CKV-82-536B2-B	0.750		Check	TK SUP B2 (A)
	1,2-CKV-82-539A2-A	0.750		Check	TK SUP A2 (B)
	1,2-CKV-82-539B2-B	0.750		Check	TK SUP B2 (B)
	1,2-CKV-82-543A2-A	0.750		Check	CROSS CONN CK A2
	1,2-CKV-82-543B2-B	0.750		Check	CROSS CONN CK B2
	1,2-CKV-82-557A2-A	0.750		Check	RELAY CK A2 (A)
	1,2-CKV-82-557B2-B	0.750		Check	RELAY CK B2 (A)
	1,2-SPV-82-558A2-A	0.375		Check	SLIDE VLV A2
	1,2-SPV-82-558B2-B	0.375		Check	SLIDE VLV B2
	1,2-CKV-82-565A2-A	0.750		Check	RELAY CK A2 (B)
	1,2-CKV-82-565B2-B	0.750		Check	RELAY CK B2 (B)
Radiation Monitoring (90)	1-FCV-90-107	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-108	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-109	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-110	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-111	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-113	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-114	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-115	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-116	1-1/2	Air	Globe	Containment Isolation
	1-FCV-90-117	1-1/2	Air	Globe	Containment Isolation

*Unit 2 valves were deleted under ECN 5663
**Valves not prefixed by 1-, 2- or 0-are understood to be prefixed by 1- and 2

TABLE 3.9-25 (Sheet 1 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Main Steam (1)	FCV-1-4	32	Air	Globe	Main steam and containment isolation
	PCV-1-5	6	Air	Globe	SG pressure control and relief
	FCV-1-7	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-11	32	Air	Globe	Main steam and containment isolation
	PCV-1-12	6	Air	Globe	SG pressure control and relief
	FCV-1-14	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-22	32	Air	Globe	Main steam and containment isolation
	PCV-1-23	6	Air	Globe	SG pressure control and relief
	FCV-1-25	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-29	32	Air	Globe	Main steam and containment isolation
	PCV-1-30	6	Air	Globe	SG pressure control and relief
	FCV-1-32	4	Solenoid	Globe	Blowdown and containment isolation
	FCV-1-51	4	Motor	Globe	AFW pump turbine operation
	FCV-1-52	4	Hydraulic	Globe	AFW pump turbine operation
	FCV-1-147	2	Air	Globe	Containment isolation
	FCV-1-148	2	Air	Globe	Containment isolation
	FCV-1-149	2	Air	Globe	Containment isolation
	FCV-1-150	2	Air	Globe	Containment isolation
	FCV-1-181	4	Solenoid	Globe	Blowdown isolation
	FCV-1-182	4	Solenoid	Globe	Blowdown isolation
	FCV-1-183	4	Solenoid	Globe	Blowdown isolation
	FCV-1-184	4	Solenoid	Globe	Blowdown isolation
	SFV-1-512	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-513	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-514	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-515	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-516	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-517	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-518	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-519	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-520	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-521	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-522	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-523	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-524	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-525	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-526	6x10	Self-actuated	Angle-relief	Steam generator safety relief
	SFV-1-527	6x10	Self-actuated	Angle-relief	Steam generator safety relief
	SFV-1-528	6x10	Self-actuated	Angle-relief	Steam generator safety relief
	SFV-1-529	6x10	Self-actuated	Angle-relief	Steam generator safety relief

TABLE 3.9-25 (Sheet 2 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Main Steam (1)	SFV-1-530	6x10	Self-actuated	Angle relief	Steam generator safety relief
	SFV-1-531	6x10	Self-actuated	Angle relief	Steam generator safety relief
	FCV-1-15	4	Motor	Gate	AFW pump turbine operation
	FCV-1-16	4	Motor	Gate	AFW pump turbine operation
	FCV-1-17	4	Motor	Gate	AFW pump turbine operation
	FCV-1-18	4	Motor	Gate	AFW pump turbine operation
	CKV-1-891	4	Self-actuated	Check	Prvnt bkflw btwn stm gen
	CKV-1-892	4	Self-actuated	Check	Prvnt bkflw btwn stm gen
Feedwater (3)	FCV-3-33	16	Motor	Gate	Main feedwater isolation
	FCV-3-47	16	Motor	Gate	Main feedwater isolation
	FCV-3-87	16	Motor	Gate	Main feedwater isolation
	FCV-3-100	16	Motor	Gate	Main feedwater isolation
	FCV-3-116A	4	Motor	Gate	AFW flow path integrity
	FCV-3-116B	4	Motor	Gate	AFW flow path integrity
	PCV-3-122 (Note)	4	Air/Nitrogen	Globe	Maintain head on pump
	FCV-3-126A	4	Motor	Gate	AFW flow path integrity
	FCV-3-126B	4	Motor	Gate	AFW flow path integrity
	PCV-3-132 (Note)	4	Air/Nitrogen	Globe	Maintain head on pump
	FCV-3-136A	6	Motor	Gate	AFW flow path integrity
	FCV-3-136B	6	Motor	Gate	AFW flow path integrity
	LCV-3-148 (Note)	4	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-148A (Note)	2	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-156 (Note)	4	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-156A (Note)	2	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-164 (Note)	4	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-164A (Note)	2	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-171 (Note)	4	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-171A (Note)	2	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-172	3	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-173	3	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-174	3	Air/Nitrogen	Globe	Steam generator level control
	LCV-3-175	3	Air/Nitrogen	Globe	Steam generator level control
	FCV-3-179A	6	Motor	Gate	AFW flow path integrity
	FCV-3-179B	6	Motor	Gate	AFW flow path integrity
	2-FCV-3-185	2	Air	Globe	FW pressure integrity
	2-FCV-3-186	2	Air	Globe	FW pressure integrity
	2-FCV-3-187	2	Air	Globe	FW pressure integrity

TABLE 3.9-25 (Sheet 3 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Feedwater (3)	2-FCV-3-188	2	Air	Globe	FW pressure integrity
Note: Unit 2 provides Appendix R credited nitrogen gas backup control and signal air supply source to allow continued control of the Turbine Driven Auxiliary Feedwater (TDAFW) Level Control Valves (LCVs), Motor Driven Auxiliary Feedwater (MDAFW) Level Control Valves and Motor Driven Auxiliary Feedwater Pressure Control Valves (PCVs)					
	FCV-3-236	6	Air	Globe	Main feedwater isolation
	FCV-3-239	6	Air	Globe	Main feedwater isolation
	FCV-3-242	6	Air	Globe	Main feedwater isolation
	FCV-3-245	6	Air	Globe	Main feedwater isolation
	FCV-3-355	2	Air	Globe	Recirc. Isolation
	FCV-3-359	2	Air	Globe	Recirc. Isolation
	CKV-3-508	16	Self-actuated	Check	Pipe break protection
	CKV-3-509	16	Self-actuated	Check	Pipe break protection
	CKV-3-510	16	Self-actuated	Check	Pipe break protection
	CKV-3-511	16	Self-actuated	Check	Pipe break protection
	CKV-3-638	6	Self-actuated	Check	Pipe break protection
	CKV-3-644	6	Self-actuated	Check	Pipe break protection
	CKV-3-645	6	Self-actuated	Check	Pipe break protection
	CKV-3-652	6	Self-actuated	Check	Pipe break protection
	CKV-3-655	6	Self-actuated	Check	Pipe break protection
	CKV-3-656	6	Self-actuated	Check	Pipe break protection
	CKV-3-669	6	Self-actuated	Check	Pipe break protection
	CKV-3-670	6	Self-actuated	Check	Pipe break protection
	CKV-3-678	6	Self-actuated	Check	Pipe break protection
	CKV-3-679	6	Self-actuated	Check	Pipe break protection
	CKV-3-805	6	Self-actuated	Check	ERCW flowpath integrity
	CKV-3-806	6	Self-actuated	Check	ERCW flowpath integrity
	CKV-3-810	10	Self-actuated	Check	ERCW flowpath integrity
	CKV-3-830	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-831	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-832	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-833	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-861	4	Self-actuated	Check	Containment Isolation
	CKV-3-862	4	Self-actuated	Check	Containment Isolation
	CKV-3-871	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-872	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-873	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-874	4	Self-actuated	Check	Maintain TDAFW pump flow path integrity
	CKV-3-921	4	Self-actuated	Check	Containment Isolation
	CKV-3-922	4	Self-actuated	Check	Containment Isolation

TABLE 3.9-25 (Sheet 4 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Fuel Oil (18)	CKV-18-556A	1.500	Self-actuated	Check	7-DAY MDP #1 ISOL
	CKV-18-556B	1.500	Self-actuated	Check	7-DAY MDP #1 ISOL
	CKV-18-557A	1.500	Self-actuated	Check	7-DAY MDP #2 ISOL
	CKV-18-557B	1.500	Self-actuated	Check	7-DAY MDP #2 ISOL
	CKV-18-558A	1.000	Self-actuated	Check	DAY TK #1 MDP ISOL
	CKV-18-558B	1.000	Self-actuated	Check	DAY TK #1 MDP ISOL
	CKV-18-559A	1.000	Self-actuated	Check	DAY TK #1 EDP ISOL
	CKV-18-559B	1.000	Self-actuated	Check	DAY TK #1 EDP ISOL
	RFV-18-560A	0.375	Self-actuated	Relief	DAY TK #1 MDP DISC
	RFV-18-560B	0.375	Self-actuated	Relief	DAY TK #1 MDP DISC
	CKV-18-563A	1.000	Self-actuated	Check	DAY TK #1 MDP DISC
	CKV-18-563B	1.000	Self-actuated	Check	DAY TK #1 MDP DISC
	CKV-18-565A	1.000	Self-actuated	Check	DAY TK #2 MDP ISOL
	CKV-18-565B	1.000	Self-actuated	Check	DAY TK #2 MDP ISOL
	CKV-18-566A	1.000	Self-actuated	Check	DAY TK #2 EDP ISOL
	CKV-18-566B	1.000	Self-actuated	Check	DAY TK #2 EDP ISOL
	RFV-18-567A	0.375	Self-actuated	Relief	DAY TK #2 MDP DISC
	RFV-18-567B	0.375	Self-actuated	Relief	DAY TK #2 MDP DISC
	CKV-18-570A	1.000	Self-actuated	Check	DAY TK #2 MDP DISC
	CKV-18-570B	1.000	Self-actuated	Check	DAY TK #2 MDP DISC
High-Pressure Fire Protection System (26)	FCV-26-240	4	Motor	Gate	Containment isolation
	FCV-26-243	4	Motor	Gate	Containment isolation
	CKV-26-1260	4	Self-actuated	Check	Containment isolation
	CKV-26-1296	4	Self-actuated	Check	Containment isolation
Ventilation (30)	FCV-30-2	24	Air	Butterfly	Isolation Valve
	FCV-30-5	24	Air	Butterfly	Isolation Valve
	FCV-30-7	24	Air	Butterfly	Containment Isolation
	FCV-30-8	24	Air	Butterfly	Containment isolation
	FCV-30-9	24	Air	Butterfly	Containment isolation
	FCV-30-10	24	Air	Butterfly	Containment isolation
	FCV-30-12	24	Air	Butterfly	Isolation Valve
	FCV-30-14	24	Air	Butterfly	Containment isolation
	FCV-30-15	24	Air	Butterfly	Containment isolation
	FCV-30-16	24	Air	Butterfly	Containment isolation
	FCV-30-17	24	Air	Butterfly	Containment isolation
	FCV-30-19	10	Air	Butterfly	Containment isolation
	FCV-30-20	10	Air	Butterfly	Containment isolation
	FCV-30-37	88	Air	Butterfly	Containment isolation

TABLE 3.9-25 (Sheet 5 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
	FCV-30-40	24	Air	Butterfly	Containment isolation
	FCV-30-50	24	Air	Butterfly	Containment isolation
	FCV-30-51	24	Air	Butterfly	Containment isolation
	FCV-30-52	24	Air	Butterfly	Containment isolation
	FCV-30-53	24	Air	Butterfly	Containment isolation
	FCV-30-54	24	Air	Butterfly	Isolation Valve
	FCV-30-56	24	Air	Butterfly	Containment isolation
	FCV-30-57	10	Air	Butterfly	Containment isolation
	FCV-30-58	10	Air	Butterfly	Containment isolation
	FCV-30-59	24	Air	Butterfly	Containment isolation
	FCV-30-61	24	Air	Butterfly	Isolation Valve
	FCV-30-62	24	Air	Butterfly	Isolation Valve
Air-Conditioning System (31)	FCV-31-305	2	Air	Gate	Containment Isolation
	FCV-31-306	2	Air	Gate	Containment Isolation
	FCV-31-308	2	Air	Gate	Containment Isolation
	FCV-31-309	2	Air	Gate	Containment Isolation
	FCV-31-326	2	Air	Gate	Containment Isolation
	FCV-31-327	2	Air	Gate	Containment Isolation
	FCV-31-329	2	Air	Gate	Containment Isolation
	FCV-31-330	2	Air	Gate	Containment Isolation
	CKV-31-3378	½	Self Actuated	Check	Containment Isolation
	CKV-31-3392	½	Self Actuated	Check	Containment Isolation
	CKV-31-3407	½	Self Actuated	Check	Containment Isolation
	CKV-31-3421	½	Self Actuated	Check	Containment Isolation
Control Air System (32)	FCV-32-70	1	Motor	Ball	Aux. Dryer Purge Control
	CKV-32-70A	1	Self Actuated	Check	Air Dryer Purge Check Valve
	CKV-32-70B	1	Self Actuated	Check	Air Dryer Purge Check Valve
	CKV-32-70C	1	Self Actuated	Check	Air Dryer Purge Check Valve
	CKV-32-70D	1	Self Actuated	Check	Air Dryer Purge Check Valve
	FCV-32-71	1	Motor	Ball	Aux. Dryer Purge Control
	FCV-32-72	1	Motor	Ball	Aux. Dryer Purge Control
	FCV-32-73	1	Motor	Ball	Aux. Dryer Purge Control
	FCV-32-80	2	Air	Globe	Containment Isolation
	FCV-32-81	2	Air	Globe	Containment Isolation
	FCV-32-82	2	Air	Globe	Control Air Normal Flow Isol
	FCV-32-85	2	Air	Globe	Control Air Normal Flow Isol
	FCV-32-94	2	Motor	Ball	Aux. Dryer Purge Control
	CKV-32-94A	1	Self Actuated	Check	Air Dryer Purge Check Valve

TABLE 3.9-25 (Sheet 6 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Control Air System (32)	CKV-32-94B	1	Self Actuated	Check	Air Dryer Purge Check Valve
	CKV-32-94C	1	Self Actuated	Check	Air Dryer Purge Check Valve
	CKV-32-94D	1	Self Actuated	Check	Air Dryer Purge Check Valve
	FCV-32-95	1	Motor	Ball	Aux. Dryer Purge Control
	FCV-32-96	1	Motor	Ball	Aux. Dryer Purge Control
	FCV-32-97	1	Motor	Ball	Aux. Dryer Purge Control
	1-FCV-32-102	2	Air	Globe	Containment isolation
	2-FCV-32-103	2	Air	Globe	Containment isolation
	1-FCV-32-110	2	Air	Globe	Containment isolation
	2-FCV-32-111	2	Air	Globe	Containment isolation
	0-CKV-32-240	2	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-256	2	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-264	2	Self Actuated	Check	Air Dryer Purge Check Valve
	0-CKV-32-279	2	Self Actuated	Check	Air Dryer Purge Check Valve
	1-CKV-32-293	2	Self Actuated	Check	Containment isolation
	1-CKV-32-303	2	Self Actuated	Check	Containment isolation
	1-CKV-32-313	2	Self Actuated	Check	Containment isolation
	2-CKV-32-323	2	Self Actuated	Check	Containment isolation
	2-CKV-32-333	2	Self Actuated	Check	Containment isolation
	2-CKV-32-343	2	Self Actuated	Check	Containment isolation
Sampling and Water Quality (43)	FCV-43-2	3/8	Air	Globe	Containment isolation
	FCV-43-3	3/8	Air	Globe	Containment isolation
	FCV-43-11	3/8	Air	Globe	Containment isolation
	FCV-43-12	3/8	Air	Globe	Containment isolation
	FCV-43-22	3/8	Air	Globe	Containment isolation
	FCV-43-23	3/8	Air	Globe	Containment isolation
	FCV-43-34	3/8	Air	Globe	Containment isolation
	FCV-43-35	3/8	Air	Globe	Containment isolation
	FCV-43-54D	3/8	Air	Globe	SGBD Sample Isolation
	FCV-43-56D	3/8	Air	Globe	SGBD Sample Isolation
	FCV-43-59D	3/8	Air	Globe	SGBD Sample Isolation
	FCV-43-63D	3/8	Air	Globe	SGBD Sample Isolation
	FCV-43-55	3/8	Air	Globe	Containment isolation
	FCV-43-58	3/8	Air	Globe	Containment isolation
	FCV-43-61	3/8	Air	Globe	Containment isolation
	FCV-43-64	3/8	Air	Globe	Containment isolation
	1-FCV-43-75	3/8	Solenoid	Globe	Containment isolation
	1-FCV-43-77	3/8	Air	Globe	Containment isolation
	1-PCV-43-200A	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg

TABLE 3.9-25 (Sheet 7 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Sampling and Water Quality (43)	1-PCV-43-200B	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	FCV-43-201	3/8	Solenoid	Globe	Containment isolation
	FCV-43-202	3/8	Solenoid	Globe	Containment isolation
	1-FCV-43-207B	3/8	Solenoid	Globe	Containment isolation
	1-FCV-43-208B	3/8	Solenoid	Globe	Containment isolation
	1-PCV-43-210A	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	1-PCV-43-210B	1/4	Self Actuated	Regulating	LOCA H2 Cntmt Monitor Press Reg
	FCV-43-433	3/8	Solenoid	Globe	Containment isolation
	FCV-43-434	3/8	Solenoid	Globe	Containment isolation
	1-FCV-43-435	3/8	Solenoid	Globe	Containment isolation
	1-FCV-43-436	3/8	Solenoid	Globe	Containment isolation
	1-CKV-43-834	1/2	Self Actuated	Check	Cntmt Isolation PASS Waste Holdup Tank
	1-CKV-43-841	1/2	Self Actuated	Check	Cntmt Isolation PASS Waste Holdup Tank
	1-CKV-43-883	1/2	Self Actuated	Check	Cntmt Isolation PASS Cntmt Air Return
	1-CKV-43-884	1/2	Self Actuated	Check	Cntmt Isolation PASS Cntmt Air Return
	1-PREG-43-1470A-A	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
	1-PREG-43-1470B-A	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
	1-PREG-43-1471A-B	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
	1-PREG-43-1471B-B	1/4	Self Actuated	Regulating	O2 Reagent Gas for H2 Monitor Press Reg
Ice Condenser (61)	CKV-61-533	3/8	Self Actuated	Check	Containment isolation
	CKV-61-680	3/8	Self Actuated	Check	Containment isolation
	CKV-61-692	3/8	Self Actuated	Check	Containment isolation
	CKV-61-745	3/8	Self Actuated	Check	Containment isolation
	FCV-61-96	2	Air	Diaphragm	Containment isolation
	FCV-61-97	2	Air	Diaphragm	Containment isolation
	FCV-61-110	2	Air	Diaphragm	Containment isolation
	FCV-61-122	2	Air	Diaphragm	Containment isolation
	FCV-61-191	4	Air	Diaphragm	Containment isolation
	FCV-61-192	4	Air	Diaphragm	Containment isolation
	FCV-61-193	4	Air	Diaphragm	Containment isolation
	FCV-61-194	4	Air	Diaphragm	Containment isolation
Emergency Gas Treatment System (65)	1-FCV-65-8	8	Air	Butterfly	EGTS Tr A Unit 1 Suction Damper
	1-FCV-65-10	24	Air	Butterfly	EGTS Tr A Unit 1 Suction Damper
	0-FCV-65-24	8	Air	Butterfly	EGTS Tr A Fan Isolation Damper
	0-FCV-65-28A	8	Air	Butterfly	EGTS Tr A Decay Cooling Damper
	0-FCV-65-28B	8	Air	Gate	EGTS Tr A Decay Cooling Damper
	1-FCV-65-30	24	Air	Butterfly	EGTS Tr B Unit 1 Suction Damper
	0-FCV-65-43	8	Air	Butterfly	EGTS Tr B Fan Isolation Damper
Emergency Gas	0-FCV-65-47A	8	Air	Gate	EGTS Tr B Decay Cooling Damper

TABLE 3.9-25 (Sheet 8 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Treatment	0-FCV-65-47B	8	Air	Gate	EGTS Tr B Decay Cooling Damper
System (65)	1-FCV-65-51	8	Air	Butterfly	EGTS Tr B Unit 1 Suction Damper
	1-FCV-65-52	14	Air	Butterfly	Cntmt Annulus Vacuum Fan Isolation Damper
	1-FCV-65-53	14	Air	Butterfly	Cntmt Annulus Vacuum Fan Isolation Damper
	PCV-65-81	16	Air	Butterfly	Shield Bldg & Cntmt Annulus Isol Damper
	PCV-65-83	16	Air	Butterfly	Shield Bldg & Cntmt Annulus Isol Damper
	PCV-65-86	16	Air	Butterfly	EGTS Cntmt Annulus Isolation Damper
	PCV-65-87	16	Air	Butterfly	EGTS Cntmt Annulus Isolation Damper
	2-FCV-65-4	14	Air	Butterfly	Cntmt Annulus Vacuum Fan Isolation Damper
	2-FCV-65-5	14	Air	Butterfly	Cntmt Annulus Vacuum Fan Isolation Damper
	2-FCV-65-7	8	Air	Butterfly	EGTS Tr A Unit 2 Suction Damper
	2-FCV-65-9	24	Air	Butterfly	EGTS Tr A Unit 2 Suction Damper
	2-FCV-65-29	24	Air	Butterfly	EGTS Tr B Unit 2 Suction Damper
	2-FCV-65-50	8	Air	Butterfly	EGTS Tr B Unit 2 Suction Damper
Essential Raw	FCV-67-9A	4	Motor	Ball	ERCW strainer
Cooling Water (67)	FCV-67-9B	4	Motor	Ball	ERCW strainer
(See Note **)	FCV-67-10A	4	Motor	Ball	ERCW strainer
	FCV-67-10B	4	Motor	Ball	ERCW strainer
	FCV-67-66	8	Motor	Butterfly	Diesel Generator Cooling
	FCV-67-67	8	Motor	Butterfly	Diesel Generator Cooling
	FCV-67-83	6	Motor	Butterfly	Containment isolation
	FCV-67-87	6	Motor	Butterfly	Containment isolation
	FCV-67-88	6	Motor	Butterfly	Containment isolation
	FCV-67-89	6	Motor	Butterfly	Containment isolation
	FCV-67-91	6	Motor	Butterfly	Containment isolation
	FCV-67-95	6	Motor	Butterfly	Containment isolation
	FCV-67-96	6	Motor	Butterfly	Containment isolation
	FCV-67-97	6	Motor	Butterfly	Containment isolation
	FCV-67-99	6	Motor	Butterfly	Containment isolation
	FCV-67-103	6	Motor	Butterfly	Containment isolation
	FCV-67-104	6	Motor	Butterfly	Containment isolation
	FCV-67-105	6	Motor	Butterfly	Containment isolation
	FCV-67-107	6	Motor	Butterfly	Containment isolation
	FCV-67-111	6	Motor	Butterfly	Containment isolation
	FCV-67-112	6	Motor	Butterfly	Containment isolation
	FCV-67-113	6	Motor	Butterfly	Containment isolation
	FCV-67-123	18	Motor	Butterfly	Containment spray cooling
	FCV-67-124	18	Motor	Butterfly	Containment spray cooling
	FCV-67-125	18	Motor	Butterfly	Containment spray cooling

TABLE 3.9-25 (Sheet 9 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Essential Raw	FCV-67-126	18	Motor	Butterfly	Containment spray cooling
Cooling Water (67)	FCV-67-130	2	Motor	Plug	Containment isolation
(See Note **)	FCV-67-131	2	Motor	Plug	Containment isolation
	FCV-67-133	2	Motor	Plug	Containment isolation
	FCV-67-134	2	Motor	Plug	Containment isolation
	FCV-67-138	2	Motor	Plug	Containment isolation
	FCV-67-139	2	Motor	Plug	Containment isolation
	FCV-67-141	2	Motor	Plug	Containment isolation
	FCV-67-142	2	Motor	Plug	Containment isolation
	FCV-67-143	12	Motor	Globe	ERCW flow path integrity
	O-FCV-67-144	16	Motor	Globe	ERCW flow path integrity
	1-FCV-67-146	24	Motor	Butterfly	ERCW from CCS Heat Exchanger A
	2-FCV-67-146	24	Motor	Butterfly	ERCW from CCS Heat Exchanger B
	O-FCV-67-152	24	Motor	Butterfly	CCS flow path integrity
	1-TCV-67-158	4	Self -Actuated	Globe	SBR temperature control
	2-TCV-67-158	4	Self-Actuated	Globe	SBR temperature control
	1-FCV-67-162	2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-164	2	Air	Globe	ERCW to pump room coolers
	FCV-67-176	1-1/2	Air	Globe	ERCW to pump room coolers
	FCV-67-182	1-1/2	Air	Globe	ERCW to pump room coolers
	FCV-67-184	1-1/2	Air	Globe	ERCW to pump room coolers
	FCV-67-186	1-1/2	Air	Globe	ERCW to pump room coolers
	0-FCV-67-205	4	Motor	Butterfly	Nonessential equipment isolation
	0-FCV-67-208	4	Motor	Butterfly	Nonessential Equipment isolation
	1-FCV-67-213	1-1/2	Air	Globe	ERCW to pump room coolers
	1-FCV-67-215	1-1/2	Air	Globe	ERCW to pump room coolers
	2-FCV-67-217	2	Air	Globe	ERCW to pump room coolers
	2-FCV-67-219	2	Air	Globe	ERCW to pump room coolers
	FCV-67-295	2	Motor	Plug	Containment isolation
	FCV-67-296	2	Motor	Plug	Containment isolation
	FCV-67-297	2	Motor	Plug	Containment isolation
	FCV-67-298	2	Motor	Plug	Containment isolation
	2-FCV-67-336	1	Air	Globe	ERCW to pump room coolers
	2-FCV-67-338	1	Air	Globe	ERCW to pump room coolers
	FCV-67-342	2	Air	Globe	ERCW to pump room coolers
	FCV-67-344	2	Air	Globe	ERCW to pump room coolers
	FCV-67-346	1-1/2	Air	Globe	ERCW to pump room coolers
	FCV-67-348	1-1/2	Air	Globe	ERCW to pump room coolers
	FCV-67-350	1-1/2	Air	Globe	ERCW to pump room coolers
	FCV-67-352	1-1/2	Air	Globe	ERCW to pump room coolers

