UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NO. 50-160

GEORGIA INSTITUTE OF TECHNOLOGY

1.0 INTRODUCTION

By application received by the Commission on March 11, 1968, and amendments thereto dated July 13, 1971, October 22, 1971, June 23, 1972, October 30, 1972, and November 13, 1972, the Georgia Institute of Technology (Georgia Tech) requested amendment of Facility Operating License No. R-97 to modify the Georgia Tech Research Reactor (GTRR) to permit operation at increased power levels up to 5 MWt. The modifications associated with the proposed increase in power level consist of the installation of: (1) additional heat exchanger capacity, (2) additional pumps and piping, (3) modified reactor safety system instrumentation and (4) an emergency core cooling system.

The GTRR is a heterogeneous, heavy water moderated, cooled and reflected reactor, fueled with fully enriched aluminum-uranium plates. This reactor is similar to the heavy-water reactors at Massachusetts Institute of Technology and Argonne National Laboratory (CP-5), both of which have been operated extensively at a power level up to 5 MWt. The GTRR had been operated at an initial power level of up to 1 MWt; however, it was designed and constructed for eventual operation at 5 MWt with the indicated modifications. The modifications were anticipated with the initial design and are proposed to be made at the present time under authority of a construction permit.

The original GTRR construction was authorized by Construction Permit No. CPRR-57 issued June 13, 1960, and subsequent operation of the facility was authorized by Facility Operating License No. R-97 issued December 29, 1964.

2.0 SITE

The GTRR is housed in a building complex known as the Nuclear Research Center located on the Georgia Tech campus in the city of Atlanta, Georgia. The facility is situated on a two-acre site, with the closest adjoining Georgia Tech building about 200 feet away and the nearest private residence about 490 feet from the GTRR containment building.

A restricted area is formed by a six-foot security fence and facility buildings with the closest public access at 45 feet from the containment building. An exclusion area is provided at the 490-foot radius (approximately 150 meters) from the reactor building. The Georgia Tech campus has a student population of about 7000 and is surrounded by residential and commercial areas. Approximately 30,000 people are within one mile of the facility. Site characteristics, including topography and drainage, geology, meteorology and seismology, have not changed from that reviewed during the original GTRR construction. Meteorological data for 1970 was submitted for determining the appropriate dispersion factors for normal operations and accident conditions as discussed in Section 4 and Section 6, respectively.

3.0 DESCRIPTION OF FACILITY

The GTRR is a heavy-water research reactor fueled with fully-enriched plates of aluminum-uranium alloy similar to the plate-type fuel of the Materials Testing Reactor. The reactor and associated systems are housed within a steel containment building. The major systems and components of the facility are described below.

3.1 Containment

The GTRR containment building is basically a cylindrical steel tank with a diameter of 82 feet, flat bottom and torispherical dome which is 50 feet above grade. Concrete, one foot thick and extending to a height of 34 feet above grade, at the inside surface of the steel container provides radiation shielding for personnel and structural support for a 20-ton polar crane. The containment building houses the reactor process equipment, ventilation equipment, electrical power supplies, experimental facilities, and the reactor control room. Penetrations into the containment are provided for personnel and equipment access and for process streams.

The performance of GTRR containment has been satisfactory during the 1 MWt operations to date. No change to the containment function or performance is necessary for operation at 5 MWt. Leakage rate tests have been made approximately every year since startup and are specified to be repeated annually during the proposed operation at 5 MWt. The specified allowable leakage rate has been and will continue to be maintained at one percent of containment volume per day at 2 psig. Conduct of the containment leakage rate tests generally conforms with the Type A test requirements of proposed Appendix J of 10 CFR Part 50. The results of the annual leakage rate testing has shown leakage rates of 0.45% to 0.7% of building volume in 24 hours at 2 psig.