TABLE 3.9-25 (Sheet 10 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Essential Raw	FCV-67-354	1-1/2	Air	Globe	ERCW to pump room coolers
Cooling Water (67)	FCV-67-356	1-1/2	Air	Globe	ERCW to pump room coolers
(See Note **)	0-CKV-67-503A	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503B	20	Self-actuated	Check	ERCW flow path integrity
	0-CKV-67-503C	20	Self-actuated	Check	ERCW flow path integrity
	0-TCV-67-1050-A	3	Self-actuated	Globe	EBR condenser temperature control
	0-TCV-67-1051-A	3	Self-actuated	Globe	MCR condenser temperature control
	0-TCV-67-1052-B	3	Self-actuated	Globe	EBR condenser temperature control
	0-TCV-67-1053-B	3	Self-Actuated	Globe	MRC condenser temperature control
	CKV-67-508A	8	Self-Actuated	Check	ERCW flow path integrity
	CKV-67-508B	8	Self-Actuated	Check	ERCW flow path integrity
	0-CKV-67-517A	10	Self-Actuated	Check	ERCW flow path integrity
	0-CKV-67-512A	10	Self-Actuated	Check	ERCW flow path integrity
	1-CKV-67-940A	3	Self-actuated	Check	ERCW flowpath integrity
	2-CKV-67-935B	3	Self-actuated	Check	ERCW flowpath integrity
	0-CKV-67-503D	20	Self-actuated	Check	ERCW flowpath integrity
	0-CKV-67-503E	20	Self-actuated	Check	ERCW flowpath integrity
	0-CKV-67-503F	20	Self-actuated	Check	ERCW flowpath integrity
	0-CKV-67-503G	20	Self-actuated	Check	ERCW flowpath integrity
	0-CKV-67-503H	20	Self-actuated	Check	ERCW flowpath integrity
	CKV-67-575A	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-575B	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-575C	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-575D	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-580A	2	Self-actuated	Check	Containment Isolation
	CKV-67-580B	2	Self-actuated	Check	Containment Isolation
	CKV-67-580C	2	Self-actuated	Check	Containment Isolation
	CKV-67-580D	2	Self-actuated	Check	Containment Isolation
	CKV-67-585A	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-585B	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-585C	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-585D	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-1054A	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-1054B	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-1054C	1/2	Self-actuated	Check	Containment Isolation
	CKV-67-1054D	1/2	Self-actuated	Check	Containment Isolation
	0-CKV-67-502A	2	Self-actuated	Check	Air release
	0-CKV-67-502B	2	Self-actuated	Check	Air release
	0-CKV-67-502C	2	Self-actuated	Check	Air release
	0-CKV-67-502D	2	Self-actuated	Check	Air release

TABLE 3.9-25 (Sheet 11 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Essential Raw Cooling Water (67) (See Note **)	0-CKV-67-502E	2	Self-actuated	Check	Air release
	0-CKV-67-502F	2	Self-actuated	Check	Air release
	0-CKV-67-502G	2	Self-actuated	Check	Air release
	0-CKV-67-502H	2	Self-actuated	Check	Air release
	0-FSV-67-1221-A	1	Solenoid	Globe	ERCW Supply
	0-PCV-67-1222	1	Self-actuated	Regulating	Press Cntr for ERCW to Aux Air Comp A
	0-TCV-67-1222A	1/2	Self-actuated	Regulating	Throttles ERCW to Aux Air Comp A cyl jacket
	0-TCV-67-1222B	3/4	Self-actuated	Regulating	Throttles ERCW to Aux Air Comp A cyl jacket
	0-FSV-67-1223-B	1	Solenoid	Globe	ERCW Supply
	0-PCV-67-1224	1	Self-actuated	Regulating	Press Cntr for ERCW to Aux Air Comp B
	0-TCV-67-1224A	1/2	Self-actuated	Regulating	Throttles ERCW to Aux Air Comp B cyl jacket
	0-TCV-67-1224B	3/4	Self-actuated	Regulating	Throttles ERCW to Aux Air Comp B cyl jacket
Component Cooling Water (70)	FCV-70-85	6	Air	Butterfly	Containment Isolation
	FCV-70-87	3	Motor	Gate	Containment Isolation
	FCV-70-89	6	Motor	Butterfly	Containment Isolation
	FCV-70-90	3	Motor	Gate	Containment Isolation
	FCV-70-92	6	Motor	Butterfly	Containment Isolation
	FCV-70-100	6	Motor	Butterfly	Containment Isolation
	FCV-70-133	3	Motor	Gate	RCP Thermal Barrier Isolation
	FCV-70-134	3	Motor	Gate	Containment Isolation
	FCV-70-140	6	Motor	Butterfly	Containment Isolation
	FCV-70-143	6	Motor	Butterfly	Containment Isolation
	FCV-70-156	18	Motor	Butterfly	RHR Hx Isolation
	FCV-70-194	20	Motor	Butterfly	Spent Fuel Pit Hx Isolation
	FCV-70-197	20	Motor	Butterfly	Spent fuel pit Hx isolation
	0-CKV-70-504	16	Self-actuated	Check	Prevent backflow through CCP
	CKV-70-504A	16	Self-actuated	Check	Prevent backflow through CCP
	CKV-70-504B	16	Self-actuated	Check	Prevent backflow through CCP
	CKV-70-679	3	Self-actuated	Check	Containment isolation CCP
	CKV-70-698	3/4	Self-actuated	Check	Containment isolation CCP
	CKV-70-790	3/4	Self-actuated	Check	Containment isolation
	FCV-70-183	3	Motor	Gate	Sample Hx Isolation
	FCV-70-215	3	Motor	Gate	Sample Hx Isolation
	1-FCV-70-66	2	Air	Angle	CCS surge tank isolation
	2-FCV-70-66	2	Air	Angle	CCS surge tank isolation
	RFV-70-538	3X4	Self actuated	Relief	Surge tank 1A/1B relief
	CKV-70-681-A,B, C, D	1-1/2	Self actuated	Check	Overpressurization protection
	CKV-70-682-A,B, C, D	1-1/2	Self actuated	Check	Overpressurization protection
	CKV-70-687	3/4	Self actuated	Check	RCP thermal barrier isolation

TABLE 3.9-25 (Sheet 12 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Component Cooling Water (70)	1-RFV-70-703	3X4	Self actuated	Relief	Excess letdown HX relief
	2-RFV-70-703	3X4	Self actuated	Relief	Excess letdown HX relief
	RFV-70-835	3/4 X 1	Self actuated	Relief	RCP thermal barrier supply relief
Containment Spray System (72)	2-FCV-72-13-B	2	Motor	Globe	Min-flow pump recirculation control valve
	2-FCV-72-34-A	2	Motor	Globe	Min-flow pump recirculation control valve
	2-RFV-72-40	3/4 x 1	Self-Actuated	Relief	Bonnet Overpressure Relief Valve for 2-FCV-72-40
	2-RFV-72-41	3/4 x 1	Self-Actuated	Relief	Bonnet Overpressure Relief Valve for 2-FCV-72-41
Primary Water System (81)	FCV-81-12	3	Air	Gate	Primary water to RCS PRT
	CKV-81-502	3	Self-actuated	Gate	Primary water to RCS PRT
Standby Diesel Generators (82)	1-FCV-82-160	1.500	Air	Diaph	Flow Cont 1A1 TK A
	1-FSV-82-160-A	0.370	Elec	Sol V	Air ST SOL 1A1 (A)
	1-FCV-82-161	1.500	Air	Diaph	Flow Cont 1A2 TK A
	1-FSV-82-161-A	0.370	Elec	Sol V	Air ST SOL 1A2 (A)
	1-PCV-82-162A	2.000	Air	Diaph	PR RED V 1A1 TK A
	1-PREG-82-162B	0.250	Self	Diaph	PR REG V 1A1 TK A
	1-PCV-82-163A	2.000	Air	Diaph	PR RED V 1A2 TK A
	1-PREG-82-163B	0.250	Self	Diaph	PR REG V 1A2 TK A
	1-FCV-82-170	1.500	Air	Diaph	FLOW CONT 1A1 TK B
	1-FSV-82-170-A	0.370	Elec	Sol V	AIR ST SOL 1A1 (B)
	1-FCV-82-171	1.500	Air	Diaph	FLOW CONT 1A2 TK B
	1-FSV-82-171-A	0.370	Elec	Sol V	AIR ST SOL 1A2 (B)
	1-PCV-82-172A	2.000	Air	Diaph	PR RED V 1A1 TK B
	1-PREG-82-172B	0.250	Self	Diaph	PR REG V 1A1 TK B
	1-PCV-82-173A	2.000	Air	Diaph	PR RED V 1A2 TK B
	1-PREG-82-173B	0.250	Self	Diaph	PR REG V 1A2 TK B
	1-FCV-82-190	1.500	Air	Diaph	FLOW CONT 1B1 TK A
	1-FSV-82-190-A	0.370	Elec	Sol V	AIR ST SOL 1B1(A)
	1-FCV-82-191	1.500	Air	Diaph	FLOW CONT 1B2 TK A
	1-FSV-82-191-A	0.370	Elec	Sol V	AIR ST SOL 1B2 (A)
	1-PCV-82-192A	2.000	Air	Diaph	PR RED V 1B1 TK A
	1-PREG-82-192B	0.250	Self	Diaph	PR REG V 1B1 TK A
	1-PCV-82-193A	2.000	Air	Diaph	PR RED V 1B2 TK A
	1-PREG-82-193B	0.250	Self	Diaph	PR REG V 1B2 TK A
	1-FCV-82-200	1.500	Air	Diaph	FLOW CONT 1B1 TK B
	1-FSV-82-200-A	0.370	Elec	Sol V	AIR ST SOL 1B1 (B)

TABLE 3.9-25 (Sheet 13 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Standby Diesel Generators (82)	1-FCV-82-201	1.500	Air	Diaph	FLOW CONT 1B2 TK B
	1-FSV-82-201-A	0.370	Elec	Sol V	AIR ST SOL 1B2 (B)
	1-PCV-82-202A	2.000	Air	Diaph	PR RED V 1B1 TK B
	1-PREG-82-202B	0.250	Self	Diaph	PR REG V 1B1 TK B
	1-PCV-82-203A	2.000	Air	Diaph	PR RED V 1B2 TK B
	1-PREG-82-203B	0.250	Self	Diaph	PR REG V 1B2 TK B
	2-FCV-82-220	1.500	Air	Diaph	FLOW CONT 2A1 TK A
	2-FSV-82-220-A	0.370	Elec	Sol V	AIR ST SOL 2A1 (A)
	2-FCV-82-221	1.500	Air	Diaph	FLOW CONT 2A2 TK A
	2-FSV-82-221-A	0.370	Elec	Sol V	AIR ST SOL 2A2 (A)
	2-PCV-82-222A	2.000	Air	Diaph	PR RED V 2A1 TK A
	2-PREG-82-222B	0.250	Self	Diaph	PR REG V 2A1 TK A
	2-PCV-82-223A	2.000	Air	Diaph	PR RED V 2A2 TK A
	2-PREG-82-223B	0.250	Self	Diaph	PR REG V 2A2 TK A
	2-FCV-82-230	1.500	Air	Diaph	FLOW CONT 2A1 TK B
	2-FSV-82-230-A	0.370	Elec	Sol V	AIR ST SOL 2A1 (B)
	2-FCV-82-231	1.500	Air	Diaph	FLOW CONT 2A2 TK B
	2-FSV-82-231-A	0.370	Elec	Sol V	AIR ST SOL 2A2 (B)
	2-PCV-82-232A	2.000	Air	Diaph	PR RED V 2A1 TK B
	2-PREG-82-232B	0.250	Self	Diaph	PR REG V 2A1 TK B
	2-PCV-82-233A	2.000	Air	Diaph	PR RED V 2A2 TK B
	2-PREG-82-233B	0.250	Self	Diaph	PR REG V 2A2 TK B
	2-FCV-82-250	1.500	Air	Diaph	FLOW CONT 2B1 TK A
	2-FSV-82-250-A	0.370	Elec	Sol V	AIR ST SOL 2B1(A)
	2-FCV-82-251	1.500	Air	Diaph	FLOW CONT 2B2 TK A
	2-FSV-82-251-A	0.370	Elec	Sol V	AIR ST SOL 2B2 (A)
	2-PCV-82-252A	2.000	Air	Diaph	PR RED V 2B1 TK A
	2-PREG-82-252B	0.250	Self	Diaph	PR REG V 2B1 TK A
	2-PCV-82-253A	2.000	Air	Diaph	PR RED V 2B2 TK A
	2-PREG-82-253B	0.250	Self	Diaph	PR REG V 2B2 TK A
	2-FCV-82-260	1.500	Air	Diaph	FLOW CONT 2B1 TK B
	2-FSV-82-260-A	0.370	Elec	Sol V	AIR ST SOL 2B1 (B)
	2-FCV-82-261	1.500	Air	Diaph	FLOW CONT 2B2 TK B
	2-FSV-82-261-A	0.370	Elec	Sol V	AIR ST SOL 2B2 (B)
	2-PCV-82-262A	2.000	Air	Diaph	PR RED V 2B1 TK B
	2-PREG-82-262B	0.250	Self	Diaph	PR REG V 2B1 TK B
	2-PCV-82-263A	2.000	Air	Diaph	PR RED V 2B2 TK B
	2-PREG-82-263B	0.250	Self	Diaph	PR REG V 2B2 TK B
	1,2-CKV-82-502A1-A	0.750	Self	Check	TK SUP A1 (A)
	1,2-CKV-82-502B1-B	0.750	Self	Check	TK SUP B1 (A)

TABLE 3.9-25 (Sheet 14 of 14)
UNIT 2

VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS

System Name	Valve No.	Size Inches	Actuation	Type	Function/Description
Standby Diesel Generators (82)	1,2-CKV-82-505A1-A	0.750	Self	Check	TK SUP A1 (B)
	1,2-CKV-82-505B1-B	0.750	Self	Check	TK SUP B1 (B)
	1,2-CKV-82-509A1-A	0.750	Self	Check	CROSS CONN CK A1
	1,2-CKV-82-509B1-B	0.750	Self	Check	CROSS CONN CK B1
	1,2-CKV-82-523A1-A	0.750	Self	Check	RELAY CK A1 (A)
	1,2-CKV-82-523B1-B	0.750	Self	Check	RELAY CK B1 (A)
	1,2-SPV-82-524A1-A	0.375	Self	Check	SLIDE VLV A1
	1,2-SPV-82-524B1-B	0.375	Self	Check	SLIDE VLV B1
	1,2-CKV-82-531A1-A	0.750	Self	Check	RELAY CK A1 (B)
	1,2-CKV-82-531B1-B	0.750	Self	Check	RELAY CK B1 (B)
	1,2-CKV-82-536-A2-A	0.750	Self	Check	TK SUP A2 (A)
	1,2-CKV-82-536B2-B	0.750	Self	Check	TK SUP B2 (A)
	1,2-CKV-82-539-A2-A	0.750	Self	Check	TK SUP A2 (B)
	1,2-CKV-82-539B2-B	0.750	Self	Check	TK SUP B2 (B)
	1,2-CKV-82-543-A2-A	0.750	Self	Check	CROSS CONN CK A2
	1,2-CKV-82-543B2-B	0.750	Self	Check	CROSS CONN CK B2
	1,2-CKV-82-557-A2-A	0.750	Self	Check	RELAY CK A2 (A)
	1,2-CKV-82-557B2-B	0.750	Self	Check	RELAY CK B2 (A)
	1,2-SPV-82-558-A2-A	0.375	Self	Check	SLIDE VLV A2
	1,2-SPV-82-558B2-B	0.375	Self	Check	SLIDE VLV B2
	1,2-CKV-82-565-A2-A	0.750	Self	Check	RELAY CK A2 (B)
	1,2-CKV-82-565B2-B	0.750	Self	Check	RELAY CK B2 (B)
Radiation Monitoring (90)	1,2-FCV-90-107	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-108	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-109	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-110	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-111	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-113	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-114	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-115	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-116	1-1/2	Air	Globe	Containment Isolation
	1,2-FCV-90-117	1-1/2	Air	Globe	Containment Isolation

**Valves not prefixed by 1-, 2- or 0-are understood to be prefixed by 1- and 2

TABLE 3.9-26 (Sheet 1 of 4)
UNIT 1

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
MAIN STEAM 47W801-1	B	B	FCV-1-4, FCV-1-11, FCV-1-22, FCV-1-29, FCV-1-147, FCV-1-148, FCV-1-149, FCV-1-150
	B	B	PCV-1-5, PCV-1-12, PCV-1-23, PCV-1-30, 1-ISV-1-619, 1-ISV-1-620, 1-ISV-1-621, 1-ISV-1-622
	C	B	1-512, 1-513, 1-514, 1-515, 1-516, 1-517, 1-518, 1-519, 1-520, 1-521, 1,522, 1-523, 1-524, 1-525, 1-526, 1-527, 1-528, 1-529, 1-530, 1-531
STEAM GENERATOR BLOWDOWN SYSTEM 47W801-2	B	B	FCV-1-7, FCV-1-14, FCV-1-25, FCV-1-32, FCV-1-181, FCV-1-182, FCV-1-183, FCV-1-184
CONDENSATE 47W804-1	C	Note 1	2-667
FEEDWATER 47W803-1	B	B	FCV-3-33, FCV-3-47, FCV-3-87, FCV-3-100 FCV-3-236, FCV-3-239, FCV-3-242, FCV-3-245
	B	Note 1 & C	FCV-3-35, FCV-3-35A, FCV-3-48, FCV-3-48A, FCV-3-90, FCV-3-90A, FCV-3-103, FCV-3-103A
	C	B	3-508, 3-509, 3-510, 3-511, 3-644, 3-645, 3-655, 3-656, 3-670, 3-679
	AC	B	3-638, 3-652, 3-669, 3-678
AUXILIARY FEEDWATER 47W803-2	B	B	FCV-1-15, FCV-1-16, FCV-1-18
	B	C	LCV-3-148, LCV-3-148A, LCV-3-156, LCV-3-156A, LCV-3-164, LCV-3-164A, LCV-3-171, LCV-3-171A, LCV-3-172, LCV-3-173, LCV-3-174, FCV-3-175, FCV-1-17, FCV-3-116A, FCV-3-116B, FCV-3-126A, FCV-3-126B, FCV-3-136A, FCV-3-136B, FCV-3-179A, FCV-3-179B,FCV-1-51, PCV-3-122, PCV-3-132 , FCV-3-355, FCV-3-359
	C	B	3-830, 3-831, 3-832, 3-833, 3-861, 3-862, 3-864, 3-871, 3-872, 3-873, 3-874, 1-891, 1-892, 3-921, 3-922
	C	C	3-805, 3-806, 3-810, 3-814, 3-815, 3-818
CHEMICAL & VOLUME CONTROL 47W809-1, -2, -3,-5	A	B	FCV-62-61, FCV-62-63, FCV-62-72, FCV-62-73 FCV-62-74, FCV-62-77, FCV-62-76
	AC	B	62-639, 62-662, 62-523, 62-525, 62-530, 62-532
	B	B	LCV-62-132, LCV-62-133, LCV-62-135, LCV-62-136, ,FCV-62-1228, FCV-62-1229, FCV-62-90, FCV-62-91, PCV-62-81
	C	B	62-504, 62-505, 62-518 (Note 3) 62-636, 62-649, 62-675, 62-1220 (Note 3), 62-1221, 62-1222
	B	C	FCV-62-138
	C	C	CKV-62-930, CKV-62-1052-A, CKV-62-1052-B
	C	B	RFV-62-955, RFV-62-1079
RESIDUAL HEAT REMOVAL 47W810-1	A	A	FCV-74-1, FCV-74-2, FCV-74-8, FCV-74-9

TABLE 3.9-26 (Sheet 2 of 4)
UNIT 1

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
RESIDUAL HEAT REMOVAL 47W810-1 (cont.)	B	B	FCV-74-3, FCV-74-12, FCV-74-21,FCV-74-24, FCV-74-33, FCV-74-35
	C	B	74-505, 74-514, 74-515, 74-544, 74-545
SAFETY INJECTION 47W811-1	A	B	FCV-63-23, FCV-63-64, FCV-63-71, FCV-63-84
	AC	A	63-543, 63-545, 63-547, 63-549, 63-551, 63-553, 63-555, 63-557, 63-558, 63-559, 63-560, 63-561, 63-562, 63-563, 63-622, 63-623, 63-624, 63-625, 63-632, 63-633, 63-634, 63-635, 63-640, 63-641, 63-643, 63-644, 63-581, 63-586, 63-587, 63-588, 63-589
	AC	B	CKV-63-524,CKV-63-526, CKV-63-528, CKV-63-530, CKV-63-868, 1-RFV-63-28
	B	B	FCV-63-1, FCV-63-3, FCV-63-4, FCV-63-5, FCV-63-6, FCV-63-7, FCV-63-8, FCV-63-11, FCV-63-22, FCV-63-25, FCV-63-26, FCV-63-47, FCV-63-48, FCV-63-72, FCV-63-73, FCV-63-93, FCV-63-94, FCV-63-152, FCV-63-153, FCV-63-156, FCV-63-157, FCV-63-172, FCV-63-175, FCV-63-185
	C	B	63-502, 63-510, 63-511, 63-534, 63-535, 63-536, 63-577, 63-602, 63-603, 63-604, 63-605, 63-626, 63-627, 63-637, 63-725, 63-835
CONTAINMENT SPRAY 47W812-1	A	B	FCV-72-2, FCV-72-39, FCV-72-40, FCV-72-41, RFV-72-40, RFV-72-41
	B	B	FCV-72-13, FCV-72-21, FCV-72-22, FCV-72-34, FCV-72-44, FCV-72-45
	C	B	72-506, 72-507, 72-508, 72-509, 72-524, 72-525, 72-548, 72-549, 72-562, 72-563
REACTOR COOLANT 47W813-1	A	B	FCV-68-305, FCV-68-307, FCV-68-308
	B	A	FCV-68-332, FCV-68-333, PCV-68-334, PCV-68-340A
	B	B	FSV-68-394, FSV-68-395, FSV-68-396, FSV-68-397
	C	A	68-563, 68-564, 68-565
	C	B	68-559, 68-849
ESSENTIAL RAW COOLING WATER 47W845-1,2,3,4,5,7	A	B	FCV-67-83, FCV-67-87, FCV-67-88, 1-FCV-67-89,FCV-67-91, FCV-67-95, FCV-67-96, 1-FCV-67-97, FCV-67-99, FCV-67-103, FCV-67-104, 1-FCV-67-105, FCV-67-107, FCV-67-111, FCV-67-112, 1-FCV-67-113, FCV-67-130, FCV-67-131, FCV-67-133, FCV-67-134, FCV-67-138, FCV-67-139, FCV-67-141, FCV-67-142, FCV-67-295, FCV-67-296, FCV-67-297, FCV-67-298
	AC	B	67-575A, 67-575B, 67-575C, 67-575D, 67-580A, 67-580B, 67-580C, 67-580D, 67-585A, 67-585B, 67-585C, 67-585D, 1-67-1060A, 1-67-1060B, 1-67-1060C, 1-67-1060D
	B	C	FCV-67-65, FCV-67-66, FCV-67-67, FCV-67-68, FCV-67-123, FCV-67-124, FCV-67-125, FCV-67-126, FCV-67-143, FCV-67-144, FCV-67-146, FCV-67-152, FCV-67-162, FCV-67-164, FCV-67-176, FCV-67-182, FCV-67-184, FCV-67-186, FCV-67-205, FCV-67-208, FCV-67-213, FCV-67-215, FCV-67-342, FCV-67-344, FCV-67-346, FCV-67-348, FCV-67-350, FCV-67-352, FCV-67-354, FCV-67-356, 2-FCV-67-217, 2-FCV-67-219, 2-FCV-67-336, 2-FCV-67-338, FCV-67-9A, FCV-67-9B, FCV-67-10A, FCV-67-10B, 2-FCV-67-354, 2-FCV-67-356

TABLE 3.9-26 (Sheet 3 of 4)
UNIT 1

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
	C	C	67-503A, 67-503B, 67-503C, 67-503D, 67-503E, 67-503F, 67-503G, 67-503H, 1-67-508A, 2-67-508A, 1-67-508B, 2-67-508B, 1-67-513A, 2-67-513A, 1-67-513B, 2-67-513B, 67-502A thru H, 1-67-940A, 2-67-935B, 0-RFV-67-671, 0-RFV-67-672, RFV-67-539A, RFV-67-539B
COMPONENT COOLING WATER 47W859-1, -2, -4	A	B	FCV-70-85, FCV-70-87, FCV-70-89, FCV-70-90, FCV-70-92, 1-FCV-70-100, FCV-70-134, FCV-70-140, FCV-70-143
COMPONENT COOLING WATER 47W859-1, -2, -4 (continued)	AC	B	70-679, 70-687, 70-698, 70-703, 1-70-790
	B	C	FCV-70-66, FCV-70-133, 1, 2-FCV-70-153, FCV-70-156, FCV-70-183, FCV-70-197, FCV-70-215
	C	C	70-504, 70-504A, 70-504B, 70-538, 70-753, 70-539, 70-681A, 70-681B, 70-681C, 70-681D, 70-682A, 70-682B, 70-682C, 70-682D, RFV-70-551A, RFV-70-551B, RFV-70-556A, RFV-70-556B, RFV-70-565A, RFV-70-565B, RFV-70-578, RFV-70-835
PRIMARY WATER 47W819-1	A	B	FCV-81-12
	AC	B	81-502
WASTE DISPOSAL 47W830-1 47W851-1	A	B	FCV-77-9, FCV-77-10, FCV-77-16, FCV-70-17, FCV-77-18, FCV-77-19, FCV-77-20, FCV-77-127, FCV-77-128
	AC	B	77-2875
FIRE PROTECTION 47W850-9	A	B	FCV-26-240, FCV-26-243
	AC	B	26-1260, 26-1296
HEATING AND VENTILATION 47W866-1	A	B	FCV-30-7, FCV-30-8, FCV-30-9, FCV-30-10, FCV-30-14, FCV-30-15, FCV-30-16, FCV-30-17, FCV-30-19, FCV-30-20, FCV-30-37, FCV-30-40, FCV-30-50, FCV-30-51, FCV-30-52, FCV-30-53, FCV-30-56, FCV-30-57, FCV-30-58, FCV-30-59, FSV-30-134, FSV-30-135
AIR CONDITIONING 47W865-5, -3, -7, -8	A	B	FCV-31-305, FCV-31-306, FCV-31-308, FCV-31-309, FCV-31-326, FCV-31-327, FCV-31-329, FCV-31-330
	AC	B	31-3378, 31-3392, 31-3407, 31-3421
	C	Note 1	0-31-2210, 0-31-2252, 0-31-2326, 0-31-2383, 0-31-2623, 0-31-2665
CONTROL AIR 47W848-1	A	B	FCV-32-80, FCV-32-102, FCV-32-110, BYV-32-288, BYV-32-298, BYV-32-308
	AC	B	32-293, 32-303, 32-313
SERVICE AIR 47W846-2	A	B	33-713, 33-714

TABLE 3.9-26 (Sheet 4 of 4)
UNIT 1

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
RADIATION SAMPLING 47W625-1, -2, -7, -11, -15	A	B	FCV-43-2, FCV-43-3, FCV-43-11, FCV-43-12, FCV-43-22, FCV-43-23, FCV-43-34, FCV-43-35, FCV-43-75, FCV-43-77, FCV-43-201, FCV-43-202, FCV-43-207, FCV-43-208,FSV-43-250, FSV-43-251, FSV-43-287, FSV-43-288, FSV-43-307, FSV-43-309, FSV-43-310, FSV-43-318,FSV-43-319, FSV-43-325, FSV-43-341, FSV-43-342
	B	B	FCV-43-54D, FCV-43-55, FCV-43-56D, FCV-43-58, FCV-43-59D, FCV-43-61, FCV-43-63D, FCV-43-64
	AC	B	43-834, 43-841, 43-883, 43-884
SYSTEM TEST FACILITY 47W331-3	A	B	52-500, 52-501, 52-502, 52-503, 52-504, 52-505, 52-506, 52-507
DEMINERALIZED WATER AND CASK DECONTAMINATION 47W856-1	A	B	59-522, 59-698
ICE CONDENSER 47W814-2	A	B	FCV-61-96, FCV-61-97, FCV-61-110, FCV-61-122, FCV-61-191, FCV-61-192, FCV-61-193, FCV-61-194
	AC	B	61-533, 61-680, 61-692, 61-745
	C	Note 1	61-658, 61-659, 61-660, 61-661, 61-662, 61-663, 61-664, 61-665, 61-666, 61-667, 61-668, 61-669, 61-670, 61-671, 61-672, 61-673, 61-674, 61-675, 61-676, 61-677
BORATION MAKEUP 47W809-7	A	B	84-530
RADIATION MONITORING 47W610-90-3	A	B	FCV-90-107, FCV-90-108, FCV-90-109, FCV-90-110, FCV-90-111, FCV-90-113, FCV-90-114, FCV-90-115, FCV-90-116, FCV-90-117
FUEL POOL COOLING AND CLEANING 47W855-1	A	B	78-557, 78-558, 78-560, 78-561

Note 1: Not Constructed to an ASME Code Class.

Note 2: Valves listed in Table 3.9-26 are not necessarily Reg Guide 1.48 active and/or event active. Non-active valves may be added for convenience of plant operations to improve a valve's reliability, etc. All active valves are reflected in Tables 3.9-17 and 3.9-25.

Note 3: Valve is abandoned in place.

TABLE 3.9-26 (Sheet 1 of 4)
UNIT 2

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
MAIN STEAM 47W801-1	B	B	FCV-1-4, FCV-1-11, FCV-1-22, FCV-1-29, FCV-1-147, FCV-1-148, FCV-1-149, FCV-1-150
	B	B	PCV-1-5, PCV-1-12, PCV-1-23, PCV-1-30, 1-ISV-1-619, 1-ISV-1-620, 1-ISV-1-621, 1-ISV-1-622
	C	B	1-512, 1-513, 1-514, 1-515, 1-516, 1-517, 1-518, 1-519, 1-520, 1-521, 1,522, 1-523, 1-524, 1-525, 1-526, 1-527, 1-528, 1-529, 1-530, 1-531
STEAM GENERATOR BLOWDOWN SYSTEM 47W801-2	B	B	FCV-1-7, FCV-1-14, FCV-1-25, FCV-1-32, FCV-1-181, FCV-1-182, FCV-1-183, FCV-1-184
CONDENSATE 47W804-1	C	Note 1	2-667
FEEDWATER 47W803-1	B	B	FCV-3-33, FCV-3-47, FCV-3-87, FCV-3-100, 2-FCV-3-185, 2-FCV-3-186, 2-FCV-3-187, 2-FCV-3-188, FCV-3-236, FCV-3-239, FCV-3-242, FCV-3-245
	B	Note 1 & C	FCV-3-35, FCV-3-35A, FCV-3-48, FCV-3-48A, FCV-3-90, FCV-3-90A, FCV-3-103, FCV-3-103A
	C	B	3-508, 3-509, 3-510, 3-511, 3-644, 3-645, 3-655, 3-656, 3-670, 3-679
	AC	B	3-638, 3-652, 3-669, 3-678
AUXILIARY FEEDWATER 47W803-2	B	B	FCV-1-15, FCV-1-16, FCV-1-18
	B	C	LCV-3-148, LCV-3-148A, LCV-3-156, LCV-3-156A, LCV-3-164, LCV-3-164A, LCV-3-171, LCV-3-171A, LCV-3-172, LCV-3-173, LCV-3-174, FCV-3-175, FCV-1-17, FCV-3-116A, FCV-3-116B, FCV-3-126A, FCV-3-126B, FCV-3-136A, FCV-3-136B, FCV-3-179A, FCV-3-179B,FCV-1-51, PCV-3-122, PCV-3-132, FCV-3-355, FCV-3-359
	C	B	3-830, 3-831, 3-832, 3-833, 3-861, 3-862, 3-864, 3-871, 3-872, 3-873, 3-874, 1-891, 1-892, 3-921, 3-922
	C	C	3-805, 3-806, 3-810, 3-814, 3-815, 3-818
CHEMICAL & VOLUME CONTROL 47W809-1, -2, -3,-5	A	B	FCV-62-61, FCV-62-63, FCV-62-72, FCV-62-73 FCV-62-74, FCV-62-77, FCV-62-76
	AC	B	62-639, 62-662, 62-523, 62-525, 62-530, 62-532
	B	B	LCV-62-132, LCV-62-133, LCV-62-135, LCV-62-136, ,FCV-62-1228, FCV-62-1229, FCV-62-90, FCV-62-91, PCV-62-81
	C	B	62-504, 62-505, 62-636, 62-649, 62-675, 62-1221, 62-1222, CKV-62-543
	B	C	FCV-62-138
	C	C	CKV-62-930, CKV-62-1052-A, CKV-62-1052-B
	C	B	RFV-62-955, RFV-62-1079
RESIDUAL HEAT REMOVAL 47W810-1	A	A	FCV-74-1, FCV-74-2, FCV-74-8, FCV-74-9

TABLE 3.9-26 (Sheet 2 of 4)
UNIT 2

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
RESIDUAL HEAT REMOVAL 47W810-1 (cont.)	B	B	FCV-74-3, FCV-74-12, FCV-74-21,FCV-74-24, FCV-74-33, FCV-74-35
	C	B	74-505, 74-514, 74-515, 74-544, 74-545
SAFETY INJECTION 47W811-1	A	B	FCV-63-23, FCV-63-64, FCV-63-71, FCV-63-84
	AC	A	63-543, 63-545, 63-547, 63-549, 63-551, 63-553, 63-555, 63-557, 63-558, 63-559, 63-560, 63-561, 63-562, 63-563, 63-622, 63-623, 63-624, 63-625, 63-632, 63-633, 63-634, 63-635, 63-640, 63-641, 63-643, 63-644, 63-581, 63-586, 63-587, 63-588, 63-589
	AC	B	CKV-63-524,CKV-63-526, CKV-63-528, CKV-63-530, CKV-63-868, RFV-63-28
	B	B	FCV-63-1, FCV-63-3, FCV-63-4, FCV-63-5, FCV-63-6, FCV-63-7, FCV-63-8, FCV-63-11, FCV-63-22, FCV-63-25, FCV-63-26, FCV-63-47, FCV-63-48, FCV-63-72, FCV-63-73, FCV-63-93, FCV-63-94, FCV-63-152, FCV-63-153, FCV-63-156, FCV-63-157, FCV-63-172, FCV-63-175, FCV-63-185
	C	B	63-502, 63-510, 63-511, 63-534, 63-535, 63-536, 63-577, 63-602, 63-603, 63-604, 63-605, 63-626, 63-627, 63-637, 63-725, 63-835
CONTAINMENT SPRAY 47W812-1	A	B	FCV-72-2, FCV-72-39, FCV-72-40, FCV-72-41, RFV-72-40, RFV-72-41
	B	B	FCV-72-13, FCV-72-21, FCV-72-22, FCV-72-34, FCV-72-44, FCV-72-45
	C	B	72-506, 72-507, 72-508, 72-509, 72-524, 72-525, 72-548, 72-549, 72-562, 72-563
REACTOR COOLANT 47W813-1	A	B	FCV-68-305, FCV-68-307, FCV-68-308
	B	A	FCV-68-332, FCV-68-333, PCV-68-334, PCV-68-340A
	B	B	FSV-68-394, FSV-68-395, FSV-68-396, FSV-68-397
	C	A	68-563, 68-564, 68-565
	C	B	68-559, 68-849
ESSENTIAL RAW COOLING WATER 47W845-1,2,3,4,5,7	A	B	FCV-67-83, FCV-67-87, FCV-67-88, 1-FCV-67-89, FCV-67-91, FCV-67-95, FCV-67-96, 1-FCV-67-97, FCV-67-99, FCV-67-103, FCV-67-104, 1-FCV-67-105, FCV-67-107, FCV-67-111, FCV-67-112, 1-FCV-67-113, FCV-67-130, FCV-67-131, FCV-67-133, FCV-67-134, FCV-67-138, FCV-67-139, FCV-67-141, FCV-67-142, FCV-67-295, FCV-67-296, FCV-67-297, FCV-67-298
	AC	B	67-575A, 67-575B, 67-575C, 67-575D, 67-580A, 67-580B, 67-580C, 67-580D, 67-585A, 67-585B, 67-585C, 67-585D, 1-67-1054A, 1-67-1054B, 1-67-1054C, 1-67-1054D

TABLE 3.9-26 (Sheet 3 of 4)
UNIT 2

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
ESSENTIAL RAW COOLING WATER 47W845-1,2,3,4,5,7 (Continued)	B	C	FCV-67-65, FCV-67-66, FCV-67-67, FCV-67-68, FCV-67-123, FCV-67-124, FCV-67-125, FCV-67-126, FCV-67-143, FCV-67-144, 1-FCV-67-146, FCV-67-152, FCV-67-162, FCV-67-164, FCV-67-176, FCV-67-182, FCV-67-184, FCV-67-186, FCV-67-205, FCV-67-208, FCV-67-213, FCV-67-215, FCV-67-342, FCV-67-344, FCV-67-346, FCV-67-348, FCV-67-350, FCV-67-352, FCV-67-354, FCV-67-356, 2-FCV-67-217, 2-FCV-67-219, 2-FCV-67-336, 2-FCV-67-338, FCV-67-9A, FCV-67-9B, FCV-67-10A, FCV-67-10B, 2-FCV-67-354, 2-FCV-67-356
	C	C	67-503A, 67-503B, 67-503C, 67-503D, 67-503E, 67-503F, 67-503G, 67-503H, 1-67-508A, 2-67-508A, 1-67-508B, 2-67-508B, 1-67-513A, 2-67-513A, 1-67-513B, 2-67-513B, 67-502A thru H, 1-67-940A, 2-67-935B, 0-RFV-67-671, 0-RFV-67-672, RFV-67-539A, RFV-67-539B
COMPONENT COOLING WATER 47W859-1, -2, -3, -4	A	B	FCV-70-85, FCV-70-87, FCV-70-89, FCV-70-90, FCV-70-92, 1-FCV-70-100, FCV-70-134, FCV-70-140, FCV-70-143
	AC	B	70-679, 70-687, 70-698, 70-703, 1-70-790
	B	C	FCV-70-66, FCV-70-133, FCV-70-156, FCV-70-183, FCV-70-197, FCV-70-215
	C	C	70-504, 70-504A, 70-504B, 70-538, 70-753, 70-539, 70-681A, 70-681B, 70-681C, 70-681D, 70-682A, 70-682B, 70-682C, 70-682D, RFV-70-551A, RFV-70-551B, RFV-70-556A, RFV-70-556B, RFV-70-565A, RFV-70-565B, RFV-70-578, RFV-70-835
PRIMARY WATER 47W819-1	A	B	FCV-81-12
	AC	B	81-502
WASTE DISPOSAL 47W830-1, 47W851-1	A	B	FCV-77-9, FCV-77-10, FCV-77-16, FCV-70-17, FCV-77-18, FCV-77-19, FCV-77-20, FCV-77-127, FCV-77-128
	AC	B	77-2875
FIRE PROTECTION 47W850-9	A	B	FCV-26-240, FCV-26-243
	AC	B	26-1260, 26-1296
HEATING AND VENTILATION 47W866-1	A	B	FCV-30-7, FCV-30-8, FCV-30-9, FCV-30-10, FCV-30-14, FCV-30-15, FCV-30-16, FCV-30-17, FCV-30-19, FCV-30-20, FCV-30-37, FCV-30-40, FCV-30-50, FCV-30-51, FCV-30-52, FCV-30-53, FCV-30-56, FCV-30-57, FCV-30-58, FCV-30-59, FSV-30-134, FSV-30-135
AIR CONDITIONING 47W865-5, -3, -7, -8	A	B	FCV-31-305, FCV-31-306, FCV-31-308, FCV-31-309, FCV-31-326, FCV-31-327, FCV-31-329, FCV-31-330
	AC	B	31-3378, 31-3392, 31-3407, 31-3421
	C	Note 1	0-31-2193, 0-31-2235, 0-31-2210, 0-31-2252, 0-31-2307, 0-31-2364, 0-31-2326, 0-31-2383, 0-31-2607, 0-31-2649, 0-31-2623, 0-31-2665
CONTROL AIR 47W848-1	A	B	FCV-32-80, FCV-32-102, FCV-32-110, BYV-32-288, BYV-32-298, BYV-32-308
	AC	B	32-293, 32-303, 32-313

TABLE 3.9-26 (Sheet 4 of 4)
UNIT 2

INSERVICE INSPECTION CATEGORY VALVES (Note 2)

SYSTEM	CATEGORY	CLASS	VALVES
SERVICE AIR 47W846-2	A	B	33-713, 33-714
RADIATION SAMPLING 47W625-1, -2, -7, -11, -15	A	B	FCV-43-2, FCV-43-3, FCV-43-11, FCV-43-12, FCV-43-22, FCV-43-23, FCV-43-34, FCV-43-35, 1-FCV-43-75, 1-FCV-43-77, FCV-43-201, FCV-43-202, 1-FCV-43-207, 1-FCV-43-208, 1-FSV-43-250, 1-FSV-43-251, 1-FSV-43-287, 1-FSV-43-288, 1-FSV-43-307, 1-FSV-43-309, 1-FSV-43-310, 1-FSV-43-318, 1-FSV-43-319, 1-FSV-43-325, 1-FSV-43-341, 1-FSV-43-342 FCV-43-433, FCV-43-434, 1-FCV-43-435, 1-FCV-43-436
	B	B	FCV-43-54D, FCV-43-55, FCV-43-56D, FCV-43-58, FCV-43-59D, FCV-43-61, FCV-43-63D, FCV-43-64
	AC	B	1-43-834, 1-43-841, 1-43-883, 1-43-884
SYSTEM TEST FACILITY 47W331-3	A	B	52-500, 52-501, 52-502, 52-503, 52-504, 52-505, 52-506, 52-507
DEMINERALIZED WATER AND CASK DECONTAMINATION 47W856-1	A	B	59-522, 59-698
ICE CONDENSER 47W814-2	A	B	FCV-61-96, FCV-61-97, FCV-61-110, FCV-61-122, FCV-61-191, FCV-61-192, FCV-61-193, FCV-61-194
	AC	B	61-533, 61-680, 61-692, 61-745
	C	Note 1	61-658, 61-659, 61-660, 61-661, 61-662, 61-663, 61-664, 61-665, 61-666, 61-667, 61-668, 61-669, 61-670, 61-671, 61-672, 61-673, 61-674, 61-675, 61-676, 61-677
BORATION MAKEUP 47W809-7	A	B	84-530
RADIATION MONITORING 47W610-90-3	A	B	FCV-90-107, FCV-90-108, FCV-90-109, FCV-90-110, FCV-90-111, FCV-90-113, FCV-90-114, FCV-90-115, FCV-90-116, FCV-90-117
FUEL POOL COOLING AND CLEANING 47W855-1	A	B	78-557, 78-558, 78-560, 78-561

Note 1: Not Constructed to an ASME Code Class.

Note 2: Valves listed in Table 3.9-26 are not necessarily Reg Guide 1.48 active and/or event active. Non-active valves may be added for convenience of plant operations to improve a valve's reliability, etc. All active valves are reflected in Tables 3.9-17 and 3.9-25.

TABLE 3.9-27 (Sheet 1 of 2)

ACTIVE SECTION III ASME-CODED COMPONENTS (EXCEPT VALVES) IN TVA SCOPE OF SUPPLY[1]

SYSTEM NAME	COMPONENT	ANS SAFETY CLASS	NORMAL MODE	POST LOCA MODE	REASON
Feedwater (3)	Auxiliary feedwater pumps:				
	Motor driven				
	1A-A	2B	OFF	ON	Provide heat removal for chapter 15 events.
	1B-B	2B	OFF	ON	
	2A-A	2B	OFF	ON	
	2B-B	2B	OFF	ON	
	Steam driven				
	1A-S	2B	OFF	ON	Provide heat removal for chapter 15 events.
	2A-S	2B	OFF	ON	
Control Air (32)	Auxiliary air compressors A & B	2B	OFF	ON	Provide control air for safety-related equipment.
Essential Raw Cooling Water (67)	ERCW pump A-A	2B	ON	ON	Provide cooling water flow for component cooling system and other heat removal systems.
	B-A	2B	ON	ON	
	C-A	2B	ON	ON	
	D-A	2B	ON	ON	
	E-B	2B	ON	ON	
	F-B	2B	ON	ON	
	G-B	2B	ON	ON	
	H-B	2B	ON	ON	
	Screen Wash Pump				
	1AA	2B	ON	ON	Prevents fouling of ERCW pump station.
	2AA	2B	ON	ON	
	1BB	2B	ON	ON	
	2BB	2B	ON	ON	

TABLE 3.9-27 (Sheet 2 of 2)

ACTIVE SECTION III ASME-CODED COMPONENTS (EXCEPT VALVES) IN TVA SCOPE OF SUPPLY[1]

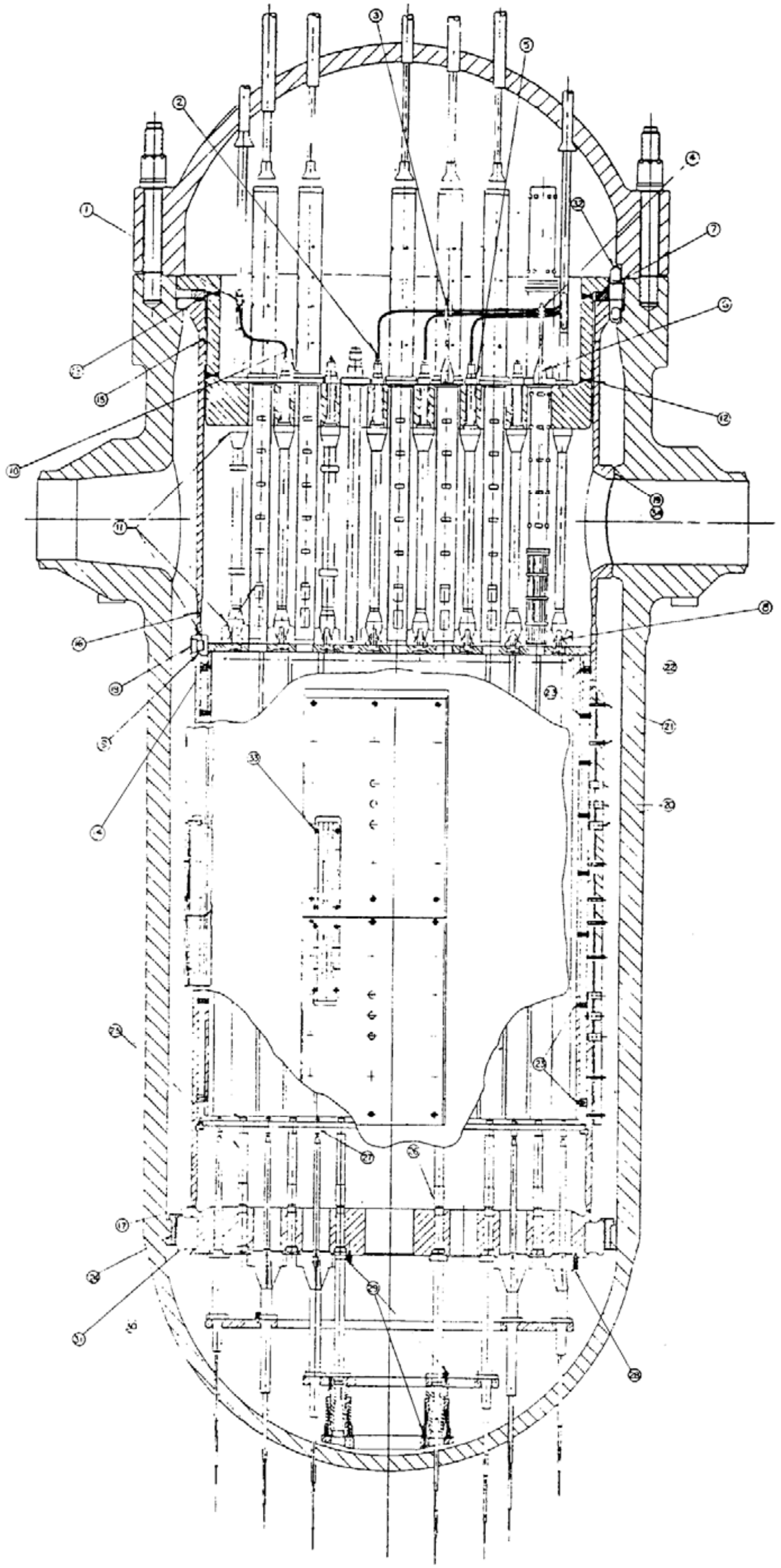
SYSTEM NAME	COMPONENT	ANS SAFETY CLASS	NORMAL MODE	POST LOCA MODE	REASON
Component Cooling (70)	Component cooling pumps1A	2B	ON	ON	Provide cooling water flow for required equipment served by the CCS.
	1B	2B	ON	ON	
	2A	2B	ON	ON	
	2B	2B	ON	ON	
	C-S	2B	ON	ON	
	1A TB Booster	2B	ON	OFF	Provides flow RCP thermal barriers
	1B TB Booster	2B	ON	OFF	
	2A TB Booster	2B	ON	OFF	
	2B TB Booster	2B	ON	OFF	

Note:
(1) As defined in Regulatory Guide 1.48

TABLE 3.9-28

ACTIVE PUMPS FOR PRIMARY FLUID SYSTEMS IN WESTINGHOUSE SCOPE OF SUPPLY AS APPLIED TO WATTS BAR NUCLEAR PLANT

SYSTEM NAME	COMPONENT	ANS SAFETY CLASS	NORMAL MODE	POST LOCA MODE	REASON
CVCS (62)	Centrifugal Charging Pumps				
	1A-A	2A	ON	ON	To provide emergency core cooling, reactivity control and RCP seal injection flow.
	2A-A	2A	ON	ON	
	1B-B	2A	ON	ON	
	2B-2B	2A	ON	ON	
Safety Injection Systems (63)	SIS Pump				
	1A-A	2A	OFF	ON	To provide emergency core cooling and reactivity control.
	1B-B	2A	OFF	ON	
	2A-A	2A	OFF	ON	
	2B-B	2A	OFF	ON	
Containment Spray (72)	Pump				
	1A-A	2A	OFF	ON	Provide cooling water flow to control containment temperature and pressure.
	2A-A	2A	OFF	ON	
	1B-B	2A	OFF	ON	
	2B-B	2A	OFF	ON	
Residual Heat Removal (74)	Pump				
	1A-A	2A	OFF	ON	To provide emergency core cooling and reactivity control and containment temperature and pressure control.
	1B-B	2A	OFF	ON	
	2A-A	2A	OFF	ON	
	2B-B	2A	OFF	ON	
Spent Fuel Pool Cooling (78)	Pump				
	A-A	2B	ON	ON	Provide adequate spent fuel cooling.
	B-B	2B	ON	ON	
	C-S	2B	OFF	OFF	



SEE NOTES	STEP	FEATURES TO BE EXAMINED	COMMENTS AND OBSERVATIONS BEFORE FUNCTIONAL TEST	COMMENTS AND OBSERVATIONS AFTER FUNCTIONAL TEST
1	1.	THERMOCOUPLE CONDUIT CLAMPS INSIDE THE THERMOCOUPLE COLUMN	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 5	2.	CONDUIT SWAGLOCK FITTINGS, THEIR BANDINGS, AND THE TAB TYPE LOCKS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	3.	CLAMP ARRANGEMENTS AT THE MOUNTING BRACKET LOCATIONS.	ACCEPTED	NO APPARENT DEFECTS
1	4.	CONDUIT FITTING TO SUPPORT BRACKET WELDS ADJACENT TO T/C COLUMNS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	5.	UPPER SUPPORT COLUMN NUT TO EXTENSION WELDS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	6.	ACCESSIBLE CONDUIT SUPPORT BRACKET WELDS	ACCEPTED	NO APPARENT DEFECTS
6	7.	HOLD DOWN SPRING INTERFACE SURFACE CONDITION.	NO DEFECTS	NO APPARENT DEFECTS
1	8.	ACCESSIBLE WELDS ON SUPPORT COLUMN LOWER NOZZLES.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
3, 6	9.	UPPER CORE PLATE INSERTS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
2	10.	THERMOCOUPLE COLUMN, FLOW COLUMN, AND GUIDE TUBE SCREW LOCKING DEVICES.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
2	11.	ACCESSIBLE SUPPORT COLUMN, AND CORE PLATE INSERT SCREW LOCKING DEVICES.	ACCEPTED	NO APPARENT DEFECTS
1	12.	UPPER SUPPORT SKIRT TO PLATE GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	13.	UPPER SUPPORT SKIT TO FLANGE GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	14.	ACCESSIBLE GUIDE TUBE WELDS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	15.	UPPER BARREL TO FLANGE GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	16.	UPPER BARREL TO LOWER BARREL GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	17.	LOWER BARREL TO CORE SUPPORT GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 4	18.	UPPER CORE PLATE ALIGNING PIN WELDS AND BEARING SURFACES.	ACCEPTED	NO APPARENT DEFECTS
6	19.	OUTLET NOZZLE INTERFACE SURFACE CONDITION.	ACCEPTED	NO APPARENT DEFECTS
1	20.	NEUTRON SHIELD PANEL DOWEL PIN COVER PLATE WELDS.	ACCEPTED	NO APPARENT DEFECTS
1, 2	21.	NEUTRON SHIELD PANEL SCREW LOCKING DEVICES	ACCEPTED	NO APPARENT DEFECTS
10	22.	INTERFACE SURFACES AT THE SPACER PADS ALONG THE TOP AND BOTTOM ENDS OF THE NEUTRON PANELS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	23.	BAFFLE ASSEMBLY SCREW LOCKING ARRANGEMENTS AT THE TWO TOP AND THE TWO BOTTOM FORMER ELEVATIONS.	ACCEPTED	NO APPARENT DEFECTS
1, 3, 4	24.	VESSEL CLEVIS LOCKING ARRANGEMENTS AND BEARING SURFACES.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 2, 7	25.	CORE SUPPORT COLUMNS AND THEIR SCREW LOCKING DEVICES.	ACCEPTED	NO APPARENT DEFECTS
8	26.	CORE SUPPORT COLUMN ADJUSTING SLEEVES.	ACCEPTED	NO APPARENT DEFECTS
9	27.	ACCESSIBLE (2) INTRUMENTATION GUIDE COLUMN LOCKING COLLARS NEAREST THE MANWAY.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 2, 10	28.	LOCKING DEVICES AND CONTACT OF THE CURCIFORM SHAPED BOTTOM INSTRUMENTATION GUIDE COLUMNS WHERE ATTACHED TO THE CORE SUPPORT AND TIE PLATES.	NO DEFECTS	NO APPARENT DEFECTS
1, 2	29.	LOCKING DEVICES OF THE SECONDARY CORE SUPPORT BUTT COLUMNS AT THE CORE SUPPORT, TIE PLATE AND BASE PLATE.	ACCEPTED	NO APPARENT DEFECTS
1	30.	RADIAL SUPPORT KEY WELDS	ACCEPTED	NO APPARENT DEFECTS
1, 4	31.	RADIAL SUPPORT KEY LOCKING ARRANGEMENTS AND BEARING SURFACES.	NO DEFECTS	NO APPARENT DEFECTS
1, 4	32.	HEAD AND VESSEL ALIGNING PIN SCREW LOCKING DEVICES AND BEARING SURFACES.	NO DEFECTS	NO APPARENT DEFECTS
1, 2	33.	IRRADIATION SPECIMEN GUIDE SCREW LOCKING DEVICES AND DOWEL PINS	NO DEFECTS	NO APPARENT DEFECTS
6	34.	VESSEL NOZZLE INTERFACE SURFACE CONDITION.	NO DEFECTS	NO APPARENT DEFECTS