In addition to the containment leakage rate test program, our review of the proposed 5 MWt operation considers effectiveness and reliability of containment isolation and adequacy of containment for the design basis accident (DBA). As a result of this review, Georgia Tech has agreed to provide a manually-operated isolation valve in series with each of two vacuum breaker valves to assure, through redundancy, the capability of closure of these two containment penetrations. The containment design pressure of 2 psig continues to be adequate for the maximum pressure to which the building could be subjected in the event of a DBA at the proposed power level of 5 MWt.

On the basis of satisfactory results of containment leakage rate tests, provision of additional valves for isolation and adequate design pressure for the new operating conditions, we conclude that the GTRR primary containment is adequate for the proposed operation at 5 MWt.

3.2 Reactor

The GTRR core is two feet in diameter, two feet in height and contains a maximum of 19 fuel assemblies spaced 6 inches apart in a triangular array. This fuel is centrally located and contained within a 6-foot diameter aluminum vessel that holds about 1100 gallons of heavy water (D_2O) which provides the core with a 2-foot thick D_2O reflector.

The reactor vessel is a 6-foot 4-inch high cylinder fabricated of 3/8-inch thick aluminum alloy. It is designed for a pressure of 9 psig. The vessel is mounted on a steel support structure and is suspended in a 2-foot thick graphite reflector which is surrounded and sealed by a steel shield tank. The biological shield, consisting of layers of boral, steel, lead and concrete, surrounds the encased graphite reflector and reactor vessel.

The fuel and core for the proposed 5 MWt operation are essentially the same as for the present 1 MWt operation except that additional fuel will be loaded to compensate for the additional reactivity requirements for 5 MWt operation. The standard fuel assembly (Mark II) for the proposed 5 MWt operation contains 16 individual, curved, aluminum-uranium alloy plates. The initial fuel (Mark I) contains 10 plates per assembly. The present core is made up of both Mark I and Mark II fuel assemblies. This type of combined loading will be discontinued for the proposed 5 MWt operation and all Mark I fuel will be removed from the reactor. The reference core size for the proposed 5 MWt operation is 16 Mark II fuel assemblies.

The reactor is controlled by four shim safety rods and one regulating rod. The shim safety rods are flat, hollow blades 5.5 inches wide by 1 inch thick and are made of cadmium clad with aluminum. The enclosed gas space is filled with helium and sealed. The regulating rod is a 24-inch long tube of cadmium encased inside and outside with aluminum alloy. The blades are mounted at the top of the reactor tank and swing between adjacent rows of fuel assemblies through an arc of 55 degrees. Each blade is driven through its arc by a drive shaft connected through an electromagnetic clutch to a drive motor. The reactivity worth of each blade is at least 0.09 delta k/k, with a range of 0.097 to 0.152 delta k/k shown by measurements made on a core containing 14 fuel assemblies. Since each blade controls at least 9% delta k/k and the excess reactivity is limited to 11.9% delta k/k, an adequate margin for shutdown is provided even if one blade fails to scram.

The core fuel and control rods have performed satisfactorily during the 1 MWt operation and there are no indications that their performance will not be equally satisfactory during operation at 5 MWt. Although heat flux and corresponding fuel temperatures are increased in proportion to the power level increase, these parameters are limited to values well below damage at 5 MWt operations. On this basis, we have concluded that the reactor core and control rods are acceptable for the proposed operation at 5 MWt.

3.3 Reactor Coolant Systems

The reactor is cooled by circulation of heavy water through the core. The heavy water flows from the primary heat exchanger to a plenum at the base of the core support assembly, and upward through the fuel elements. The heavy water flows from the top of the fuel bearing section of the fuel elements into the reactor vessel where it serves as both moderator and reflector. It is discharged from the reactor vessel through a pipe located in the lower head of the vessel to the suction side of the primary pumps. It is then pumped back to the primary heat exchanger where the heat transferred from the core is transferred to the light water secondary cooling system. The primary system is designed for 5 MWt to maintain the temperature of coolant entering the core at 114°F with a 16°F temperature rise through the core. At a design flow rate of 1800 gpm, the maximum fuel plate surface temperature when operating at 5 MWt has been calculated to be at least 8°F below the local D_2O saturation temperature.