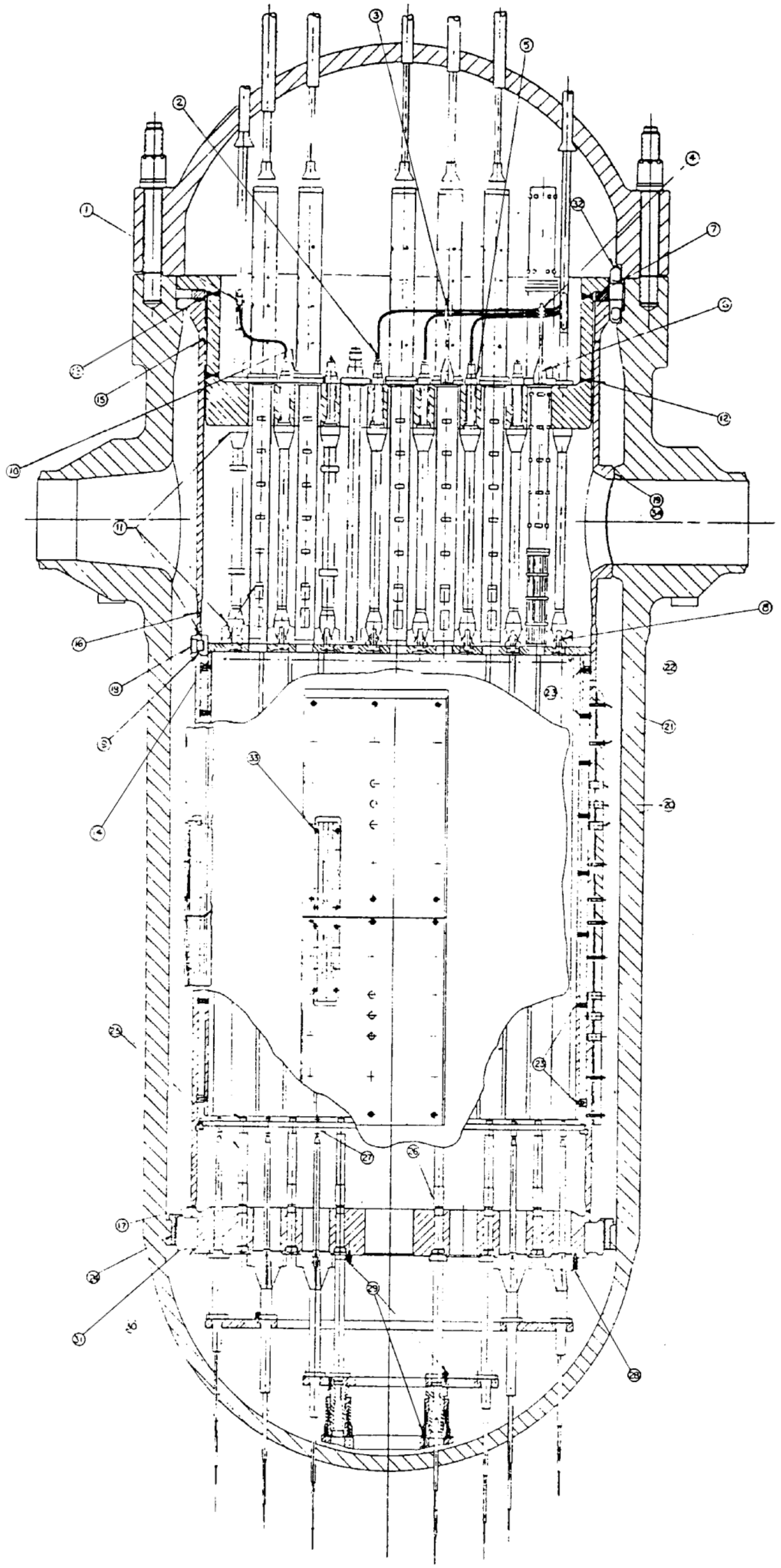
NOTES

1. VISUALLY EXAMINE WELDS USING 5-10X MAGNIFICATION. NOT CRACKS ALLOWED.
2. VERIFY THAT LOCKING DEVICES ARE CRIMPED AND UNDAMAGED.
3. VERIFY THAT INSERTS ARE SEATED (.0015 FEELER MUST NOT PASS THRU INTERFACE).
4. VISUALLY EXAMINE FACES FOR DAMAGE USING 5-10X MAGNIFICATION.
5. VERIFY THAT FITTINGS AER TIGHT.
6. VISUALLY EXAMINE INTERFACE SURFACES FOR ANY EVIDENCE OF DAMAGE.
7. VERIFY THAT ACCESSIBLE COLUMNS ARE SEATED ON LOWER CORE PLATE. A .0015 FEELER MUST NOT PASS THRU 90°.
8. VERIFY THAT ACCESSIBLE SLEEVES ARE SEATED ON COLUMNS. A .0015 FEELER MUST NOT PASS THRU 90°.
9. VERIFY THAT LOCKING COLLARS ARE TIGHT. NO MOVEMENT ALLOWED.
10. VERIFY SEATING USING A .0015 FEELER GAGE. FEELER MUST NOT PASS THRU 90°.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Unit 1
Vibration Check-out
Functional Test Inspection Data

FIGURE 3.9-1 (U1)



SEE NOTES	STEP	FEATURES TO BE EXAMINED	COMMENTS AND OBSERVATIONS BEFORE FUNCTIONAL TEST	COMMENTS AND OBSERVATIONS AFTER FUNCTIONAL TEST
1	5.	UPPER SUPPORT COLUMN NUT TO EXTENSION WELDS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
6	7.	HOLD DOWN SPRING INTERFACE SURFACE CONDITION.	NO DEFECTS	NO APPARENT DEFECTS
1	8.	ACCESSIBLE WELDS ON SUPPORT COLUMN LOWER NOZZLES.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
3, 6	9.	UPPER CORE PLATE INSERTS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
2	10.	SUPPORT COLUMN, FLOW COLUMN, AND GUIDE TUBE SCREW LOCKING DEVICES.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
2	11.	ACCESSIBLE SUPPORT COLUMN, AND CORE PLATE INSERT SCREW LOCKING DEVICES.	ACCEPTED	NO APPARENT DEFECTS
1	12.	UPPER SUPPORT SKIRT TO PLATE GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	13.	UPPER SUPPORT SKIT TO FLANGE GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	14.	ACCESSIBLE GUIDE TUBE WELDS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	15.	UPPER BARREL TO FLANGE GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	16.	UPPER BARREL TO LOWER BARREL GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	17.	LOWER BARREL TO CORE SUPPORT GIRTH WELD.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 4	18.	UPPER CORE PLATE ALIGNING PIN WELDS AND BEARING SURFACES.	ACCEPTED	NO APPARENT DEFECTS
6	19.	OUTLET NOZZLE INTERFACE SURFACE CONDITION.	ACCEPTED	NO APPARENT DEFECTS
1	20.	NEUTRON SHIELD PANEL DOWEL PIN COVER PLATE WELDS.	ACCEPTED	NO APPARENT DEFECTS
1, 2	21.	NEUTRON SHIELD PANEL SCREW LOCKING DEVICES	ACCEPTED	NO APPARENT DEFECTS
10	22.	INTERFACE SURFACES AT THE SPACER PADS ALONG THE TOP AND BOTTOM ENDS OF THE NEUTRON PANELS.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1	23.	BAFFLE ASSEMBLY SCREW LOCKING ARRANGEMENTS AT THE TWO TOP AND THE TWO BOTTOM FORMER ELEVATIONS.	ACCEPTED	NO APPARENT DEFECTS
1, 3, 4	24.	VESSEL CLEVIS LOCKING ARRANGEMENTS AND BEARING SURFACES.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 2, 7	25.	CORE SUPPORT COLUMNS AND THEIR SCREW LOCKING DEVICES.	ACCEPTED	NO APPARENT DEFECTS
8	26.	CORE SUPPORT COLUMN ADJUSTING SLEEVES.	ACCEPTED	NO APPARENT DEFECTS
9	27.	ACCESSIBLE (2) INTRUMENTATION GUIDE COLUMN LOCKING COLLARS NEAREST THE MANWAY.	NO APPARENT DEFECTS	NO APPARENT DEFECTS
1, 2, 10	28.	LOCKING DEVICES AND CONTACT OF THE CURCIFORM SHAPED BOTTOM INSTRUMENTATION GUIDE COLUMNS WHERE ATTACHED TO THE CORE SUPPORT AND TIE PLATES.	NO DEFECTS	NO APPARENT DEFECTS
1, 2	29.	LOCKING DEVICES OF THE SECONDARY CORE SUPPORT BUTT COLUMNS AT THE CORE SUPPORT, TIE PLATE AND BASE PLATE.	ACCEPTED	NO APPARENT DEFECTS
1	30.	RADIAL SUPPORT KEY WELDS	ACCEPTED	NO APPARENT DEFECTS
1, 4	31.	RADIAL SUPPORT KEY LOCKING ARRANGEMENTS AND BEARING SURFACES.	NO DEFECTS	NO APPARENT DEFECTS
1, 4	32.	HEAD AND VESSEL ALIGNING PIN SCREW LOCKING DEVICES AND BEARING SURFACES.	NO DEFECTS	NO APPARENT DEFECTS
1, 2	33.	IRRADIATION SPECIMEN GUIDE SCREW LOCKING DEVICES AND DOWEL PINS	NO DEFECTS	NO APPARENT DEFECTS
6	34.	VESSEL NOZZLE INTERFACE SURFACE CONDITION.	NO DEFECTS	NO APPARENT DEFECTS

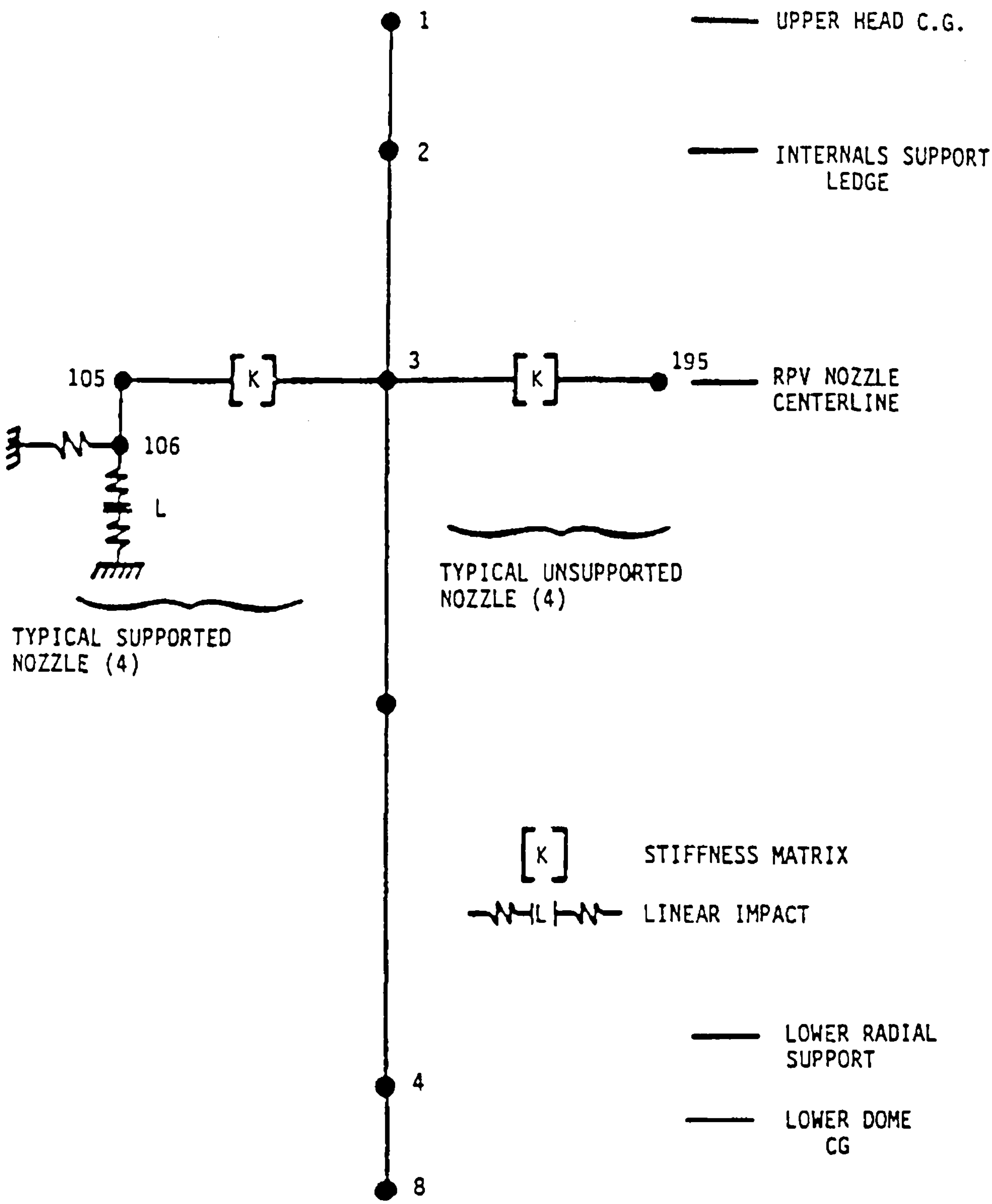
NOTES

1. VISUALLY EXAMINE WELDS USING 5-10X MAGNIFICATION. NOT CRACKS ALLOWED.
2. VERIFY THAT LOCKING DEVICES ARE CRIMPED AND UNDAMAGED.
3. VERIFY THAT INSERTS ARE SEATED (.0015 FEELER MUST NOT PASS THRU INTERFACE).
4. VISUALLY EXAMINE FACES FOR DAMAGE USING 5-10X MAGNIFICATION.
5. VERIFY THAT FITTINGS AER TIGHT.
6. VISUALLY EXAMINE INTERFACE SURFACES FOR ANY EVIDENCE OF DAMAGE.
7. VERIFY THAT ACCESSIBLE COLUMNS ARE SEATED ON LOWER CORE PLATE. A .0015 FEELER MUST NOT PASS THRU 90°.
8. VERIFY THAT ACCESSIBLE SLEEVES ARE SEATED ON COLUMNS. A .0015 FEELER MUST NOT PASS THRU 90°.
9. VERIFY THAT LOCKING COLLARS ARE TIGHT. NO MOVEMENT ALLOWED.
10. VERIFY SEATING USING A .0015 FEELER GAGE. FEELER MUST NOT PASS THRU 90°.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Unit 2
Vibration Check-out
Functional Test Inspection Data

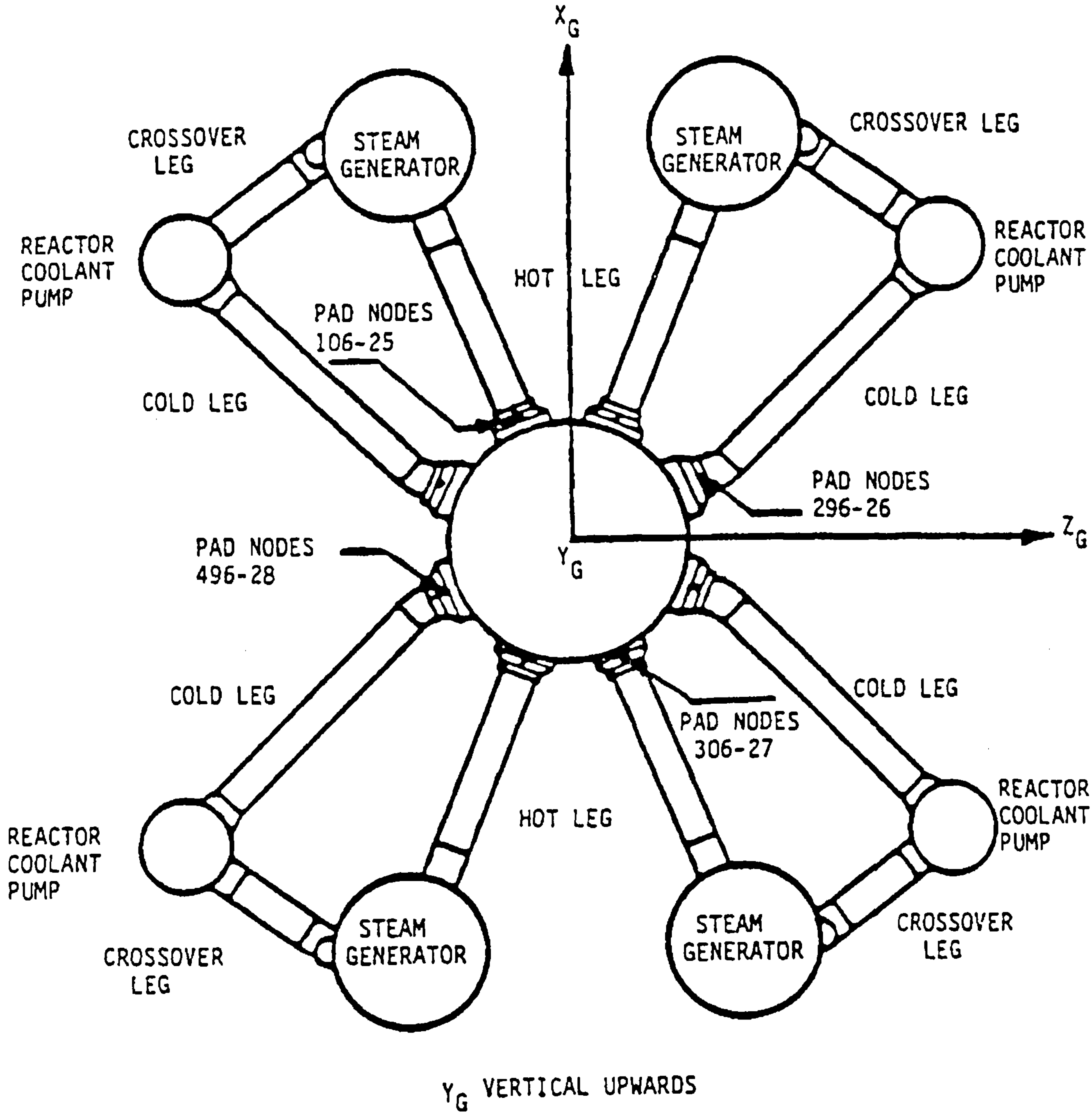
FIGURE 3.9-1 (U2)



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RPV Shell Submodel

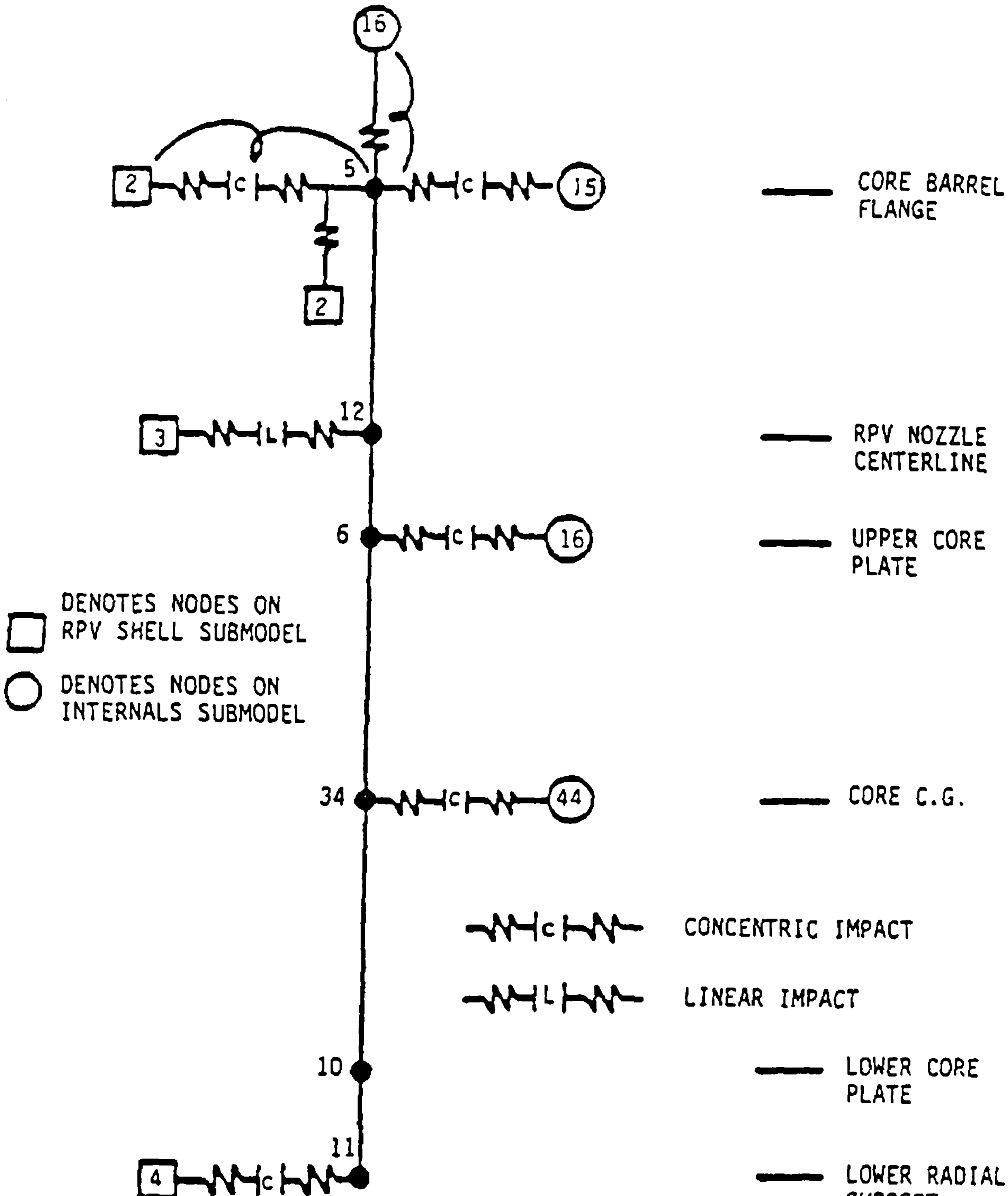
FIGURE 3.9.2A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RPB Support Pads and
WECAN Global Coordinates

FIGURE 3.9-2B



WATTS BAR NUCLEAR PLANT
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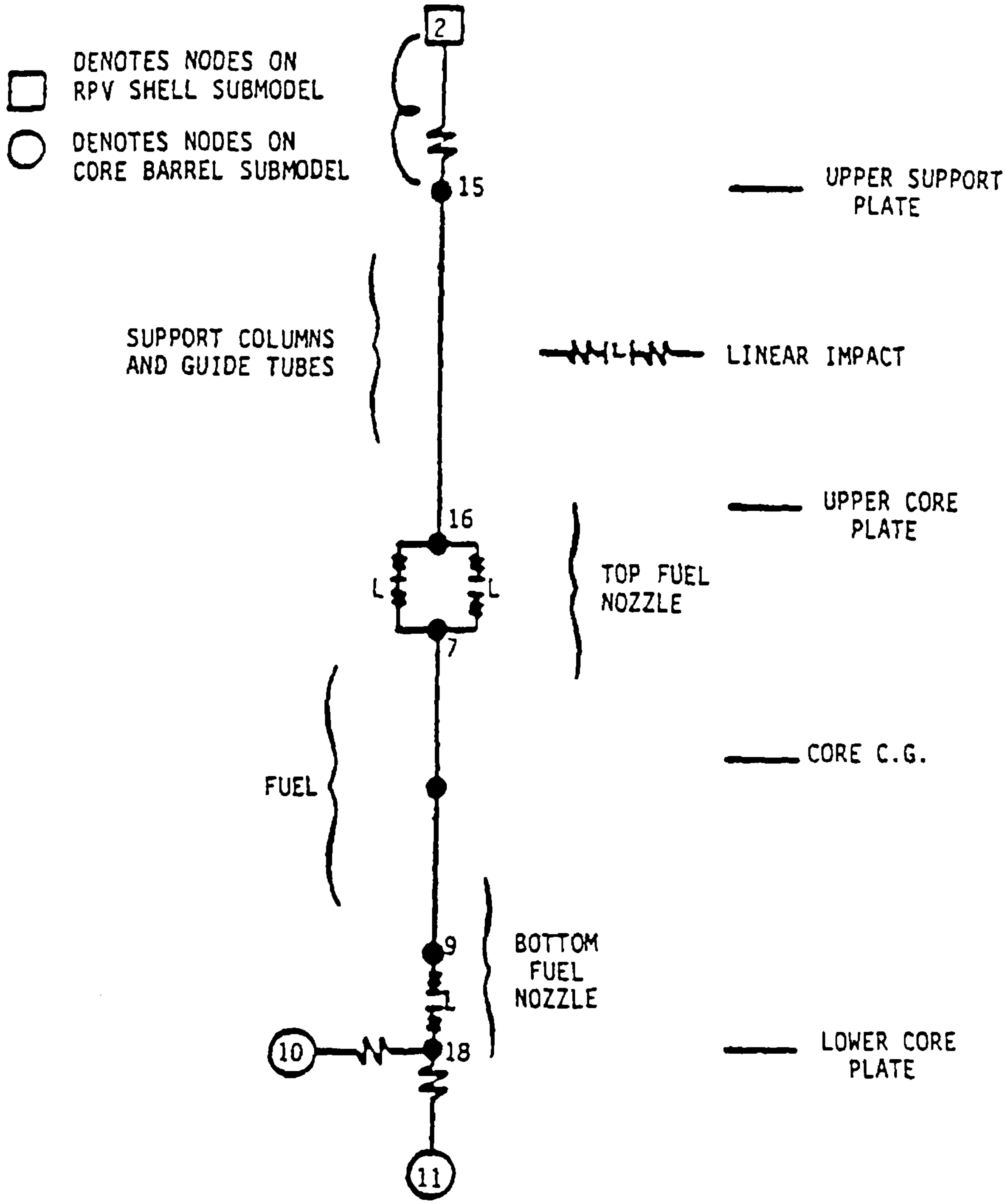
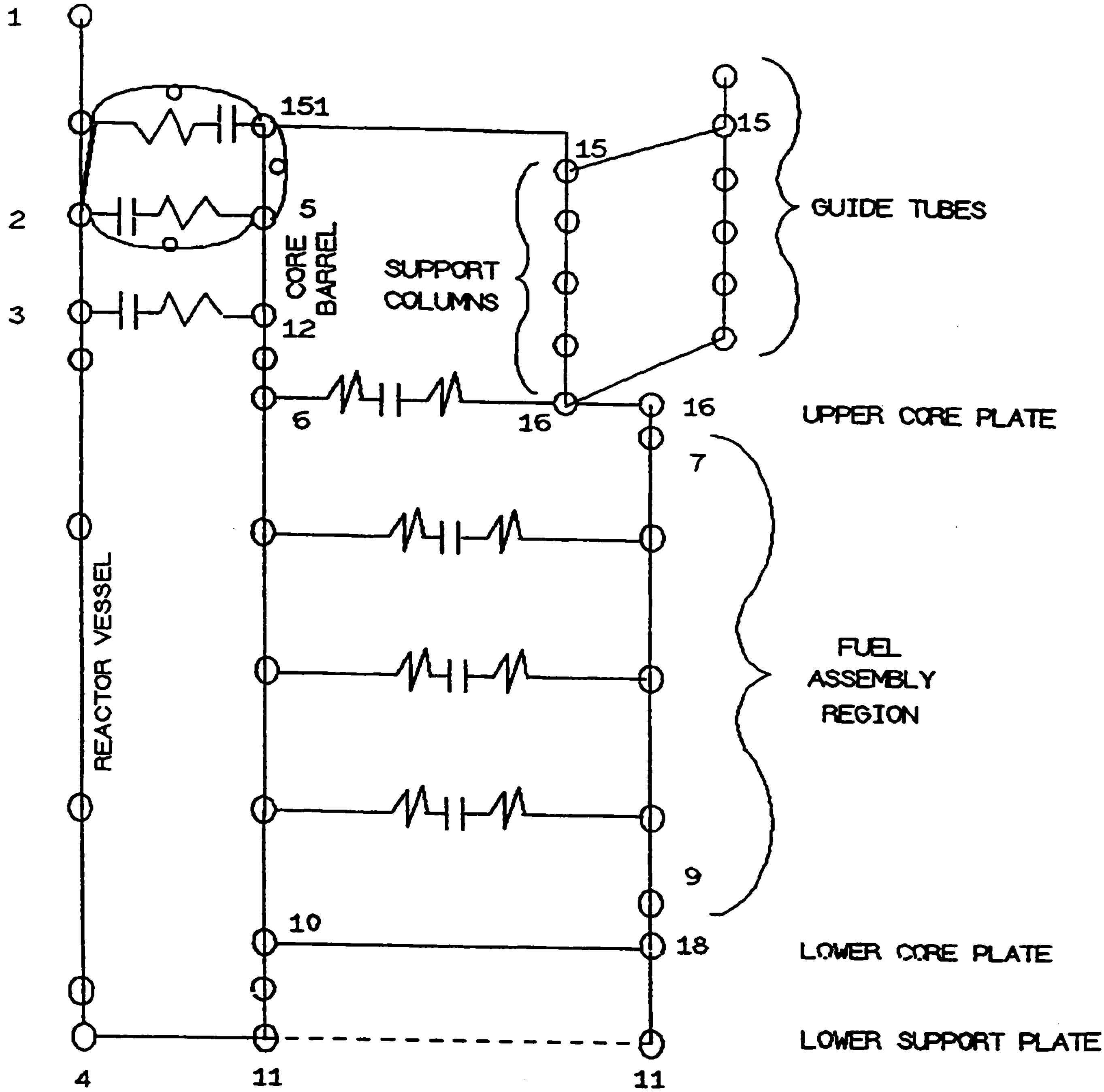
Core Barrel Submodel

FIGURE 3.9-3A



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Internals Submodel FIGURE 3.9-3B



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Reactor Pressure Vessel
System Model

FIGURE 3.9-3C

3.10 SEISMIC DESIGN OF CATEGORY 1 INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Seismic Category I instrumentation and electrical equipment for the Watts Bar Nuclear Plant was either furnished by Westinghouse or purchased by TVA. TVA's seismic qualification program for instrumentation and electrical equipment at Watts Bar is based on the requirements of IEEE 344-1971 and the NRC Standard Review Plan, Section 3.10 (specifically, acceptance criteria for plants with Construction Permit application docketed before October 27, 1972) as discussed in Section 3.7.3.16.

Instrumentation and electrical equipment was purchased in assemblies except for local panel instrumentation as described in Section 3.10.1. TVA provided the vendor with a required response spectrum as a part of the equipment specification in order that the vendor could qualify the equipment. The derivation of the response spectrum is described in Section 3.7.

3.10.1 Seismic Qualification Criteria

TVA Supplied

Class 1E Power Equipment

Table 3.10-1 lists the procurement packages for Class 1E power equipment. TVA's seismic qualification criteria is based on IEEE 344-1971 or IEEE 344-1975 as discussed above.

The capability of ESF circuits and the Class 1E system to withstand seismic disturbances is established by seismic analysis and/or testing of each system component. The qualification criteria used in the design of Seismic Category I electrical equipment are given below.

1. Safety-related equipment designated as Seismic Category I, when subjected to the vertical and horizontal acceleration of the safe shutdown earthquake (SSE), shall perform as follows:

    a. Equipment shall retain its structural integrity during and after the seismic event.

    b. Equipment shall be capable of performing its design function during and after the seismic event.

    c. Maximum displacement of the equipment during the earthquake shall not cause loss of function of any externally connected parts, such as conduit, cable, or bus connections.

Equipment anchorage/support design is discussed in Section 3.10.3.1. Other considerations for the seismic qualification of Category I electrical equipment are described in Section 3.7.3.

Local Instrumentation

TVA supplied instruments were classified as Seismic Category I in accordance with the system served and instrument function. Seismic Category I systems are qualified in accordance with IEEE-344-1971 or 1975, as applicable, and are listed in Table 3.10-1.

Type testing for seismic qualification has been performed on the Seismic Category I instruments. The active instruments are capable of performing their function during and following a SSE. They are qualified to the response acceleration which exceeds the response of the support structure. Tests and/or analyses were conducted on critical rack-instrumentation configurations to confirm the conservatism of the seismic test level for the instruments.

Westinghouse Supplied (Unit 1 Only)

The reactor trip system, and engineered safety features actuation system are designed so that they are capable of providing the necessary protective actions during and after a SSE; therefore, the reactor protection system is capable of tripping the reactor during and after a SSE. The engineered safety features actuation system and the safety features systems are designed to initiate their protective functions during and after an SSE.

The following list identifies the instrumentation and electrical equipment requiring seismic qualification by the supplier of the Nuclear Steam Supply System (NSSS).

1. Foxboro Model E-11 pressure transmitter and Model E-13 differential pressure transmitter.

2. Foxboro Process Control Equipment cabinets.

3. Westinghouse Solid-State Protection System cabinets.

4. Nuclear Instrumentation System cabinets.[1]

5. Safeguards Test Racks.

6. Resistance Temperature Detectors.

7. Power range Neutron Detectors.

8. Reactor trip breakers.

9. Barton Models 332 and 386 differential pressure transmitters.

10. Eagle-21 Process Protection System

[1] NOTE: Unit 1 Source Intermediate Range Electronics and Detectors were replaced with Gamma Metrics Equipment (DCN 03206). The equipment was placed into the Nuclear Instrumentation System Cabinets and subsequently qualified by TVA calculation WCG-ACQ-0104.

Seismic qualification testing of Items 1 through 9 is documented in References [1] through [10]. Reference [10] presents the theory and practice, as well as justification, for the use of single axis sine beat test inputs used in the seismic qualification of electrical equipment. In addition, it is noted that Westinghouse has conducted a seismic qualification "Demonstration Test Program" (reference Letter NS-CE-692, C. Eicheldinger (W), to D. B. Vassallo (NRC), 7/10/75) to confirm equipment operability during a seismic event. This program is documented in References [12] through [15] (Proprietary) and References [16] through [20] (Non-Proprietary). Seismic qualification testing of Item 10 to IEEE 344-1975 is documented in References [21], [22], and [23].

The Watts Bar Nuclear Plant complies with paragraph IV, "Conclusions and Regulatory Positions" of the "Mechanical Engineering Branch Report on Seismic Audit of Westinghouse Electrical Equipment." All topical reports have been completed and are included in the reference list. The non-proprietary topical reports have been referenced as a group above. The structural capability of the NIS rack is discussed in References [14], [19], and [24].

The demonstration test program, in conjunction with the justification for the use of single axis sine beat tests, presented in WCAP-8373, and the original tests, documented in References [2] through [10], meet the requirements of IEEE Standard 344-1975 "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" for the seismic qualification of Westinghouse electrical equipment outside of containment. Environmental qualification for equipment inside of containment is described in Section 3.11.

The peak accelerations used in the type testing are conservative values that are checked against those derived by structural analyses of SSE loadings of the Watts Bar Nuclear Plant. For the SSE there may be permanent deformation of the equipment provided that the capability to perform its function is maintained.

Resistance temperature detectors used to sense the temperature in the main coolant loops are rigid, ruggedly built devices designed to withstand the high temperature, high pressure, and flow vibration induced acceleration forces which they are subjected to when installed in the coolant loops. The natural frequency of this device is designed to be higher than the frequencies associated with the seismic disturbance. Seismic qualification of these resistance temperature detectors is presented in Reference [9].

The nuclear instrumentation system power range neutron detector has been vibration tested in both the transverse (horizontal) direction and longitudinal (vertical) direction at acceleration levels greater than those expected during a seismic disturbance at the Watts Bar Nuclear Plant site. Detector current measurements were made during the tests and neutron sensitivity, resistance, and capacitance checks were made after the test. No significant changes were seen. There was no mechanical damage to the detector.

Typical switches and indicators which could defeat automatic operation of a required safety function have been tested to determine their ability to withstand seismic excitation without malfunction. The control boards are stiff and past experience indicates that the amplification due to the board structure is sufficiently low so that the acceleration seen by the device is considerably less than that used in testing.

All safety-related instruments of the reactor protection system and the engineered safety feature circuits are mounted on Seismic Category I supporting structures. They are designed to withstand horizontal and vertical accelerations at each floor level for the SSE. The instrument supporting structures located throughout the plant (local panels) have been standardized in design and have been seismically qualified by testing. The local panels were tested using response spectra for the highest elevation on which any of these panels are mounted. The test criteria were in accordance with IEEE 344-1971 or IEEE 344-1975, as discussed above.

Where space requirements preclude the use of the standard local panels, a small wall-mounted panel is used. This panel is qualified, to the same criteria as the local panels, by analysis and/or test.

Reactor Protection System (Unit 2 only)

The reactor trip system, and engineered safety features actuation system are designed so that they are capable of providing the necessary protective actions during and after a SSE; therefore, the reactor protection system is capable of tripping the reactor during and after a SSE. The engineered safety features actuation system and the safety features systems are designed to initiate their protective functions during and after an SSE.

The following list identifies the instrumentation and electrical equipment requiring seismic qualification.

1. Foxboro Process Control Equipment cabinets.
2. Westinghouse Solid-State Protection System and cabinets.
3. Nuclear Instrumentation System cabinets.
4. Nuclear Instrumentation System Power Range Electronics.
5. Safeguards Test Racks.
6. esistance Temperature Detectors.
7. Power range Neutron Detectors.
8. Reactor trip breakers.
9. Cameron Model 764 differential pressure transmitters and Model 763A pressure transmitters.
10. Eagle-21 Process Protection System
11. Nuclear Instrumentation Source and Intermediate Range Electronics
12. Combined Source and Intermediate Range Neutron Detectors
13. Process Transmitters (not supplied by Westinghouse)

Seismic qualification testing/analysis of Items 1 through 9 is documented in References [1]

through [10] and [26]. Reference [10] presents the theory and practice, as well as justification, for the use of single axis sine beat test inputs used in the seismic qualification of electrical equipment. In addition, it is noted that Westinghouse has conducted a seismic qualification "Demonstration Test Program" (reference Letter NS-CE-692, C. Eicheldinger (W), to D. B. Vassallo (NRC), 7/10/75) to confirm equipment operability during a seismic event. This program is documented in References [12] through [14] (Proprietary) and References [16] through [19] (Non-Proprietary). Seismic qualification testing of Item 10 to IEEE 344-1975 is documented in References [21], [22], [23], [31], and [32]. Reference [26] documents the Westinghouse qualification by analysis of the Nuclear Instrumentation System cabinet 2-M-13 with Gamma Metrics Source and Intermediate Range hardware installed.

The Watts Bar Nuclear Plant complies with paragraph IV, "Conclusions and Regulatory Positions" of the "Mechanical Engineering Branch Report on Seismic Audit of Westinghouse Electrical Equipment." All topical reports have been completed and are included in the reference list. The non-proprietary topical reports have been referenced as a group above. The structural capability of the NIS cabinets and power range electronics is discussed in References[14], [19], [24], and [26].

The demonstration test program, in conjunction with the justification for the use of single axis sine beat tests, presented in WCAP-8373, and the original tests, documented in References [2] through [10], meet the requirements of IEEE Standard 344-1975 "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" for the seismic qualification of Westinghouse electrical equipment outside of containment. Environmental qualification for equipment inside of containment is described in Section 3.11.

The peak accelerations used in the type testing are conservative values that are checked against those derived by structural analyses of SSE loadings of the Watts Bar Nuclear Plant. For the SSE there may be permanent deformation of the equipment provided that the capability to perform its function is maintained.

Resistance temperature detectors used to sense the temperature in the main coolant loops are rigid, ruggedly built devices designed to withstand the high temperature, high pressure, and flow vibration induced acceleration forces which they are subjected to when installed in the coolant loops. The natural frequency of this device is designed to be higher than the frequencies associated with the seismic disturbance. Seismic qualification of these resistance temperature detectors is presented in Reference [9].

The nuclear instrumentation system power range neutron detector has been vibration tested in both the transverse (horizontal) direction and longitudinal (vertical) direction at acceleration levels greater than those expected during a seismic disturbance at the Watts Bar Nuclear Plant site. Detector current measurements were made during the tests and neutron sensitivity, resistance, and capacitance checks were made after the test. No significant changes were seen. There was no mechanical damage to the detector.

Typical switches and indicators which could defeat automatic operation of a required safety function have been tested to determine their ability to withstand seismic excitation without malfunction. The control boards are stiff and past experience indicates that the amplification due to the board structure is sufficiently low so that the acceleration seen by the device is considerably less than that used in testing.

All safety-related instruments of the reactor protection system and the engineered safety feature circuits are mounted on Seismic Category I supporting structures. They are designed to withstand

horizontal and vertical accelerations at each floor level for the SSE. The instrument supporting structures located throughout the plant (local panels) have been standardized in design and have been seismically qualified by testing. The local panels were tested using response spectra for the highest elevation on which any of these panels are mounted. The test criteria were in accordance with IEEE 344-1971 or IEEE 344-1975, as discussed above.

Where space requirements preclude the use of the standard local panels, a small wall-mounted panel is used. This panel is qualified, to the same criteria as the local panels, by analysis and/or test.

Seismic qualification testing of the Gamma-Metrics supplied source and intermediate range neutron detection system (Items 11 and 12 including all interconnections) is documented in Reference [25].

Seismic qualification testing of the protection system process transmitters not supplied by Westinghouse (Item 13) is documented in References [27] through [30].

Seismic qualification on testing of safety related radiation monitors is documented in References [33] through [39].

3.10.2 Methods And Procedures For Qualifying Electrical Equipment And Instrumentation

For the seismic qualification methods of selected Category I electrical equipment and instrumentation, see Tables 3.10-1 and 3.10-2.

Instrumentation

The seismic type testing performed by the NSSS supplier (Westinghouse) is described in References [1] through [10], and [21]. For References [1] through [10], the test method used was the sine beat procedure described in IEEE 344-1971 and Reference [11]. In addition, as noted in Section 3.10.1, Westinghouse conducted a "Demonstration Test Program" which, when considered in conjunction with the tests presented in References [11] through [13], results in meeting the requirements of IEEE 344-1975.

Supporting Structures (Panels, Racks, Cabinets, and Boards)

The qualification of the supporting structures for Seismic Category I instruments has been accomplished by either analysis or testing. The method commonly used is testing under simulated conditions. All tests by TVA before September 1, 1974 on these supporting structures were similar. The support structure was mounted on a vibration generator in a manner that simulated the intended service mounting. The vibratory forces were applied to each of the three major perpendicular axes independently. Maximum service dead loads were simulated. Selected points were monitored to establish amplification of loads. Testing was done at the structure's resonant frequencies. The resonant frequencies were determined by an exploratory test using a sinusoidal steady-state input of low amplitude, (two continuous sweeps from 1 to 33 Hz at a rate of 1 octave per minute). The qualification test was conducted using the sine beat method at the resonant frequencies using the appropriate acceleration input as determined from the building response acceleration spectra. Also, reference Section 3.7.3.16 for additional details.

Later qualification tests typically used multi-frequency time history input motion for which the test response spectra enveloped the required response spectra in accordance with IEEE 344-1975 guidelines.

3.10.3 Methods of Qualifying TVA-Designed Supports for Electrical Equipment Instrumentation and Cables

The methods and procedures of design and analysis or testing of electrical equipment and instrumentation supports, cable trays, cable tray supports, conduit, conduit supports, and conduit banks are provided in the following sections.

3.10.3.1 Electrical Equipment and Instrumentation Assemblies

TVA-designed supports and anchorage for Category I electrical equipment assemblies ensure compatibility with the equipment seismic qualifications test or analysis as described in Section 3.7.3.16.5. Design of these supports is in accordance with Section 3.8.4.5.2.

All floor/wall mounted Category I electrical equipment assemblies such as battery racks, instrument racks, and control consoles are attached by TVA to the building structure. The attachments are made by bolting or welding to structural members. Anchorages to concrete are made by welding to embedded plates cast in the concrete with stud anchors, or by bolting to anchors set in the hardened concrete (self-drilling bolts, wedge bolts, undercut expansion anchors, or grouted anchors).

3.10.3.2 Cable Trays and Supports

3.10.3.2.1 Cable Trays

Cable trays containing Class 1E cables located in Category I structures are considered safety-related and are designed to resist gravity and SSE forces.

Cable tray acceptance criteria are derived from testing. A factor of safety of 1.25 against the tested capacity, is maintained for the vertical moment. A ductility factor of 3 is used to establish tray capacity in the transverse direction. These limits are used in an interaction equation to evaluate tray sections for the SSE loading condition. Seismic loadings are developed based on the applicable response spectra. In addition, all trays are evaluated to ensure a minimum factor of safety of 3 against test capacity for dead load only.

Figure 3.10-1 defines the orientation of the transverse and vertical moments.

Cable tray X and T fittings are evaluated for vertical loading to ensure a minimum factor of safety of 1.25 against the formation of a first hinge.

All other cable tray components are evaluated using AISI or AISC allowables (as applicable) with increase factors as allowed by Standard Review Plan Section 3.8.4. Where test data is used to establish capacities of bolted parts, a factor of safety of 1.5 is maintained against the ultimate test load for the SSE loading condition.

3.10.3.2.2 Supports

All cable tray supports located in Category I structures are designated Seismic Category I and designed to resist seismic forces applied to the weight of trays and cables. Each support in Category I structures is designed independently to support its appropriate length of tray. Seismic load inputs are based on the methods described in Section 3.7 and the damping requirements described in Table 3.7-2.

Trays are designed to carry a load of 30 pounds per square foot (which is equivalent to 45 pounds per linear foot for an 18 inch wide tray) and an additional construction load of 30 pounds per linear foot on the top tray. Actual tray loading may be used on a case by case basis.

For load combinations and allowables applicable for cable tray supports, see Table 3.10-5.

Welding was in accordance with the American Welding Society (AWS), "Structural Welding Code," AWS D1.1 with revision 1-73 and 1-74, except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders. Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS. Visual inspection of structural welds will meet the minimum requirements of NCIG-01 and NCIG-02 as specified on the design drawings or other design output. Inspectors performing visual examination to the criteria of NCIG-01 are trained in the subject criteria.

3.10.3.3 Conduit and Supports

3.10.3.3.1 Conduit

Conduit containing Class 1E cables located in Category I structures are considered safety-related and designed to resist gravity and SSE forces applied to the conduit and cable. The seismic qualification utilizes the same analysis methods as Seismic Category I subsystems described in Section 3.7.3 and limits allowable stress to 90% of the yield stress of the conduit material. The applicable damping requirements are defined in Table 3.7-2.

3.10.3.3.2 Supports

All conduit supports in Category I structures are designed to resist gravity and SSE forces applied to the conduit and cables. Supports for conduit containing Class 1E cables are designated Category I and stresses are limited to 90% of the yield stress of the material involved. Seismic load inputs are based on methods described in Section 3.7 and damping requirements are defined in Table 3.7-2. Supports for conduit containing only non-Class 1E cables are designated Category I(L) and designed and constructed to preclude a failure which could reduce the ability of Category I structures, systems, and components to perform their intended safety function.

Welding was in accordance with the American Welding Society (AWS), "Structural Welding Code," AWS D1.1 with revision 1-73 and 1-74, except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders. Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS. Visual inspection of structural welds will meet the minimum requirements of NCIG-01 and NCIG-02 as specified on the design drawings or other design output. Inspectors performing visual examination to the criteria of NCIG-01 are trained in the subject criteria.

3.10.3.4 Conduit Banks

The Category I underground electrical conduit banks, which run from the Auxiliary Building to the Diesel Generator Building and to the Intake Pumping Station, were analyzed for seismic loads by the method outlined in Section 3.7.2.1.3. The conduit banks are designed in accordance with Section 3.8.4.2.

3.10.4 Operating License Review

3.10.4.1 TVA Supplied Instrumentation and Electrical Equipment

The results of the seismic qualification program for the Watts Bar Nuclear Plant described in Section 3.10.1, 3.10.2, and 3.10.3 are summarized by the following listing for Class 1E equipment and by Tables 3.10-1, 3.10-3, and 3.10-4.

	Equipment	TVA Contract No.
AC Auxiliary Power System	6.9kV Switchgear	74C2-84376
	6.9kV Shutdown Logic Relay Panels	75K2-85354
	6.9kV/480V Transformer	74C2-84647
	480V Switchgear	74C2-84647
	480V Motor Control Centers	74C5-84646
	480V Distribution Panelboards for Pressurizer Heater Backup Groups	75K3-86476
	Diesel Generator	74C63-83090
	Transfer Switches	75K5-87048
125V DC Class 1E System	Transfer Switches	75K5-87048
	Spare Transfer Switches	00072332
	Battery Chargers	00072332
	Vital Batteries	76K3-85763
	Vital Battery Boards	75C2-85281
120V AC Class 1E System	AC Vital Instrument Power Boards	74C4-85216
	Vital Inverters	34327, 69414

Miscellaneous Class 1E Equipment	Electrical Penetrations 7	76K61-87064
	BOP I&C Equipment	Multiple Contract No.
	Emergency DC Lighting	75C2-85737-1

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1. Morrone, A., "Seismic Vibration Testing with Sine Beats," WCAP-7558, October, 1971.

2. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment," WCAP-7397-L (Proprietary) and WCAP-7817 (NonProprietary), December, 1971.

3. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7817, Supplement 1, December, 1971.

4. Potochnik, L. M., "Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP-7817, Supplement 2, December, 1971.

5. Vogeding, E. L., "Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)," WCAP-7817, Supplement 3, December, 1971.

6. Reid, J. B., "Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)," WCAP-7817, Supplement 4, November, 1972.

7. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel)," WCAP-7817, Supplement 5, March, 1974.

8. Faust III, C. E., Tang, D. T., and Yalich, M., "Equipment Qualification Test Report Reactor Trip Switchgear," WCAP 8687, Supplement 2-E20B, Revision 1, July, 1981.

9. Resistance Temperature Detectors - Equipment Qualification Data Packages: EQDP-ESE-6, Revision 7; EQDP-ESE-7, Revision 7; and, EQDP-ESE-66A, Revision 1 - WCAP-8587.

10. Fischer, E. G. and Jerecki, S. J., "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974," WCAP-8373, August, 1974.

11. Wyle Laboratory T. R. 42377-1, "Seismic Simulation Test Program on Instrument Rack."

12. Jareck, S. J., "General Method of Developing Multifrequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary) September, 1975.

13. Jareck, S. J. and Vogeding, E. L., "Multifrequency and Direction Seismic Testing of Relays," WCAP-8673 (Proprietary) December, 1975.

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15. Jareck, S. J. Coslow B. J., Ellis, A. E., and Miller, R. B., "Seismic Operability Demonstration Testing of the Foxboro H-Line Series Process Instrumentation System Bistables," WCAP-8848 (Proprietary) November, 1976.

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22. WCAP-8687, Supplement 2 - E69B, "Equipment Qualification Test Report, Eagle 21 Process Protection System Components" (Proprietary), February 1990.

23. WCAP-8687, Supplement 2 - E69C, "Equipment Qualification Test Report, Eagle 21 Process Protection System Components" (Proprietary), February 1991.

24. System Description, N3-92-4003, "Neutron Monitoring System".

25. Thermo Fisher Scientific Qualification Report No. 864, Rev. 0, Class 1E Qualification of Source Range Intermediate Range and Wide Range Channels.

26. Westinghouse Report EQ-EV-39-WBT, Seismic Evaluation of Nuclear Instrumentation System Console 2-M-13 with Gammametrics Equipment for Watts Bar Unit 2, Revision 1, March 2009.

27. Rosemount D8400102 Rev. F, Qualification Report for Pressure Transmitter Model 1154.

28. Rosemount D8700096 Rev. I, Qualification Report for Rosemount Model 1154 Series H Pressure Transmitter.

29. Ametek Report No. TR-1136, Qualification Documentation Review Package for Ametek Aerospace Gulton-Statham Products Nuclear Qualified Pressure Transmitter Series Enveloping --- Gage Pressure Transmitter Series PG 3200, Differential Pressure Transmitter Series PO 3200 Differential High Pressure Transmitter Series PDH 3200, Draft Range Pressure Transmitter Series DR 3200, Remote Diaphragm Seal Differential Pressure Transmitter Series PO 3218, Remote Diaphragm Seal Differential High Pressure Transmitter Series PDH 3218.

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32. EQLR-126, Qualification of Eagle 21 AC Distribution Panel Circuit Breaker (Unit 2 only).

33. General Atomics Electronic Systems 04508905-1SP, Qualification Test Report Supplement, RM-1000 Upgrade.

34. General Atomics Electronic Systems 04508905-2SP, Qualification Test Report Supplement, I-F Converter Upgrades.

35. General Atomics Electronic Systems 04038903-7SP, Qualification Basis for 04034101 (2-RE-90-271, 272, 273 & 274).

36. General Atomics Electronic Systems 04038903-QSR, Qualification Summary Report for Watts Bar Nuclear Plant Unit 2 Replacement Radiation Monitors.