The water level in the reactor vessel is maintained at a constant level by means of an overflow line installed in the vessel. A small flow through this overflow line to the D_2O storage tank is maintained. The heavy water is withdrawn from the storage tank, passed through a cleanup system, and returned to the primary system. In addition, the reactor vessel is connected to the D_2O storage tank by a top-reflector dump pipe which can drain the reactor vessel to a minimum of 1.0 inch above the top of the core. The top-reflector dump pipe is provided with two quick opening valves in parallel and can be used to drain the upper reflector of the reactor to reduce the core reactivity by 2.75% delta k/k.

To minimize degradation of the heavy water coolant with light water, a helium blanket is maintained at a slight overpressure in the D_2O storage tank and in the reactor vessel. A quantity of 10 cfm of the cover gas is continuously withdrawn, dried, and passed through a catalytic recombiner of heavy hydrogen and oxygen. Experimental data indicate that this system should limit the maximum equilibrium D_2 concentration to less than 0.1% which is well below the explosive concentration of 4.0%.

The modifications of the reactor coolant system required for 5 MWt operation consist of adding a heat exchanger and an additional pump in each of the primary and secondary coolant systems with the necessary process piping. The new heat exchanger will be connected in series with the existing one. The additional pumps will be of full capacity in both systems and will be installed in parallel with the present pumps. All of the additional equipment will be of a quality consistent with the present installation or consistent with current code requirements. On the basis of the above quality and that the additional components perform the same cooling function with the only change being that of additional capacity, we conclude that the proposed cooling systems are adequate for operation at 5 MWt.

3.4 Emergency Core Cooling System

The main modification associated with the proposed 5 MWt operation consists of providing an emergency core cooling system (ECCS) to sustain fuel element cooling in the event of loss of D_2O from the reactor vessel. This emergency cooling system consists of a 300-gallon storage tank located in the containment building at an elevation above the reactor tank so that gravity provides the required flow. Initiation of the ECCS is automatic with indication of low D_2O level in the reactor vessel or loss of electrical power.

Functionally, the D_2O emergency coolant will flow from the emergency storage tank through parallel stop valves in the gravity flow line to a circular distribution manifold in the reactor vessel. This emergency cooling system will be tested periodically to assure its operability. Isolation valves will be installed in the reactor coolant inlet and outlet lines to provide additional protection against fuel damage in the event of a loss-of-coolant accident (LOCA) and a simultaneous failure of a single active component in the emergency cooling system. The isolation valves function to retain the primary coolant level above the core. Additional cooling can be provided by connection of the city water supply to the emergency tank; a backup coolant supply will be provided by a gasoline-driven pump and fire hose using the fuel storage pool as the source of water. On the basis of our review, we have concluded that the emergency cooling system is adequate to prevent fuel melting in event of a LOCA.

3.5 Experimental Facilities

A number of experimental facilities penetrate the biological shield. Excluding fuel element penetrations and regulating rod and shim blade access ports, these include 22 vertical and 16 horizontal penetrations, a thermal column, and a biomedical irradiation facility. In addition, six horizontal penetrations are provided for the nuclear instrumentation system detectors. Two of the vertical penetrations have been designated fast flux facilities and, when so used, will contain a sheet of enriched uranium. The licensee has stated, however, that this facility has not been used and is not planned to be used; therefore, we have not considered utilization of the fast flux facilities in our review.

Ten of the horizontal penetrations are beam ports and are equipped with rotating shutters to protect personnel. Each shutter is sealed to prevent leakage of argon 41 from the beam port thimbles into the containment building ventilation system. The shielding provided for these facilities is adequately cooled to assure that no thermal effects jeopardize the integrity of the shield. These facilities and their use represent no change from that previous to the facility modification for 5 MWt operation; therefore, we find these facilities acceptable for the proposed operation.