37. General Atomics Electronic Systems 04038903-1SP, Qualification Basis for 04031101-001 (2-RE-90-130 &131).

38. General Atomics Electron Systems 04038903-2SP, Qualification Basis for 04031301-001 (2-RE-90-106).

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TABLE 3.10-1 (Sheet 1 of 5)

WBNP INSTRUMENTATION AND ELECTRICAL EQUIPMENT
SEISMIC QUALIFICATION SUMMARY

EQUIPMENT	LOCATION BLDG/EL*	TVA CONTRACT NO.	VENDOR	SEISMIC QUALIFICATION CRITERIA	QUALIFICATION METHOD	TEST METHOD	TEST LAB
ELECTRICAL PENETRATIONS	R-(ALL LEVELS)	76K6I-87064	CONAX CORP	IEEE 344-1971	ANALYSIS		
125V DC VITAL BATTERIES	A-757	76K3-85763	GOULD IND.	IEEE 344-1975; IEE P535, DRAFT 2 DATED NOV. 15, 1974	TEST	RANDOM, BIAXIAL, MULTIFREQUENCY	WYLE LAB
TRANSFER SWITCHES	A-772 A-757	75K5-87048	B-K ELECT PROD	IEEE 344-1971	TEST	SINGLE AXIS, SINE BEAT	AERO NAU LAB
SPARE TRANSFER SWITCHES	A-772	00072332	AMETEK SCI	IEEE 344-1975	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
125V DC VITAL CHARGERS	A-772	00072332	AMETEK SCI	IEEE 344-1975	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
125V DC BATTERY BOARDS	A-757	75C2-85281	WESTINGHOUSE	IEEE 344-1971, ENCLOSURE NO. 5	TEST	SINE BEAT, BIAXIAL, FOUR POSITIONS	WESTINGHOUSE
120V AC VITAL INVERTERS	A-772	34327, 69414	AMETEK SCI	IEEE 344-1971, DRAFT 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
120 AC VITAL INSTR. BOARDS	A-757	74C4-85216	WESTINGHOUSE	IEEE 344-1971, ENCLOSURE NO. 5	TEST	SINE BEAT, BIAXIAL, FOUR POSITIONS	WESTINGHOUSE
6.9KV SD BD LOGIC PANELS	A-757	75K2-85354	H. K. PORTER	IEEE 344, R3 (FEB. 15, 1974)	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
6.9KV SD BDS	A-757	74C2-84376	G. E.	IEEE 344, DRAFT REV. 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
480V SD BDS, TRANSFORMERS AND PRESS. HTR. TRANSFORMERS	A-772 A-782	74C2-84647	WESTINGHOUSE	IEEE 344, R3, AND ENCLOSURE NO. 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
480 DISTRIBUTION PANELBOARDS FOR PRESSURIZER HEATER BACKUP GROUP	A-782	75K3-86476	EL TEX	IEEE 344, DRAFT R5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
480V MOTOR CONTROL CENTERS	A-772 A-757	74C5-84646	ITE	IEEE 344, DRAFT 3 (FEB. 15, 1974)	TEST	SINE BEAT	WYLE LAB

*R-REACTOR BLDG.
A-AUXILIARY BLDG
C-CONTROL BLDG.
D-DIESEL GEN. BLDG.

TABLE 3.10-1 (Sheet 2 of 5)

WBNP INSTRUMENTATION AND ELECTRICAL EQUIPMENT
SEISMIC QUALIFICATION SUMMARY

EQUIPMENT	LOCATION BLDG/EL*	TVA CONTRACT NO.	VENDOR	SEISMIC QUALIFICATION CRITERIA	QUALIFICATION METHOD	TEST METHOD	TEST LAB
CONTROL INSTRUMENT LOOPS	MULTIPLE LOCATIONS	73C3-92784	BAILEY METER CO	IEEE 344-1971	TEST	SINGLE AXIS	ACTION ENVIRON-MENTAL TESTING CORP.
INSTRUMENTAION AND CONTROLS	MULTIPLE LOCATIONS	77K3-87352	ROBERTSHAW CONTROLS CORP	IEEE 344-1974	TEST	RANDOM FREQUENCY, MULTIAXIS	WYLE LAB
FABRICATION OF LOCAL PANELS AND INSTALLATION OF INSTRUMENTS PRESSURE GAUGES, PRESSURE SWITCHES, & LEVEL SWITCHES	MULTIPLE LOCATIONS SEE NOTE 1	73C38-92800	H. K. PORTER	IEEE 344-1974 (DRAFT REVISION TO IEEE 344-1971)	TEST	MULTIFREQUENCY, BIAXIAL	WYLE LAB
STANDBY POWER SYSTEM	D-742	74C63-83090	MORRISON-KNUDSON POWER SYSTEMS DIV.				
DIESEL GENERATOR PROTECTIVE RELAY PANELS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
DIESEL GENERATOR CONTROL PANELS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
DC DISTRIBUTION PANELS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
125V DIESEL BATTERIES AND BATTERY RACKS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
DUAL BATTERY CHARGER ASSEMBLIES	A-742	129810		IEEE 344-1971	TEST	TRIAXIAL MULTIFREQUENCY	QUALTECH NP
STANDBY DIESEL GENERATORS				IEEE 344-1971	ANALYSIS		
EMERGENCY DC LIGHTING	A-757 1-772	75C2-85737-1	GRAYBAR ELECTRIC, INC.	IEEE 344-1971	TEST	MULTIFREQUENCY, BIAXIAL	WYLE LAB
NUCLEAR INSTRUMENTATION	CB 755	92NLF-75345A	GAMMA METRICS	IEEE 344-1974	TEST	BIAXIAL, MULTIFREQUENCY	WYLE LAB
FOXBORO SPEC 200 CONTROLLERS	CB 755	75514A	FOXBORO	IEEE 344-1974	TEST	BIAXIAL, MULTIFREQUENCY	FOXBORO

NOTE 1 - These instruments were purchased under various TVA contracts and were tested to IEEE 344-1971 and 344-1974 criteria.

WBNP INSTRUMENTATION AND ELECTRICAL EQUIPMENT
SEISMIC QUALIFICATION SUMMARY

EQUIPMENT	LOCATION BLDG/EL*	TVA CONTRACT NO.	VENDOR	SEISMIC QUALIFICATION CRITERIA	QUALIFICATION METHOD	TEST METHOD	TEST LAB
CONTROL INSTRUMENT LOOPS	MULTIPLE LOCATION S	73C3-92784	BAILEY METER CO	IEEE 344-1971	TEST	SINGLE AXIS	ACTION ENVIRON-MENTAL TESTING CORP.
INSTRUMENTATION AND CONTROLS	MULTIPLE LOCATION S	77K3-87352	ROBERTSHAW CONTROLS CORP	IEEE 344-1974	TEST	RANDOM FREQUENCY, MULTIAXIS	WYLE LAB
CONTROL INSTRUMENT LOOPS (Unit 2)	MULTIPLE LOCATION S	69016/71252	FOXBORO SPEC 200	IEEE 344-1975	TEST	MULTIAXIS	NUCLEAR QUALIFICATION SERVICES
PANELS 2-L-11-A AND 2-L-11-B				IEEE 344-1975	ANALYSIS		
FABRICATION OF LOCAL PANELS AND INSTALLATION OF INSTRUMENTS	MULTIPLE LOCATION S	73C38-92800	H. K. PORTER	IEEE 344-1974 (DRAFT REVISION TO IEEE 344-1971)	TEST	MULTIFREQUENCY, BIAXIAL	WYLE LAB
PROCESS TRANSMITTERS, PRESSURE GAUGES, PRESSURE SWITCHES, & LEVEL SWITCHES, REPLACEMENT PARTS FOR OBSOLETE EQUIPMENT	SEE NOTE 1						
STANDBY POWER SYSTEM	D-742	74C63-83090	MORRISON-KNUDSON POWER SYSTEMS DIV.				
DIESEL GENERATOR PROTECTIVE RELAY PANELS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
DIESEL GENERATOR CONTROL PANELS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
DC DISTRIBUTION PANELS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB

TABLE 3.10-1 (Sheet 4 of 5)

WBNP INSTRUMENTATION AND ELECTRICAL EQUIPMENT
SEISMIC QUALIFICATION SUMMARY

EQUIPMENT	LOCATION BLDG/EL*	TVA CONTRACT NO.	VENDOR	SEISMIC QUALIFICATION CRITERIA	QUALIFICATION METHOD	TEST METHOD	TEST LAB
125V DIESEL BATTERIES AND BATTERY RACKS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
BATTERY CHARGERS				IEEE 344-1971	TEST	BIAXIAL MULTIFREQUENCY	WYLE LAB
STANDBY DIESEL GENERATORS				IEEE 344-1971	ANALYSIS		
EMERGENCY DC LIGHTING	A-757 1-772	75C2-85737-1	GRAYBAR ELECTRIC, INC.	IEEE 344-1971	TEST	MULTIFREQUENCY, BIAXIAL	WYLE LAB
ELECTRICAL PENETRATIONS	R-(ALL LEVELS)	76K6I-87064	CONAX CORP	IEEE 344-1971	ANALYSIS		
125V DC VITAL BATTERIES	A-757	76K3-85763	GOULD IND.	IEEE 344-1975; IEE P535, DRAFT 2 DATED NOV. 15, 1974	TEST	RANDOM, BIAXIAL, MULTIFREQUENCY	WYLE LAB
TRANSFER SWITCHES	A-772 A-757	75K5-87048	B-K ELECT PROD	IEEE 344-1971	TEST	SINGLE AXIS, SINE BEAT	AERO NAU LAB
SPARE TRANSFER SWITCHES	A-772	00072332	AMETEK SCI	IEEE 344-1975	TEST	MULTIFREQUENCY, RANDOM BIAXIAL	WYLE LAB
125V DC VITAL CHARGERS	A-772	74C8-85251	PWR. CONV. PROD.	IEEE 344-1971, DRAFT 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
125V DC BATTERY BOARDS	A-757	75C2-85281	WESTINGHOUSE	IEEE 344-1971, ENCLOSURE NO. 5	TEST	SINE BEAT, BIAXIAL, FOUR POSITIONS	WESTINGHOUSE
120V AC VITAL INVERTERS	A-772	69414	AMETEK SCI	IEEE 344-1971, DRAFT 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
120 AC VITAL INSTR. BOARDS	A-757	74C4-85216	WESTINGHOUSE	IEEE 344-1971, ENCLOSURE NO. 5	TEST	SINE BEAT, BIAXIAL, FOUR POSITIONS	WESTINGHOUSE

WBNP INSTRUMENTATION AND ELECTRICAL EQUIPMENT
SEISMIC QUALIFICATION SUMMARY

EQUIPMENT	LOCATION BLDG/EL*	TVA CONTRACT NO.	VENDOR	SEISMIC QUALIFICATION CRITERIA	QUALIFICATION METHOD	TEST METHOD	TEST LAB
6.9KV SD BD LOGIC PANELS	A-757	75K2-85354	H. K. PORTER	IEEE 344, R3 (FEB. 15, 1974)	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
6.9KV SD BDS	A-757	74C2-84376	G. E.	IEEE 344, DRAFT REV. 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
480V SD BDS, TRANSFORMERS AND PRESS. HTR. TRANSFORMERS	A-772 A-782	74C2-84647	WESTINGHOUSE	IEEE 344, R3, AND ENCLOSURE NO. 5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
480 DISTRIBUTION PANELBOARDS FOR PRESSURIZER HEATER BACKUP GROUP	A-782	75K3-86476	EL TEX	IEEE 344, DRAFT R5	TEST	MULTIFREQUENCY, RANDOM, BIAXIAL	WYLE LAB
480V MOTOR CONTROL CENTERS	A-772 A-757	74C5-84646	ITE	IEEE 344, DRAFT 3 (FEB. 15, 1974)	TEST	SINE BEAT	WYLE LAB

*R-REACTOR BLDG.
A-AUXILIARY BLDG
C-CONTROL BLDG.
D-DIESEL GEN. BLDG.

NOTE 1 - These instruments were purchased under various TVA contracts and were tested or qualified to IEEE 344-1971 or 344-1975 criteria.

TABLE 3.10-2 (SHEET 1 of 4)

QUALIFICATION OF INSTRUMENTATION AND CONTROL EQUIPMENT

Equipment	Qualification Method*	Standard to Which Qualified*	Organization Performance Testing/Analysis and Date of Completion
Reactor Trip and Bypass Breakers	1 & 3 testing		Westinghouse
Solid State Protection System	1 & 2 testing		Westinghouse
Eagle-21 Process Protection System	3 testing		Westinghouse
Nuclear Instrument System			
Unit 1 Cabinets	1, 2 & 6 testing and analysis		TVA & Westinghouse
Unit 2 Cabinets	1, 2 & 8 testing and analysis		Westinghouse
Power Range Electronics	1 & 2 testing		Westinghouse
Source & Intermediate Range Electronics	7 analysis (Unit 2)		Thermo Fisher Scientific
Neutron Detectors			
Power Range	1 testing		Westinghouse
Source/Intermediate Range	7 analysis (Unit 2)		Themo Fisher Scientific
Process Transmitters			
Unit 1	1 & 2 testing		Westinghouse
Unit 2	1, 2 & 11-17 testing		Manufacturer & Westinghouse
Containment Pressure Transmitters	3 testing		

TABLE 3.10-2 (SHEET 2 of 4)

QUALIFICATION OF INSTRUMENTATION AND CONTROL EQUIPMENT

Equipment	Qualification Method*	Standard to Which Qualified*	Organization Performance Testing/Analysis and Date of Completion
Solid State Protection system Output Relays	1 & 2 testing		Westinghouse
Engineered Safeguards Test Cabinets	1 testing		Westinghouse
Control Room Panels	1 & 4 testing and analysis		Westinghouse
Safety System Status Monitoring System			
Post Accident Monitoring System	1 & 2 testing		Westinghouse
Unit 1 ICCM 86	1 & 2 testing		Westinghouse
Unit 2 Common Q	10 testing		Westinghouse

TABLE 3.10-2 (SHEET 3 of 4)

QUALIFICATION OF INSTRUMENTATION AND CONTROL EQUIPMENT (Cont'd)

*Qualification Method	Description of Method
1	Sine beat; single axis (Ref. WCAP-7558, WCAP-7817[[3], [4], [5], [6], [7] and its supplements, and WCAP-8373, as per IEEE-344-1971).
2	Demonstration Test Program biaxial test inputs with multifrequency forcing functions, as per requirements of IEEE-344-1975.
3	IEEE-344-1975
4	Analysis
5	Test documented in WCAP-8687, Supplement 2-E15A (Proprietary) and WCAP-8587, Supplement 2-E15A (Non-Proprietary)
6	See System Description, N3-92-4003, “Neutron Monitoring System” for a listing of the reference calculations.
7	Thermo Fisher Scientific Qualification Report No. 864, Rev. 0, Class 1E Qualification of Source Range, Intermediate Range and Wide Range Channels.
8	Westinghouse Report EQ-EV-39-WBT Revision 1, Seismic Evaluation of Nuclear Instrumentation System Console 2-M-13 with Gammametrics Equipment for Watts Bar Unit 2, Revision 1, March 2009.
9	WCAP-8587, “Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment, “Revision 6-A (NP), Dated March 1983
10	Ametek Report No. TR-1136, Qualification Documentation Review Package for Ametek Aerospace Gulton-Statham Products Neclear Qualified Pressure Transmitter Series Enveloping --- Gage Pressure Transmitter Series PG 3200, Differential Pressure Transmitter Series PO 3200 Differential High Pressure Transmitter Series PDH 3200, Draft Range Pressure Transmitter Series DR 3200, Remote Diaphragm Seal Differential Pressure Transmitter Series PO 3218, Remote Diaphragm Seal Differential High Pressure Transmitter Series PDH 3218.
11	Rosemount Report D2001019 Rev. B, Model 3051N Qualification Report

TABLE 3.10-2 (SHEET 4 of 4)

QUALIFICATION OF INSTRUMENTATION AND CONTROL EQUIPMENT (Cont'd)

*Qualification Method	Description of Method
12	Rosemount Report 117415 Rev, H. Qualification Tests for Rosemount Pressure Transmitter Model 1152
13	Rosemount Report D8300040 Rev. E, Qualification Report for Pressure Transmitters Rosemount Model 1153 Series D
14	Rosemount Report D8400102 Rev. F, Qualification Report for Pressure Transmitter Model 1154
15	Rosemount Report D8700096 Rev. I, Qualification Report for Rosemount Model 1154 Series H Pressure Transmitter
16	Weed report 16690-QTR, Revision 0, Qualification Test Report for Environmental and Seismic Qualification of Week Model DTN2010 Pressure Transmitters.

WATTS BAR SEISMIC QUALIFICATIONS

Equipment: Metalclad Switchgear

Equipment Rating: 6.9 kV, 60 Hz, 3-phase

Mounting: The switchgear was bolted to test table to simulate in-service configuration.

Seismic Test: The control circuits of the switchgear were energized with 125 VDC and subjected to the following tests:

1. Exploratory tests (Resonant Search)--Consisting of a low level single axis sweep from 1 Hz to 35 Hz at a rate of two octaves per minute and at a level of 0.2 g per peak. Resonant search test was performed in the front-to-front and side-to-side orientation.

2. Proof Test-Consisting of biaxial multifrequency random tests in front-to-back and side-to-side orientations. More than 5 OBE'S and one SSE were performed in each orientation.

Monitoring: A multichannel recorder was used to monitor electrical continuity contact chatter and change of state before, during, and after tests.

Results: The specimen's structural integrity was not compromised and circuit continuity was maintained.

Reference: Wyle Laboratories Report No. 42868-1.

Equipment: 6900V Shutdown Board Logic Panels.

Electrical Rating: Not applicable.

Mounting: The specimen was mounted with its base flush to the test table and welded to the table top, simulating the in-service configuration.

Seismic Test: The specimen control circuits were energized (125V DC) and the specimen was subjected to the following tests:

1. Exploratory Test (Resonant Search) - Consists of low-level (0.2g horizontal and 0.1g vertical) multiaxis sine sweep from 1 Hz to 35 Hz to 1 Hz in front-to-back and side-to-side orientations to determine major equipment resonance points.

2. Proof Test (Multifrequency)-- consisting of simultaneous horizontal and vertical incoherent inputs of random motion at frequencies spaced 1/3 octave apart from 1-4 Hz in front-to-back and side-to-side orientations. Aging was obtained with a minimum of five half-level SSE tests in each orientation prior to performing the full-level SSE test.

Monitoring: A multichannel recorder was used to monitor electrical continuity, contact chatter, and change of state before, during, and after the seismic test.

Results: The specimen's structural integrity was not compromised. A"b" contact of the normally de-energized DC auxiliary relay controlled by the AC undervoltage relay experienced contact chatter.
Based upon further analysis of the "b" contact, it was determined that the contact chatter was of such an extremely short-term duration that it will not in any way affect system operation.

Reference: Wyle Test Report No. 43137-1.

Equipment: Power Transformers

Electrical Ratings: 6900 to 480-V AC, 60-Hz, 3 phase in ratings of 500*, 1000* and 2000 KVA (* qualified by analysis).

Mounting: The 2000 KVA transformer was bolted to the test table to simulate actual in-service mounting.

Seismic Tests: The transformer was energized to 480V AC on the secondary and subjected to the following tests:

1. Resonance Search--A low level (0.2g) biaxial, sinusoidal sweep from 0.5 Hz to 40 Hz to 0.5 Hz was performed to determine resonance.

2. Proof Testing--The transformer was subjected to a 30 second biaxial random motion input wave with sine beats added to ensure enveloping of the required response spectra by the test response spectra. Five OBE's followed by one SSE were performed in each of four orientations (45°, 135°, 225°, and 315° to the axis of the test table). After proof tests another resonance search was performed in the last orientation.

3. Analysis--The 500 and 1000 KVA transformer were compared to the 2000 KVA transformer actually tested. It was shown that the two smaller transformers have lower weights, smaller dimensions and less coolant than, but are of the same skin thickness as the 2000 KVA transformer and therefore, have lower stresses for the same seismic excitation.

Monitoring: Data was recorded on three ink-type oscillographs.
Input acceleration was analyzed on a spectral dynamics shock spectrum analyzer

Results:	No structural damage occurred and the 2000 KVA trans-former remained fully operational during and after testing. The 500 and 1000 KVA transformers were analytically found to be capable of successfully withstanding a seismic excitation equal to that of the 2000 KVA transformer.
Reference:	Seismic Test Report No. XAL 71789, SD 3036, Westinghouse Seismic Design Analysis Report No. SBR-75-7.
Equipment:	480V metal enclosed switchgear, Westinghouse type DS.
Electrical Rating:	480V AC, 60 Hz, 3 phase.
Mounting:	The test specimen, consisting of two typical units, was bolted to the test table to simulate actual mounting.
Seismic Tests:	The test specimen was energized to 480V AC and 125V AC (control circuits) and subjected to the following tests: 1. Resonance Search--A low level (0.2g) biaxial sinusoidal sweep from 1.0 Hz to 40 Hz to 1.0 Hz was performed on the specimen in the 225° (to the test table axis) orientation. 2. Proof Test--The specimen was subjected to an input wave made up of decaying sinusoids from 1.25 Hz to 35 Hz with two sine beats added to achieve enveloping of the required response spectra by the test response spectra. A minimum of five OBE's followed by a minimum of four SSE's were performed in the 225° and 315° orientations. In the 45° and 135° orientations only SSE's were performed. A second sine sweep for resonance was performed after proof testing.
Monitoring:	Contact monitoring was performed by an event recorder and seismic monitoring by four ink-type oscillographs. Additionally, 20 accelerometers, 5 strain bolts and a strain gage were also used for monitoring.
Results:	The specimen maintained its structural integrity and there were no failures that would jeopardize proper functioning of the equipment.
Reference:	Westinghouse Seismic Qualification Report, type DS low voltage metal enclosed switchgear, with attached Westinghouse Astronuclear Laboratory Report (XAL 71706, SD 3027). (WCAP-10448)
Equipment:	480V Motor Control Centers--I-T-E Imperial Corporation, subsidiary of Gould, Inc., Type 5600.

Electrical Rating: 480V AC, 60 Hz, 3-phase, 22KA short circuit bracing.

Mounting: The specimen which consisted of 3 panels was welded to the test table simulating the in-service condition.

Seismic Test: Control circuits (120V AC) of the specimen were energized and the specimen was subjected to the following tests:

1. Exploratory Tests (Resonant Search)--Consisting of low level (approximately 0.2 g horizontal and 0.1 g vertical) biaxial sine sweep from 1 Hz to 35 Hz in front-to-back/vertical axes test and side-to-side/vertical axes test to determine major equipment resonances. Resonant frequencies at 8,75 Hz and 35 Hz were found for the front-to-back/vertical axes test and resonant frequencies of 10 Hz and 35 Hz were found for the side-to-side/vertical axes test.
2. Proof Test (Biaxial Sine Beat)--Consisting of biaxial sine beats at equipment resonances listed above. Aging was accomplished during the 8.75 and 10 Hz tests with 4 one-half level phase coherent half level tests and 4 half level incoherent tests. Full level (1.37 g input) tests were performed at each resonance for energized, de-energized, in phase, and out-of-phase conditions.

3. Main circuits (480V AC) were de-energized during tests.

Monitoring: A 14-channel recorder was used to monitor electrical continuity, current/voltage levels, spurious operation, and contact chatter before, during, and after the seismic tests.

Results: It was demonstrated that the specimen possessed sufficient integrity to withstand, without compromise of structure, the prescribed seismic environment. Some contact chatter was encountered during the testing; however, full-level tests were performed without chatter after corrective action was taken by the I-T-E Technical Representative.

Reference: I-T-E Imperial (Subsidiary of Gould, Inc.), Seismic Certification SC077 and Wyle Test Report TR-42926-1.

Equipment: 480V Power Distribution Cabinet

Electrical Rating: 480V AC, 60 Hz, 3-phase

Mounting: Specimen was wall-mounted with commercially available bolts, nuts, and washers to a wall-mounting fixture which was welded to the test table. Mounting simulated the in-service configuration.

Seismic Test: The specimen was subjected to the following tests:

1. Exploratory Test (Resonant Search)--Consisting of a low level (0.2 g horizontal and vertical) biaxial sine sweep from 1-33 Hz in both front-to back and side-to-side orientation to obtain major resonance points of the equipment.

2. Proof Test (Multifrequency)--Consists of simultaneous horizontal and vertical incoherent inputs of random motion at frequencies spaced 1/3 octave apart from 1-31.6 Hz in front-to-back and side-to-side orientations. Aging was accomplished with a minimum of 5 half-level SSE tests followed by one full level SSE test performed in both orientations.

Monitoring: A multichannel recorder was used to monitor electrical continuity, contact chatter, and change of state before, during, and after the seismic test.

Results: The specimen's structural integrity and circuit continuity was not compromised.

Reference: Wyle Test Report No. 43039-1.

Equipment: The components of the Emergency Diesel Generator.

Seismic Test: An analysis was performed on the components listed on page 12 of this table for the seismic conditions and criteria as specified in TVA Specification WB-DC-40-31.2.

1. The natural frequencies of the engine and the generator system were determined by analysis.

2. Selected critical components 1-9 were analyzed for 3 g horizontal and 2 g vertical accelerations (above seismic criteria), whereas, components 10-21 were analyzed for 1.62 g horizontal accelerations and 1.08 vertical accelerations (as specified by seismic criteria).

Results: The system was found to be rigid with natural frequencies above the seismic range. All the components analyzed had conservative margins of safety in all cases under the maximum combined loadings.

References: Corporate Consulting and Development Co., Ltd. CCL Report No. A-27-74, CCL Project No. 74-1110, DDL Report No. A-5-73A, CCL Project No. 73-1024.

TABLE 3.10-3 (SHEET 6 of 17)

WATTS BAR SEISMIC QUALIFICATIONS

Diesel Generator Components

1. Air intake filter at Mounting Bracket.
2. Accessory Rack at Base.
3. Engine Tube Oil filter at Accessory Rack.
4. Water Expansion Tank at Accessory Tack.
5. Governor (Hydraulic Actuating).
6. Primary Oil Pump at Engine.
7. Scavenging Oil Filter at Engine.
8. Scavenging Oil Pump at Engine.
9. Engine Lube Oil Cooler.
10. Engine at Base.
11. Generator at Base.
12. Engine at Foundation.
13. Generator at Foundation.
14. Heat Exchanger.
15. Air Turning Box.
16. Air Intake Filter at Foundation.
17. Air Tank Saddle at Bolts.
18. Air Tank at U-Bolts.
19. Critical Weld in Air Tank Saddle Mount.
20. Maximum Pull Out of Concrete Insert.
21. Maximum Shear in Concrete Insert.

TABLE 3.10-3 (SHEET 7 of 17)

WATTS BAR SEISMIC QUALIFICATIONS

Equipment: Diesel Generator Control Panel, D. C. Distribution Panel, and Battery Charger.

Mounting: As per test specifications, the equipment was mounted on the Wyle Multi-Axis Simulator.

Seismic Test:

1. A resonant search test was conducted consisting of two low level (approximately 0.2g horizontally and 0.1g vertically) multi-axis sine sweeps in each test one octave per minute.

2. Two sine beat tests in-phase and two sine beat tests out-of-phase were performed at each resonant frequency. The sine beat consisted of 15 oscillations per beat. A train of five beats with a two-second interval between beats was used at each test frequency.

Monitoring: Three to four electrical monitoring channels were recorded on an oscillograph recorder to ascertain electrical continuity, current/voltage levels, spurious operations, contact chatter, before, during, and after the seismic excitation.

Results: The equipment listed above withstood the prescribed simulated seismic environment without any loss of electrical functions and structural failures.

Reference: Wyle Report No. 42879-1.
Wyle Job No. 42879.

Equipment: The main components of the Emergency Diesel Generator analyzed are as follows:

1. Stator Frame
2. Rotor Shaft
3. Pole Doetail
4. Pole Heads
5. Bearings
6. End Bells
7. Mounting Feet
8. Hold Down Bolts

Seismic Test: An analysis was performed on the components listed above for seismic criteria as specified by TVA Specification WB-DC-40-31-2.

1. The components listed above were analyzed for a seismic loading of 2.7 g horizontal acceleration and 1.8 g vertical acceleration acting concurrently.

2. The mechanical response of the rotor and stator in terms of deflections and stresses was evaluated from static considerations and vertical seismic loads, acting concurrently.

Results: All the stresses calculated were within the respective material working levels as called out in the material specs.

References: Elective Products Division, Portec, Inc., Cleveland, Ohio. Analysis (B07 891005 019)

Equipment: Emergency Diesel Generator Atlas Jacket Water Cooler.

Seismic Test: An analysis was performed on the Atlas Water Jacket Cooler for seismic conditions as specified in TVA Specification WB-DC-40-31-2.

1. The analysis evaluated the base natural frequency of the structure to determine the appropriate acceleration from the response spectra.

2. For emergency, upset, and normal conditions, the analysis considered the adequacy of the anchor bolts, supports, shell at the supports, and tubes, due to a combination of seismic loads, nozzle loads, deadweight, and pressure.

3. When emergency condition stresses were less than normal condition allowables, separate upset and normal load case calculations were omitted.

Results: All the calculated stresses at the various points on the cooler were well below the allowable stresses under all possible conditions.

Reference: Dynatech Project No. AIM-20.
Dynatech Report No. 1237.

Equipment: Diesel Generator Relay Boards.

Electrical Rating: 125V DC, and 120V AC, 60 Hz, 1 phase.

Mounting: The test specimen was welded to a steel plate which in turn was welded to the seismic test machine.

Seismic Testing: 1. Resonance Search--A resonance search was conducted using an input level of 0.2 g in the frequency range of 1 to 33 Hz with a frequency sweep rate of 1 octave per minute. The resonance search was conducted with the horizontal and vertical axis simultaneously, first in phase and then with the horizontal 180° from the original phase. The resonance search was then repeated in the second horizontal and

vertical axis. The two horizontal axis were identified as longitudinal and lateral. Two control and six response accelerometers were utilized during the resonance search. The output of each accelerometer was recorded on a direct readout recorder.

2. Sine Beat Test--A sine beat test composed of ten oscillations per beat was performed at each resonance frequency determined from the resonance search. Each sine beat was applied five times at each frequency, first with one horizontal and the vertical inputs in phase and then with the horizontal input 180° from the original phase. A two-second interval between each sine beat was used. The maximum peak acceleration for the horizontal axes was 1.08 g and for the vertical axes was 0.72 g. In addition to the resonance frequencies, the sine beat test was performed at each one-half octave over the frequency range of 1 to 33 Hz.

Monitoring: A 125V dc voltage source was used to energize relay circuits during testing. A source of 120V ac was applied to monitor an ac voltmeter on the front panel of the specimen. Two chatter/transfer detectors were utilized to monitor a total of 14 channels of relay contacts during testing. The chatter/transfer detectors were set for a time duration of 1.0 millisecond or greater. During testing all monitored circuits were tested in the transfer, or open, mode. In the event of a momentary closure of a duration of 1.0 millisecond or greater, a red indicator light for that particular circuit would illuminate and remain illuminated until a reset button was pushed to reset the circuit.

Results: Visual examination of the test specimen after each test revealed no structural damage due to the seismic tests. Of the 14 channels monitored during testing, only one circuit (Westinghouse type CM relay) indicated chatter. TVA circuits show that this relay will only be used during testing phases of the diesel generator system; therefore, it cannot prevent operation of the diesel generators during or after a seismic event because of contact chatter.

Reference: Wyle Laboratories Report No. 54064.

Equipment: Emergency Generator Starting and Control System including:

Test Series 1

1 Battery (six cells), C & D 3DCU-9 with Rack PSD-007012
1 Battery Charger, C & D ARR130HK-50
1 Fuel Oil Pump, Viking GG195D
1 Soakback Oil Pump, EMD 8274507
1 Contactor/Relay System consisting of:
1 Square D Temperature Switch, Class 9025, Type BGW397

1 Fenwall Temperature Switch, No. 20800
1 Barksdale Pressure Switch, E1HM90V
1 Overspeed Trip Limit Switch, EMD 8246095
1 Crankcase Pressure Switch, EMD 8370362
2 Square D Relays, Class 8501, Type KP

Test Series 2

1 Engine Control Panel, PSD-A990F02501
1 Anode Transformer, GE-278G121AA

Test Series 3

1 Switchgear Exciter Assembly, PSD-A990F11000

Mounting: As per test specifications, the equipment was mounted on the Wyle Multi-Axis Simulator.

Seismic Test:

1. A resonant search consisting of a low level single axis sine sweep from 1 Hz to 33 Hz was performed to establish natural frequencies. The input acceleration level for all pieces of equipment was 0.2 g in the vertical direction and 0.4 g in the horizontal direction (when specified as needed).

2. Sine beat tests were performed at the natural frequencies detected in the resonant search test. The sine beat tests were performed as a train of beats with the number of beats per train depending upon the resonant frequency of each piece of equipment. The vertical and horizontal accelerations in the sine beat test were varied according to the piece of equipment being tested.

Monitoring: All specimens shall be operating during full level testing, and specified functions will be monitored and recorded. Electrical powering of 480V AC, 3 phase will be furnished for operation of the switchgear, battery charger, soakback pump and the anode transformer. For operation of the Fuel Oil Pump, 125 VDC will be furnished.

The battery discharge rate (10 amps), the charger output rate, the anode transformer secondary voltage and the discharge pressure of both pumps will be recorded on an oscillograph recorder. continuity circuits on the Switchgear Assembly, the Engine Control Panel and the contactor/Relay System will be monitored for a change of state and a monitor trace recorded on an oscillograph recorder.

Results: The equipment demonstrated the ability to withstand the prescribed seismic environment without any loss of electrical function or significant operational change. No structural degradation was noted during the tests

Reference: Wyle Report No. 42749-1.
Wyle Job No. 42749.

Equipment: Transfer Switch Circuit Breaker Assemblies and Safety Switch.

Electrical Ratings:

1. 480V AC, 400A, manual circuit breaker assembly, NEMA 1 enclosure.
2. 480V AC, 600A, manual circuit breaker assembly, NEMA 1 enclosure.
3. 600V AC, 60A, 3 pole, non-fusible safety switch NEMA 1 enclosure.

Mounting: The switches were mounted to the test fixture to simulate actual mounting. The 400A circuit breaker assembly and the safety switch were tested on the same fixture. The 600A circuit breaker assembly was tested separately.

Seismic Tests: The switches were energized to 480V and subjected to the following tests:

1. Resonance Search--The switches were subjected to a continuous sinusoidal search from 1 to 35 Hz in each of three mutually perpendicular axes. The frequency was adjusted in discrete 1 Hz steps with vibration maintained for at least 20 seconds in each step. Peak acceleration varied from 0.1 g to 0.31 g above 5 Hz and 0.01 g to 0.26 g from 1 Hz to 5 Hz.

2. Proof Testing--The switches were subjected to a 30 second 0.5 SSE of 0.76 g followed by a 30 second SSE of 1.51 g's at each half octave over the range from 1 Hz to 35 Hz where no resonant frequencies were present. Where resonance was present, a 0.5 SSE (0.76 g) followed by a SSE (1.51 g's) were performed at those resonant frequencies. Testing was conducted in the front-to-back, side-to-side and vertical axes.

Monitoring: The switches were monitored for proper operation during the testing.

Results: No external physical damage or malfunction was noted as a result of these tests.

Reference: Aero Nov Laboratories, Inc., Test Report No.
5-6156, dated October 31, 1975.

TABLE 3.10-3 (SHEET 12 of 17)

WATTS BAR SEISMIC QUALIFICATIONS

Equipment:	125V-dc Battery Charger.
Electrical Rating:	ac input-480V, 60Hz, 3. dc output--125V, 100 Amperes.
Mounting:	The battery charger channel sill was welded to the Wyle simulator table in a manner duplicating the expected in-service configuration at WBNP. The control accelerometers were recorded on tape oscillograph recorders. The resulting table motion was analyzed by a spectrum analyzer at a damping of two percent (2%) and plotted at one-third octave frequency intervals over the frequency range of interest. Five, one-half-level RRS tests followed by a full-level RRS test were performed in each orientation with the specimen energized and operating in its normal charging mode. Full-level RRS tests were also performed in the front-to-back/vertical orientation during which the specimen's ac and dc circuit breakers were tripped using a low voltage ac current source.
Monitoring:	A multichannel recorder and three accelerometer devices were used to monitor the test results. Each accelerometer device consisted of two sensors, one oriented for the vertical axis and the other for the horizontal axis, the horizontal axis accelerometers were realigned for the direction of motion after the charger was rotated 90 degrees on the horizontal plan. Three channels of the multichannel recorder were used to monitor (1) input voltage, (2) state of a parallel circuit consisting of the NO alarm contacts, and (3) output voltage.
Results:	The oscillograph traces revealed no alarm contact chatter or breaker misoperation. No apparent physical damage was noticed in the visual checks. The charger performed satisfactorily before, during, and after the tests. The two DC meters mounted on the front of the charger cabinet would "peg" during the full SSE but each time they would return pretest readings with no recalibration necessary.
Reference:	Wyle Laborator's Seismic Test Report No. 42959-1.
Equipment:	125-Volt Battery (Cell type NCX-2100).
Electrical Rating:	Two-hour rate--696 amperes to minimum battery terminal voltage of 105 volts at 60 F initial electrolyte temperature.
Mounting:	Test rack containing the batteries was welded to the test table.
Seismic Test:	Battery (3 cells) was energized to an approximate 20-ampere restive load and subjected to the following tests;

1. Exploratory test (resonant search) consisting of a low level (approximately 0.2 g horizontally and vertically) sine sweep was performed to determine the specimen resonance frequencies in each of the three orthogonal axes. The sweep rate was 1 octave per minute over the frequency range of 1 Hz to 35 Hz.

2. Proof test (multifrequency) consisting of 30-second duration simultaneous horizontal and vertical inputs of random motion consisting of frequency bandwidths spaced one-third octave apart over the frequency range of 1 Hz to 35 Hz. The amplitude of each one-third octave bandwidth was independently adjusted in each axes until the TRS enveloped the RRs.

Monitoring: The output voltage of the battery was monitored on an oscillograph recorder during the seismic excitation.

The following parameter was monitored:
Output voltage +4 Percent.

Results: The battery performed satisfactorily and all parameters monitored were within their prescribed tolerances before, during, and after the test.

References: Wyle Laboratories Test Report No. 43479-1.

Equipment: Battery Rack.

Seismic Test: The battery rack was qualified by analysis as described below:

1. Natural Frequency--The calculations for natural frequency were based upon a static analysis where all the component parts were analyzed for deflection. The sum of the deflections was then used to calculate the natural frequency of the rack.

2. After determination of the natural frequency of the rack, a stress analysis was again performed on each individual component part. The absolute combined stress was then calculated by peak value analysis.

3. Data was provided with the analysis to show that all the critical acceleration response spectra was enveloped in this test.

Results: The data showed that the rack will meet the requirements as laid out by TVA specifications and IEEE 344-1971 and will perform adequately during and after a seismic event.

TABLE 3.10-3 (SHEET 14 of 17)

WATTS BAR SEISMIC QUALIFICATIONS

Reference: Gould, Inc., Industrial Battery Division, 60 NCX-2550 and SO 7-074526-806. WYLE Report NO. 43479-1

Equipment: 125V DC Vital Battery Boards.

Electrical Rating: 125V Dc, 20,000 amperes short-circuit.

Mounting: The equipment was bolted to the vibration generator in a manner that simulated the intended service mounting including bolt size and configuration.

Seismic Tests;

1. A resonant search test in test direction No. 1 using a sinusoidal input level of approximately 0.2 g from 1 to 33 Hz, and at a sweep rate of one octave per minute.

2. A sine beat test in test direction No. 1 with maximum peak acceleration corresponding to the SSE. The beat test was conducted at each natural frequency. Before each SSE beat test, five 1/2 SSE beat tests were applied.

3. Steps 1 and 2 repeated for 3 more directions.\

Monitoring:

1. Six accelerometers were mounted on the test table and throughout the boards to monitor input and output accelerations. The output was recorded on graphs made from oscillographs.

2. Ten circuit breakers carrying 90% of rated current were monitored for contact opening.

3. Alarm reset light was energized during all tests and was visually monitored.

4. Undervoltage relay was energized and the normally closed contact was monitored during all tests.

5. Each fuse checked for continuity after testing.

Results: The testing proved the integrity of the board/component system since no failures developed.

Reference: Westinghouse Seismic Test Procedure No. CO-33697.
PEI-TR-852001-12

Equipment: 120V AC Vital Instrument Power Boards.

Electrical Rating: 120V AC, 60 Hz and 5,000 amperes short-circuit.

Mounting: The equipment was bolted to the vibration generator in a manner that simulated the intended service mounting including bolt size and configuration.

Seismic Tests:

1. A resonant search test in test direction No. 1 using a sinusoidal input level of approximately 0.2 g from 1 to 33 Hz, and at a sweep of 1 octave per minute.
2. A sine beat test in test direction No. 1 with maximum peak acceleration corresponding to the SSE. The beat test was conducted at each natural frequency. Before each SSE beat test, five 1/2 SSE beat tests were applied.
3. Steps 1 and 2 repeated for 3 more directions.

Monitoring:

1. Six accelerometers were mounted on the test table and throughout the boards to monitor input and output accelerations. The output was recorded on graphs made from oscillographs.
2. Ten circuit breakers carrying 90 percent of rated current were monitored for contact opening.
3. Lights were energized during all tests and were visually monitored.
4. Undervoltage relay was energized and the normally closed contact was monitored during all tests.
5. Fuses checked for continuity after testing.

Results: The testing proved the integrity of the board/component since no failures developed.

Reference: Westinghouse Seismic Test Procedure No. CO-33419.

Equipment: 120V AC, 60 Hz Vital Instrument Static Inverter.

Electrical Rating: AC input--480V, 60 Hz, 3 phase, DC input--125V.
AC out--120V, 60 Hz, Single phase.
KVA out--20kVa.

Mounting: The inverter channel sills were welded to the shake table in the exact manner they would be installed on steel floor plates at Watts Bar.

Seismic Test: The inverter was energized to 480V, 3 phase and subjected to the following tests:

1. Exploratory Search--A low level (approximately 0.2 g horizontally and vertically) performed on each test configuration from 1 Hz to 33 Hz to establish major resonances. The sweep rate was one octave per minute.

2. Multifrequency Tests--The specimen was subjected to simultaneous horizontal and vertical inputs of random motion consisting of frequencies spaced 1/3 octave apart over the range of 1 Hz to 40 Hz. The amplitude of each 1/3 octave frequency was independently adjusted in each axis until the test response spectra enveloped the required response spectra. The resulting test table motion was analyzed at one percent damping by a spectrum analyzer and plotted at one-third octave intervals over the frequency range of interest. The duration of the tests was 30 seconds. The horizontal and vertical input accelerations levels were phase incoherent. Five 1/2-level SSE's and one SSE were performed on the invertor

Monitoring: The equipment used to monitor the test included a visual counter for output frequency; a 3-channel recorder to monitor (1) input voltage, (2) state of a parallel circuit of 12 NO alarm relay contacts, and (3) output voltage: and 5 accelerometers. Each accelerometer device consisted of two sensors, one oriented for vertical axes, and the other for horizontal axes.

Results: The inverter withstood the seismic test satisfactorily without any failures.

Reference: Wyle Laboratories Seismic Test Report No. 51133-1.

Equipment: Electrical penetrations[1] (all types and voltages used at Watts Bar).

Seismic Test: Seismic qualification was done by analysis. The seismic analysis done on the penetrations consider the seismic loads imposed for both a safe shutdown earthquake and a 1/2 safe shutdown earthquake in accordance with paragraph NA-3250 of the ASME Boiler and Pressure Vessel Code, Section 111, Nuclear Power Plant Components.[2]

1. The analysis calculated the natural frequencies during a seismic event using standard formulas for stress and strain by the R. Roark or Rayleigh's methods.

2. Maximum stresses for the normal and seismic load conditions were calculated. Seismic loads were considered to act in the vertical direction and in two horizontal directions.

Results: The analysis indicated that the penetrations were able to withstand all seismic stresses from a one and a one-half safe shutdown earthquake without any loss of function.

Reference:

1. Conax Report IPS-212, Rev. A and addendum to IPS-212, Rev.A. IPS-209, IPS-752, IPS-1348
2. TVA Design Specification WBNP-DS-1805-2697-00.

TABLE 3.10-4 (SHEET 1 of 5)

Unit 1 Only

WATTS BAR SEISMIC QUALIFICATION
SAMPLE OF BALANCE OF PLANT INSTRUMENTATION AND CONTROL
EQUIPMENT LIST

1. Transmitters
2. Power Supplies
3. Summing Amplifiers
4. Current-to-Current Converters
5. Square Root Converters
6. Alarm Units
7. Recorders
8. Indicating controllers
9. Manual Loading Stations
10. Panels (cabinets)
11. Dual Alarm Units
12. Square Root converters
13. Proportional Amplifiers
14. Millivolt Transmitters
15. Power Supplies
16. Current Isolators
17. Controllers (single case)
18. Controllers (dual case)
19. Setpoint Stations
20. Manual Loaders
21. Recorders
22. Hi-Fi Relays
23. Instrumentation Racks (local panels)
24. Single and Four Bay Instrument Cabinets
25. Lighting Panel Boards & Cabinets

TABLE 3.10-4 (SHEET 2 of 5)

WATTS BAR SEISMIC QUALIFICATION (Cont'd)

Equipment:	BOP I&C (See equipment list on Page 1 of this table, Items 1-10)
Mounting:	Several separate tests were run to qualify the instruments and cabinets. Each instrument type was tested separately (that is not mounted in the cabinets) and the cabinets were tested fully loaded with a representation of most instrument types installed. Instruments as well as cabinets were mounted directly to the Wyle Lab seismic test device.
Seismic Test:	An exploratory test was run in the form of a continuous sweep frequency search using a sinusoidal steady-state input at approximately 0.2 g. The search included two continuous sweeps from 1 to 35 to 1 Hz at a rate of one octave per minute. The specimens were then subjected to sine beat tests consisting of ten oscillations per beat with a time pause of approximately two seconds between each of the five beats. A sine beat test was performed two times, in each of the four orientations, at one-third octave frequencies of 1, 1.25, 1.6, 2.0, 2.5, 3.2, 4, 5, 6.3, 8, 10, 12.5, 16, 20, 25, 32, and at 35 Hz. The test level was 4.25 g's or greater, within the limits of the test machine, at a location near the driving point of the actuator. The 4.25 g input yielded an effective g force of 3.0 g's in both the horizontal and vertical axes, simultaneously.
Monitoring:	Simulated inputs were made during tests and outputs were monitored for each type instrument.
Results:	All instruments (and the cabinet) performed satisfactorily with no loss of function of ability to function properly before, during, and after the test.
References:	Wyle Lab Test Report Nos: 43522-1, 42434-1, 43280-1, 43675-1, 43859-1

Equipment:	BOP I&C (see equipment list on Page 1 of this table, Items 11-22).
Mounting:	All instruments were mounted in a rack mounting chassis which was modified for seismic use.
Seismic Testing:	1. A resonant search test was conducted in each of the three orthogonal directions. Each search consisted of two sweeps over the frequency range from 1 to 35 Hz and return to 1 Hz at a sweep rate of 1 octave/minute. The input "G" level to the vibration table was 0.5 g's. 2. A sine beat test was conducted at each resonant frequency found by step 1 and at the resonant frequencies found for the single and four-bay cabinets. The number of beats at each test frequency was 10 and the number of test frequencies cycled per beat was 10. The time between beats was of sufficient duration to preclude significant superimposition of motion. The input level was 3 g's horizontal and 2 g's vertical.

Table 3.10-4 (SHEET 3 of 5)

WATTS BAR SEISMIC QUALIFICATIONS (Cont'd)

Monitoring:	1. A 30 mA signal was applied to the input of each analog device requiring an external input. The output of the analog devices was monitored on a brush recorder. 2. The electrical contact of the discrete output devices were monitored for discontinuity (chatter) in excess of 100 microseconds.
Results:	There was no evidence of physical damage, contact chatter or output shift.
Reference:	Action Environmental Testing Corp. Report No. 10348-2, -3, -4, -5, -6, -7, -8, -9, -10, -11, -12, and -13.
Equipment:	Instrumentation rack (local panel).
Mounting:	The rack was bolted directly to the seismic simulator.
Seismic Testing:	1. Two low level (approximately 0.1g) single axis sine sweeps from 1 Hz to 35 Hz to 1 Hz were performed to establish major resonances for each major axis. The sweep rate was 1 octave per minute. 2. A sine beat test was conducted at each resonant frequency found by step 1. The sine beat tests consisted of five beats per test. Each beat contained two oscillations and was separated by a sufficient time span to allow all equipment motion to cease. The sine beat test levels were approximately 0.3 g for the vertical direction and 0.8 g for each horizontal direction. 3. The specimen was also subjected to a 45-second simultaneous horizontal and vertical inputs of multi-frequency random motion consisting of frequencies spaced 1/3 octave apart over the frequency range of 1 to 40 Hz in the front-to-back/vertical and the side-to-side/vertical orientation. The excitation was biaxial and phase incoherent. The amplitude of each 1/3 octave acceleration was independently adjusted in each axis until the Test Response Spectra (TRS) enveloped the Required Response Spectra (RRS). The resulting TRS was analyzed at 2% damping by a spectrum analyzer and plotted at 1/3 octave intervals over the frequency range of interest. Approximately three one-half level or greater Safe Shutdown Earthquake (SSE) tests were performed. A minimum of one full level test was performed after completion of the one-half level tests.

TABLE 3.10-4 (SHEET 4 of 5)

WATTS BAR SEISMIC QUALIFICATIONS (Cont'd)

Monitoring:	None of the devices mounted on the rack were monitored.
Results:	There was no evidence of any physical damage. The test indicated that the instrument mounting locations would not see "g" levels greater than 2 g's in the vertical direction and 3 g's in the horizontal direction during the postulated seismic event.
Reference:	Wyle Laboratories' Seismic Simulation Test Report No. 42807-1.
Equipment:	Single bay and four bay instrument cabinets.
Mounting:	The test samples were mounted by their normal mounting means to the test fixture.
Seismic Testing:	1. A resonant search test was conducted in each of the three orthogonal directions. Each search consisted of two sweeps over the frequency range from 1 to 35 Hz and return to 1 Hz at a sweep rate of 1 octave/minute. The input "G" level to the vibration table was 0.2 g's. 2. A sine beat test was conducted at each resonant frequency found by step 1. The number of beats at each test frequency was 10 and the number of test frequencies cycle per beat was 10. The time between beats was of sufficient duration to preclude significant superimposition of motion. The input level was 0.36 g's horizontal and 0.24 g's vertical.
Monitoring:	None of the devices mounted in the cabinets were monitored.
Results:	There was no evidence of any physical damage and in no case did the "G" levels exceed 3.0 g's in the horizontal axes or 2.0 g's in the vertical axis during the beat test.
Reference:	Action Environmental Test Corporation Report Nos. 10348, 10348-1
	Lighting Panel Boards and Cabinets
Electrical Rating:	125V DC Cabinets, 120/208V AC Panels, 60 Hz, 3-Phase
Mounting:	Specimen was wall-mounted with commercially available bolts, nuts, and washers to a wall-mounting fixture which in turn was welded to the test table. Mounting simulated the in-service configuration.
Seismic Test:	The specimens were subjected to the following tests: 1. Exploratory Test (Resonant Search) - consists of low level (0.2 g horizontal and vertical) biaxial sine sweep from 1-33 Hz in front-to-back and side-to-side orientation to determine major equipment resonance points. 2. Proof Test (Multifrequency) - Consisting of simultaneous horizontal and vertical incoherent inputs of random motion at frequencies spaced 1/3 octave apart from 1-31.6 Hz in front-to-back and side-to-side orientations. Aging was obtained with three half-level SSE tests followed by one full-level SSE test performed in front-to-back and side-to-side orientations.

TABLE 3.10-4 (SHEET 5 of 5)

WATTS BAR SEISMIC QUALIFICATION

Monitoring:	A multichannel recorder was used to monitor electrical continuity, contact chatter, and change of state before, during, and after the seismic test.
Results:	The specimen's structural integrity was not compromised and circuit continuity was maintained.
Reference:	Wyle Test Report 42979-1.

TABLE 3.10-5

ALLOWABLE STRESSES FOR CABLE TRAY SUPPORTS

Load Case	Allowable Stress	Load Combination[2]
Case I	AISC Allowable	D + L
Case IA	AISC Allowable	D + E
Case IB	1.5 x AISC Allowable[1]	$D + E + T_o$
Case II	1.5 x AISC Allowable[1]	D + E'
Case IIA	1.5 x AISC Allowable[1]	$D + E' + T_o$
Case III	1.5 x AISC Allowable[1]	$D + E' + P_a + T_a$

1. Allowable stresses are limited not to exceed 0.9 F_y, except for shear, which is limited not to exceed $0.52F_y$, and buckling, which is limited not to exceed 0.9 F_{CR}.

2. Key:

D = Deadweight

L = Live loads

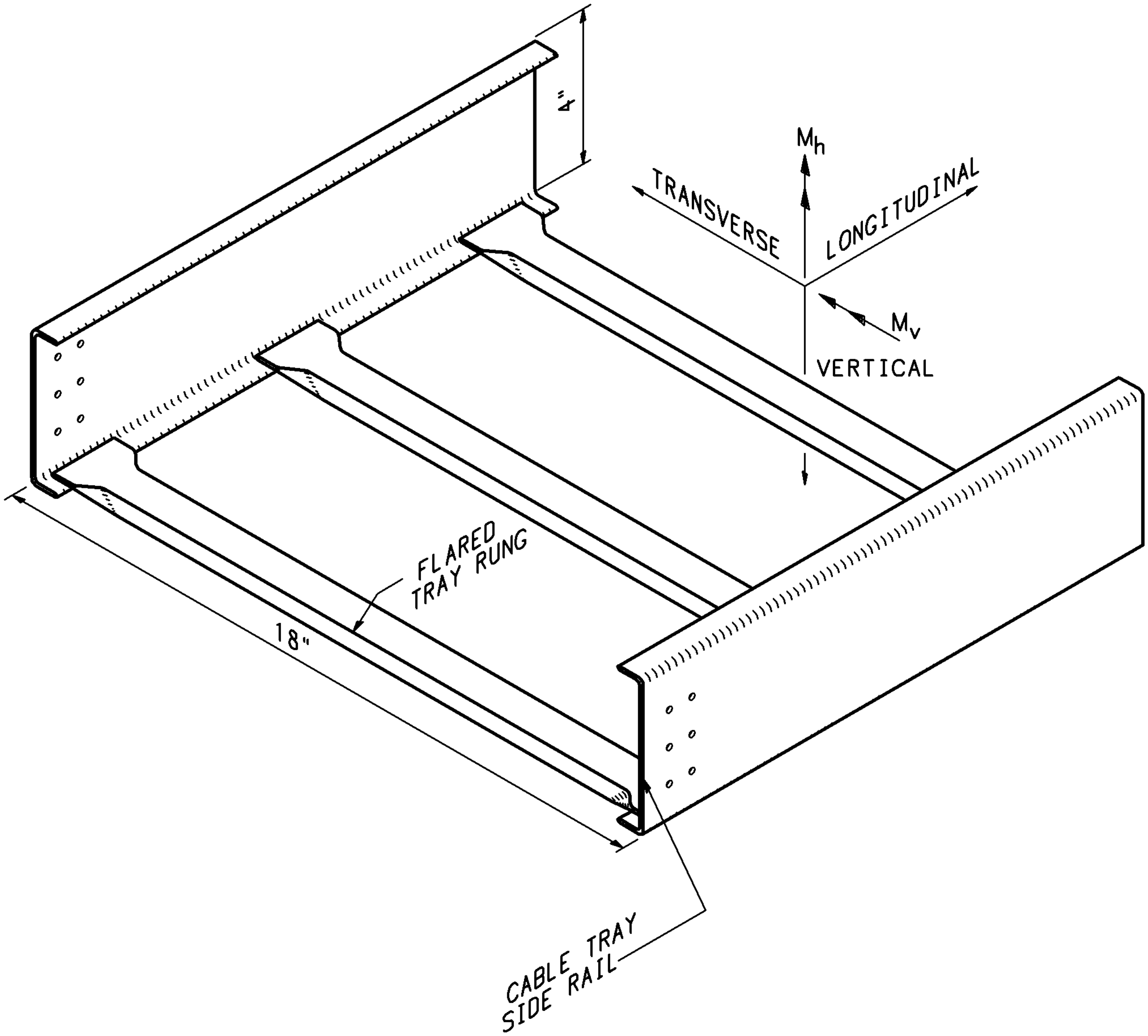
E = Operating Basis Earthquake (OBE) loads

E' = Safe Shutdown Earthquake (SSE) loads

T_o = Thermal effects and loads during normal operating or shutdown conditions based on the most critical transient or steady-state condition.

T_a = Thermal effects and loads during conditions generated by the design basis accident (DBA) transient condition and including T_o.

P_a = Pressure load effects from a DBA, such as steel containment vessel (SCV) dynamic movements and cavity pressurization.



M_h IS THE BENDING MOMENT DUE TO LOADS IN THE TRANSVERSE DIRECTION.

M_v IS THE BENDING MOMENT DUE TO LOADS IN THE VERTICAL DIRECTION (OUT OF THE PLANE OF THE TRAY).

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Orientation of Cable Tray Axes

FIGURE 3.10-1

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The method of assuring that mechanical and electrical components of safety-related equipment are qualified for their potential normal operational and worst-case accident environments is described in this section.

Two programs are in place to environmentally qualify safety-related electrical equipment (including cable) and active safety-related mechanical equipment to function or not fail for event mitigation. These programs involve:

1. Safety-related electrical equipment within the scope of 10CFR50.49.

2. Active, safety-related mechanical equipment located in a harsh environment.

Equipment within the scope of 10 CFR 50.49 excludes that equipment located in mild or essentially mild environments. A mild environment is defined as a room or building zone where (1) the temperature, pressure, or relative humidity resulting from the direct effects of a design basis event (DBE) (e.g., temperature rise due to steam release) are no more severe than those which would occur during an abnormal plant operational condition, (2) the temperature does not exceed 130°F due to the indirect effects of a DBE (e.g., increased heat loads from electrical equipment), (3) the event radiation dose is less than or equal to 1×10^4 rads, and (4) the total event plus the 40 year TID (total integrated dose) is less than or equal to 5×10^4 rad.[2]

The Mechanical Equipment Qualification (MEQ) program assures that active, safety-related mechanical equipment located in harsh environments will adequately perform the required design safety functions under all normal, abnormal, accident and post-accident environmental conditions in accordance with 10 CFR Part 50, Appendix A, General Design Criterion 4 (GDC-4).

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Identification of Safety Systems and Justification

Systems whose functioning is required to mitigate a loss-of-coolant accident (LOCA) or high-energy line break (HELB) for Watts Bar Nuclear Plant harsh environment areas are listed in Table 3.11-1. These systems were determined by identifying the systems upon which the safety analyses in the Final Safety Analysis Report and other referenced documents are dependent. Further, any systems which are necessary to support systems so identified were included in this table.

3.11.1.2 Identification of Equipment in Harsh Environments

The identification of the harsh environment is provided in Section 3.11.2. With the harsh environments defined, a survey of the safety-related electrical and active safety-related mechanical equipment in the affected areas was conducted. This survey was conducted using electrical instrument tabulations, mechanical piping drawings, mechanical heating and ventilation drawings, instrumentation and control drawings, electrical equipment drawings, and conduit and grounding drawings to identify the required components. The electrical components are identified in the Category and Operating Times Calculations. These calculations establish the 10 CFR 50.49 operating category and times for these components. 10 CFR 50.49 Category A or B electrical components located in a harsh environment are qualified in environmental qualification packages, which are referred to as the Environmental Qualification (EQ) Binders. These are a comprehensive set of documentation packages that demonstrate compliance with 10 CFR 50.49. In some instances Category A or B equipment may be contained in "Essentially Mild" (EM) calculations. EM calculations evaluate the 1E equipment located in plant harsh environments for the specific DBE(s). This evaluation concludes that for the specific DBE(s) for which the equipment must function, the environmental conditions (including normal plus accident dose) do not impose a significant environmental stress on the device.

Mechanical components are identified in the Mechanical Equipment List. This list identifies the active safety-related equipment which is required to perform a mechanical motion during the course of accomplishing a system safety function. This calculation identifies mechanical equipment in the portions of safety system flow paths which are required to mitigate 10 CFR 50.49 accidents. This equipment includes, but is not limited to valves, pumps, dampers, and fans.

Verification of qualification levels for equipment within the scope of 10 CFR 50.49 has been accomplished by a field walkdown of the installed components to provide traceability between the qualification documents and the in-situ equipment. This field verification walkdown is documented in the EQ binders.

The active safety-related mechanical equipment located within harsh environmental areas in the plant was identified by use of design data and confirmed by field verification walkdowns.

3.11.2 Environmental Conditions

3.11.2.1 Harsh Environment

Environmental conditions have been established for all harsh environment areas which contain safety-related electrical and active mechanical equipment exposed to a harsh environment resulting from a design basis event. Temperature, relative humidity, pressure, radiation dose, area type, chemical spray, and flooding were the parameters considered. (Only temperature and radiation were considered for mechanical equipment. The other parameters have no significant detrimental effect on mechanical equipment.) Values were based upon the following operational conditions:

1. Normal operating conditions - The environmental service conditions which the plant environmental control systems are designed to maintain on a normal design day.

2. Abnormal operating conditions - The environmental service conditions which result from outside temperature excursions, temporary greater than design heat loads, or degraded environmental control system operations. This condition can exist for up to 12 hours per excursion for non-Reactor Building spaces and will occur less than 1% of the plant life, unless alternate times and %0+ plant life conditions are specifically approved in Reference [4] and its associated environmental data drawings.

3. LOCA or HELB conditions resulting from small, intermediate, or large main steam line breaks inside containment.

4. High Energy Line Break conditions outside primary containment resulting from ruptures and critical cracks in various high energy lines throughout the Auxiliary Building and steam valve vaults.

5. Inadvertent containment spray initiation conditions resulting from accidental operation of the containment spray system.

6. Fuel Handling Accident (FHA)

The service conditions, resulting from the operational conditions listed above, are presented in Reference [4] and its associated TVA environmental data drawings. Temperature, pressure, and relative humidity vs. time curves are also provided on the drawings to clearly define the effects of various worst case HELB combinations on the area. These drawings include the environmental conditions for mild as well as harsh environmental areas.

For the purpose of 10 CFR 50.49, only design basis events 3, 4, and 6, above, are considered design basis accidents. Tornados, floods, or other natural phenomenon, including seismic, are expressly excluded from the scope of 10 CFR 50.49. Refer to Section 3.10 for Seismic Qualification.

3.11.2.2 Mild Environment

Mild environment qualification is applied to Class 1E electrical equipment only. Watts Bar satisfies the intent of NRC Generic Letter 82-09 by utilization of a preventive maintenance, surveillance, and testing program, as discussed in that generic letter.

For Class 1E equipment located in a mild environment and procured or installed before April 20, 1982 (date of issuance of NRC Generic letter 82-09), WBN demonstrates qualification by site preventative maintenance, testing, and surveillance programs.

For Class 1E equipment located in a mild environment and procured on or after April 20, 1982, WBN demonstrates qualification by the design or purchase specifications which identify environmental conditions and any other applicable design requirements as appropriate. These design activities are augmented by the site preventative maintenance, testing, and surveillance programs.

3.11.3 Electrical Equipment Within the Scope of 10 CFR 50.49

The process assuring that electrical equipment/cable is capable of performing its safety function is described in this section. A description of TVA's environmental qualification program is presented in Reference [1]. This reference provides documentation on the program and initial EQ binder preparation to denote that the components are qualified. TVA has implemented a program to ensure that all components will be fully qualified in accordance with 10 CFR 50.49 at fuel load. The EQ binders are maintained as controlled documents.

Safety-related electrical devices located in a harsh environment and required to function or not fail for mitigation of a specific DBA are identified on the Watts Bar Nuclear Plant 10 CFR 50.49 List (1E electrical equipment requiring qualification under 10 CFR 50.49). The methodology for establishing the 10 CFR 50.49 List for Watts Bar Unit 1 is located in Section III.2 of Reference [1]. The operating category, operating time, and safety function for the 10 CFR 50.49 devices are established by the Category and Operating Time Calculations. Devices on the 10 CFR 50.49 List are analyzed for qualification to the requirements defined by 10 CFR 50.49 and documented in the EQ binders.

3.11.4 Qualification Tests and Analyses

Qualification tests and analyses for safety-related electrical equipment were conducted in accordance with the requirements of 10 CFR 50.49 and the guidelines of NUREG-0588.[3] See Table 3.11-2 for compliance with NRC criteria and standards.

3.11.5 Qualification Test Results

Qualification test results are included or referenced in the EQ binder for safety-related electrical equipment in the 10CFR50.49 program.

3.11.6 Loss of Heating, Ventilating, and Air-Conditioning (HVAC)

Plant locations containing safety-related equipment that need a controlled environment to perform required accident mitigation operations are served by fully redundant environmental control systems, or operator actions specified to limit minimum and maximum temperatures (see Section 9.4 for details). Such redundancy and operator actions where specified, assure that no loss of safety-related equipment occurs from a single failure of HVAC equipment provided for controlling the local environment for this equipment. Data describing controlled local environmental conditions during accidents are valid for situations in which a loss of one train of HVAC is postulated.

3.11.7 Estimated Chemical and Radiation Environment

3.11.7.1 Chemical Spray

The worst case environment (normal or post-accident) chemical composition of the containment spray was based on the following sources and assumptions:

1. Ice Condenser

2. Cold Leg Injection Accumulators (4 tanks)

3. Refueling Water Storage Tank

4. Reactor Coolant System

The following assumptions were used in this analysis:

1. Calculations based on maximum pipe/tank volumes and boron concentrations and on minimum ice mass and sodium tetraborate concentration.

2. All solutions including completely melted ice mix completely.

3. Mass ratio of NaOH to boron in the ice is 1.85.

4. Density of borated water is equal to that of water.

5. Fission products, corrosion products, etc., will be neglected.

Results - The sources stated above yield a mixture of boric acid and sodium tetraborate with a pH greater than 7.5.[5]

3.11.7.2 Radiation

3.11.7.2.1 Inside Containment

The 40-year integrated normal operating dose and the maximum hypothetical accident doses are shown on the TVA environmental data drawings. The radiation exposure inside containment after a design basis LOCA was calculated based on a release to the containment atmosphere of 100% of the core inventory of noble gas, 50% of the core inventory of iodine, and 1% of the core inventory of solid fission products as determined by the ORIGEN computer code.[6] Removal of iodine is assumed to be due to interaction with the ice condenser only. The calculation of activity in containment after a LOCA is described in Section 15.5. Maximum gamma doses were calculated in the upper compartment, lower compartment, and ice condenser using a point-kernel-with-buildup computer code. Doses were integrated to determine equipment exposure for a 100-day period after the accident. Beta doses were calculated only for surfaces using the semi-infinite cloud equation in Regulatory Guide 1.4.

The calculation of radiation conditions inside containment complies with Paragraph 1.4 of NUREG-0588 except as noted below:

1. Paragraph 1.4(3) - The initial distribution of activity was assumed uniform throughout the containment even though the containment is broken up into upper compartment, lower compartment, and ice condenser. Air return deck fans are provided to aid mixing between these compartments.

2. Paragraph 1.4(5) - Natural deposition was not considered. Applicable deposition rates are unknown and actions of containment spray in the upper compartment, and steam condensation in the lower compartment can be expected to wash the deposited activity into the sump.

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For tritium production cores, the radiation exposure inside containment after a design basis LOCA was calculated based on a release to the containment atmosphere of 100% of the core inventory of noble gases, 50% of the core inventory of iodine, 1% of the core inventory of solid fission products and 100% of tritium as determined by the ORIGEN 2.1 computer code.[7] Following the same methodology as previously utilized, the resulting doses were determined to be less than those resulting from the previous determinations.

3.11.7.2.2 Radiation - Auxiliary Building Spaces

The normal operating radiation environment in the Auxiliary Building is shown on the TVA Environmental data drawings. The radiation exposure in the general spaces of the Auxiliary Building after a design basis LOCA is due to (1) containment sump fluid being circulated in the RHR, CS, and SI systems, (2) airborne activity in the Auxiliary Building, and (3) shine from activity in the containment. The source terms used for this accident are those determined by the ORIGEN computer code.[6]

Flow diagrams and equipment layouts were reviewed to determine the flow paths which would be used after an accident and to determine the volume and physical locations of contaminated fluids in the Auxiliary Building. The layout of the shield walls and equipment within the rooms were conservatively modeled. Source terms were calculated at various times after an accident. Dose rates were then calculated at several positions in the Auxiliary Building with respect to the contained sources and at various times after an accident.

The locations where dose rates were calculated were chosen to conservatively calculate the dose rates in corridors, outside equipment cubicles, in adjacent rooms, and within the equipment cubicles. These dose rates were then integrated to determine equipment exposure for a 100-day period after the accident. Airborne activity in the Auxiliary Building is due to gaseous leakage from the containment which is processed and exhausted through HEPA and charcoal filters in the Auxiliary Building gas treatment system (ABGTS). The dose rates through the Reactor Shield Building from activity released into the containment atmosphere were also calculated.

Radiation exposure due to a design basis FHA is due to airborne activity and shine from the affected spent fuel bundle and affects the refueling floor and the ABGTS room. Dose rates were calculated at a single position on the refueling floor and at several locations from the ABGTS filters. These dose rates were then integrated to determine equipment exposure for a 100-day period after the FHA.

The calculation of radiation conditions outside containment in the Auxiliary Building complies with Paragraph 1.4 of NUREG-0588.

REFERENCES

1. September 30, 1986, Letter from R. Gridley to B. Youngblood, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants - Summary Status Report - Watts Bar Nuclear Plant - Unit 1".

2. Watts Bar Design Criteria, WB-DC-40-54, "Environmental Qualification to 10 CFR 50.49," Revision 2.

3. NUREG 0588, Interim Staff Report on Environmental Qualification of Safety-Related Electrical Equipment, Revision 1, July 1, 1981.

4. Watts Bar Design Criteria, WB-DC-40-42, Revision 2,"Environmental Design".

5. WCAP-15699, “Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1 Containment Integrity Analyses for Ice Weight Optimization Engineering Report,” Revision 1, dated August 2001.

6. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. I-III, NUREG/CR-0200, Revision 5 (ORNL/NUREG/CSD-2/R5), March 1997.

7. Westinghouse Letter NDP-00-0288, dated June 27, 2000 and NDP-00-0291, dated July 14, 2000. (Unit 1 Only)

TABLE 3.11-1

SYSTEMS (OR PORTIONS OF SYSTEMS) REQUIRED TO MITIGATE LOSS-OF-COOLANT ACCIDENTS AND/OR HIGH ENERGY LINE BREAKS

Standby AC Power System (Includes Diesel Generators)
120V Vital AC System
Vital 125V DC Control Power System
Diesel Generator Fuel Oil System
Diesel Air Starting System
Emergency Lighting System
Auxiliary Control Air System
Nuclear Instrumentation System
Reactor Protection System
Hydrogen Recombination System (Unit 1 Only)
Containment Isolation Systems
Ice Condenser System
Containment Spray System
Residual Heat Removal System
Safety Injection System
Reactor Coolant System
Auxiliary Feedwater System
Containment Air Return Fan System
Essential Raw Cooling Water System
Component Cooling System
Main Steam System
Radiation Monitoring System
Chemical and Volume Control System
Emergency Gas Treatment Air Cleanup System
Auxiliary Building Gas Treatment System
Control Room Area Ventilation System
Engineered Safety Feature Coolers
Auxiliary Building Ventilation Subsystems:
- Shutdown Board Room Air Conditioning System
- Auxiliary Board Rooms Air Conditioning System
- Turbine-Driven Auxiliary Feedwater Pump Ventilation
- Shutdown Transformer Room Ventilation System

Spent Fuel Pool Cooling System
Main Feedwater System
Steam Generator Blowdown System
Feedwater Control System
Sampling System
Containment Lower Compartment Cooling System: Fan only (HELB only)
Reactor Building Purge Air Filter Trains
Diesel Generator Building Ventilation System

TABLE 3.11-2

COMPLIANCE WITH NRC CRITERIA AND STANDARDS

General Design Criteria 1	See Chapter 17
General Design Criteria 4	See Sections 3.5, 3.6
General Design Criteria 23	See Sections 7.1, 7.3
General Design Criteria 50	See Section 6.2
10 CFR 50, Appendix B, Criterion III	See Chapter 17
Regulatory Guide 1.30	Current activities generally conform to the requirements of N45.2.4.
Regulatory Guide 1.40	See Table 7.1-1
Regulatory Guide 1.63	See Paragraph 8.3
Regulatory Guide 1.73	See Table 7.1-1
Regulatory Guide 1.89	All 10 CFR 50.49 Equipment was qualified to IEEE 323-1971, or IEEE 323-1974
10 CFR 50.49/NUREG-0588	See References 1 and 3

3.12 CONTROL OF HEAVY LOADS

3.12.1 Introduction/Licensing Background

The Control of Heavy Loads program at WBN was established by the Generic Letter (GL) 81-07 Revised Response submitted as a letter from W. J. Museler (TVA) to U.S. NRC dated July 28, 1993 (T04 930728943). This letter superseded responses submitted to the NRC in letters dated February 6, 1984 and March 20, 1984.

In response to this submittal, the NRC concluded in Safety Evaluation Report Supplement 13 that WBN has adequately addressed NUREG-0612 guidelines.

3.12.2 Safety Basis

The safety basis for the Control of Heavy Loads is provided by assuring the risks associated with load-handling failures is acceptably low. This assurance is provided by meeting the requirements of NUREG-0612, Section 5.1.1, the use of an equivalent single-failure-proof crane for the reactor head lift.

3.12.3 Scope of Heavy Load Handling System

A heavy load for WBN is defined as any load weighing in excess of 2,059 lbs that is lifted in an area designated as a critical lift zone. Critical lift zones are those where an overhead handling system exists and the potential exists for a dropped load to impact irradiated fuel, impact safe shutdown equipment, or damage equipment required for spent fuel cooling. Overhead handling systems that meet these criteria are:

- Polar Crane
- Auxiliary Building Crane
- Intake Pumping Station (IPS) Hydraulic Pedestal Crane
- Hoists with capacities > 2059 lbs. as described in GL 81-07 Revised Response Table I

In addition, overhead handling systems were reviewed and excluded from this list on the basis that a load drop would not result in damage to any system required for plant shutdown or decay heat removal for one of the following reasons:

1. There is sufficient physical separation of the overhead handling system from any system or component required for safe shutdown or decay heat removal.
2. The system or component over which the load is carried is out of service while the load handling system is used.
3. The load weighs less than 2,059 lbs. and is not considered to be a heavy load.

3.12.4 Control of Heavy Loads Program

The Control of Heavy Loads Program consists of the following:

1. WBN commitments in response to NUREG-0612, Section 5.1.1 elements

2. For Reactor Pressure Vessel Head (RPVH) lifts, an equivalent Single-Failure-Proof crane

3.12.4.1 WBN Commitments in Response to NUREG-0612, Section 5.1.1

The control of heavy loads is performed by compliance with the seven guidelines outlined in NUREG 0612, Section 5.1.1.

These guidelines are met through the following:

Guideline	**Compliance Method**
1	Safe load paths - Safe load paths are as shown on drawing series 44W411. Directions contained within maintenance instructions provide requirements for control of any lift greater than 2,059 pounds, lifts in the auxiliary building, lifts in the upper compartments of the reactor buildings, and lifts at the IPS in those areas designated as critical lifting zones (CLZ). The critical lifting zones are defined as follows: 1. Reactor Building CLZ - the region inside the polar crane wall of the upper compartment El. 757’ and above as shown on drawing 44W411-7. 2. Auxiliary Building CLZ - all floor areas of the Auxiliary Building (AB) El. 757’ within the limits of hook travel of the AB crane 0-CRN-271-A1 as shown on drawing 44W411-5. 3. Intake Pumping Station (IPS) CLZ - area over Quality and/or Safety Related equipment required for safe-shutdown or decay heat removal within the IPS as shown on drawing 44W411-10. 4. Any area in which temporary hoists and rigging must be used for lifts greater than 2,059 pounds over operable quality and/or safety related equipment required for safe-shutdown or decay heat removal.

To control load movement, maintenance instructions direct the crane operator to raise and transfer the load to its destination, following safe load paths which have been designated by the 44W411 series drawings. To ensure that the established load paths are followed, all lifts performed per these instructions are done under the supervision of a designated individual (person-in-charge) who will verify the load path is clear prior to load movement. Deviations from approved load paths require prior approval of the plant operations review committee (PORC).

2 Procedures - Load handling procedures for the Heavy Load Handling Systems in Section 3.12.3 are contained in Maintenance Instructions. These instructions contain sections covering scope of control, references, prerequisites, precautions and limitations, acceptance criteria, performance, inspections, tables of approved heavy load lifts, and drawings identifying safe load paths. Tables of the various approved heavy load lifts identify the crane to be used, approved rigging or lifting devices, component weights, and reference drawings and procedures.

3 Crane Operators - Requirements for crane operator training, qualification, and conduct are contained in TVA Safety Procedures. The training includes:

1. Operating Practices and Functional Characteristics
2. Rigging Fundamentals
3. Electrical Maintenance
4. Certification Skills for Overhead cab-operated Cranes

These training programs incorporate all of Chapter 2.3 of ANSI (ASME) B30.2.

4 Special Lifting Devices - WBN Special Lifting Devices are any devices designed and dedicated to handle a specific critical load or loads, such as the reactor pressure vessel head lift rig and internals lift rig. Qualification of the head and internals lift rig devices is provided by WCAP 10313, including Addenda 1, 2, and 3, and inspection of these lift rigs is performed on a 10 year interval using Acoustic Emission Testing (AET). Acceptance of Acoustic Emission Testing for the lift rigs in lieu of the requirements of ANSI N14.6 was accepted by the NRC in a letter dated October 1, 1991 (A02 911007 002).

5 Lifting devices that are not specially designed - All slings and other lifting devices not specially designed used with cranes subject to NUREG-0612, Section 5.1, are designed, inspected, and tested in accordance with ANSI (ASME) B30.9 or ANSI N14.6, respectively. Evaluation of dynamic loads imposed by handling systems has been performed to determine if specialized selection and markings are required. Only one crane (IPS 20 ton hydraulic pedestal crane) was determined to generate dynamic loads in excess of 15% of rated load, with lifting devices used by this crane utilizing a dynamic factor of 20%. The only below-the-hook lifting device used with this crane is the stoplog lifting beam, which has been evaluated and shown to comply with the necessary design requirements. No special markings or selection criteria are necessary for the slings.

6 The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI (ASME) B30.2-1976 - Cranes and hoists at WBN are inspected, tested, and maintained in accordance with specific site maintenance (MI) and preventative maintenance (PM) instructions which implement the requirements of the applicable ANSI (ASME) standard. Each handling system as listed below has its own unique instruction or procedure to control inspection and testing. The load handling system and applicable standard are as follows:

Handling System	Procedure	Reference Standard
Polar Crane	MI	ANSI (ASME) B30.2-1976
Auxiliary Building Crane	MI	ANSI (ASME) B30.2-1976
IPS Hydraulic Pedestal Crane	PM	ANSI (ASME) B30.5-1989

7 The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976 and CMAA-70 - The actual design data for the auxiliary building crane and the reactor building crane were compared with the guidelines of CMAA-70 and ANSI (ASME) B30.2. Where specific compliance was not evident by review, an evaluation was made by imposing these guidelines on the actual design. Principally, this was the approach used for evaluating the design of major structural components by using load combinations and allowable stresses given in CMAA-70. The results of this review and analysis indicate that both cranes meet or exceed the requirements of CMAA-70 and ANSI (ASME) B30.2. The remaining overhead handling system subject to compliance with NUREG-0612 is the IPS hydraulic pedestal crane, which has been verified to be compliant with applicable industry standards.

3.12.4.2 Reactor Pressure Vessel Head (RPVH) Lifting Procedures

WBN maintenance instructions are used to control the removal and replacement of the reactor pressure vessel head. These instructions and TVA Safety Procedures contain requirements to ensure the single-failure-proof equivalency of the reactor building crane is maintained. These requirements include:

- Upper containment temperature is at least 70°F
- Periodic (at start of refueling outage) inspection of the crane has been completed.
- All safety functions of the crane are verified to be operational prior to performing the lift (per NEI 08-05).

The WBN reactor building crane was evaluated against NUREG-0554 as part of the station response to NUREG-0612, Section 5.1.3 (1) (and thus Section 5.1.6) compliance. This evaluation indicated that the crane is equipped with numerous single-failure-proof features. These features, as also described in part in UFSAR Section 3.8.6.1, include:

- Master Switches with Spring Return to Off Feature
- Cab Mounted Emergency Stop Buttons
- Floor Mounted Emergency Stop Buttons
- Overload Protection
- Overspeed Detection
- Dual Wire Ropes with a Factor of Safety Between 5:1 and 10:1
- Drum Safety Plates
- Two independent Holding Brakes of at least 125% of head lift hoisting torque each. Brakes apply automatically when power is removed from the hoisting motor.
- Dual interconnected gear trains
- Two Upper Limits Switches (2nd upper limit switch is a power disconnect)
- Stress Limits meet CMAA 70-1970
- Designed for Safe Shutdown Earthquake with the Maximum Critical Load

The reactor building crane wire rope does not provide a 10:1 factor of safety against breaking strength for the rated load. Thus, the reactor building crane is not fully single-failure-proof.

NEI 08-05 defines the requirements for an equivalent single-failure-proof crane for the purposes of lifting the reactor head. In addition to having the required safety features, the following equivalency measures are provided for the reactor head lift -

- Crane is a Class C design with a design margin of between 8% and 15%
- Ambient air temperature is at least 70° F
- All safety functions of the crane are verified to be operational prior to performing the lift
- Direct communications are provided between the Crane Operator, Person-In-Charge and Signal Person via headsets
- Emergency stop buttons are manned during lift
- Backup Emergency Stop Signal is provided
- Pre-job brief performed that includes identification of supervisory oversight, establishment of lift management protocol, acceptable travel limits of crane, verification of emergency stop button locations, and manning of emergency stop buttons
- Maintenance rule (a)(4) measures addressed in outage safety plan

With the equivalency measures provided in NEI 08-05, the reactor building crane is equivalent to single- failure-proof based for lifting the reactor head.

3.12.5 Safety Evaluation

Heavy load lifts at WBN are done safely and in accordance with NUREG-0612. Basis is provided by:

- Controls implemented by NUREG-0612, Section 5.1.1, make the risk of a load drop very unlikely.
- The use of an equivalent single-failure-proof crane makes the risk of a reactor head load drop extremely unlikely and acceptably low.
- The risk associated with the movement of heavy loads is evaluated and controlled by station maintenance instructions and the outage safety plan.