3.6 Safety and Protective Instrumentation Systems

The reactor safety system has been redesigned to give independent redundant electronic scram circuits with electromechanical backup for each safety parameter. Its function is to terminate reactor power operation under any condition that might lead to excessive fuel plate temperature. Five critical parameters are sensed and used to initiate protective action to mitigate an accident condition. These parameters are reactor power, reactor period, D_2O flow rate, D_2O temperature, and D_2O level in the reactor vessel. Loss of electrical power also will cause a reactor trip. The protective instrumentation channels are arranged in two redundant groups and some sensor diversity is provided. There are fast (electronic) and slow (electromechanical) scrams provided. Fast and slow scrams are provided by signals from the reactor power and reactor period channels. A fast scram operates all four control blades as a group while slow scrams operate the control blades in groups of two. Two control blades are sufficient to shut down the reactor.

The primary containment isolation system (PCIS) functions to prevent excessive releases of radioactive gases or particulate matter. The PCIS is initiated by signals from anyone of four channels that are capable of detecting the radioactivity present in the gas or particulate matter in the containment building air as it is discharged to the outside or by a power failure.

The reactor safety system, the emergency core cooling system, and the primary containment isolation system are all actuated on loss of ac power. Emergency battery supplies provide power for the communication system and essential lights.

As described above, our review considered the safety related instrumentation, control, and emergency power systems which include the reactor safety system, containment isolation, emergency core cooling instrumentation, and emergency electrical power systems. The adeuqacy of these systems was evaluated on the basis of the Commission's General Design Criteria, the IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE-279) with appropriate consideration for the special nature, size, and power level of the GTRR. We have concluded that these safety and protective instrumentation systems are acceptable.

4.0 RADIOACTIVE EFFLUENT RELEASES

4.1 Gaseous Effluents

The reactor ventilation system brings fresh air through dual isolation valves located above the control room, mixes the fresh air with that recirculating and moves the total to the containment building. The exhaust air is ducted to the basement and combined with the exhausts from several experimental facilities. The effluent air is monitored by an on-line G-M tube instrument at the basement duct before entering the holdup line. The holdup line provides about 12 seconds delay at a flow rate of 4000 cfm. From the holdup line, the air passes through a filter bank of roughing and high efficiency particulate filters, and is discharged from the containment building through two exhaust butterfly valves to a dilution plenum at the stack. Prior to the dilution plenum in which the above described 4000 cfm exhaust is mixed with 30,000 cfm dilution fresh air, the air stream is monitored by a moving filter particulate monitor, a high sensitivity ionization chamber (Kanne System) and a charcoal cartridge sampler. From the dilution plenum, the total volume of air is discharged through the 76-foot stack.

Previous analyses of the effluent air have shown principally Ar-41 and O-19 radioactivity. The air monitoring system for the GTRR gaseous effluents will detect releases of radioactivity from the GTRR stack. Radioactive release limits will be specified in the Technical Specifications to meet applicable regulations based upon appropriate annual average meteorology. The licensee will be required to reduce radioactive gaseous effluents to as low as practicable for the GTRR operations and well within 10 CFR Part 20 limits.

4.2 Liquid Effluents

All liquid wastes which are potentially radioactive are passed directly to the waste retention tanks. Liquid from these tanks is analyzed and, if necessary, treated by the liquid waste treatment system (demineralizers and filters) prior to release to the city sewer. All highly radioactive liquid wastes will be collected and stored in designated control areas. If significant reduction in activity cannot be achieved, the liquid will be solidified and handled as solid wastes.

All liquid effluents in the waste retention tanks will be sampled and analyzed prior to release. Release limits will be specified in the Technical Specifications to meet applicable regulations. The licensee will be required to reduce radioactive liquid effluents to as low as practicable for the GTRR operations.

4.3 Solid Wastes

All solid radioactive wastes will be collected and stored in properly labeled areas. Periodically, these wastes will be packaged according to ICC specifications and shipped to an authorized receiver of radioactive wastes. Solid wastes containing high levels of radioactivity will be stored to prevent excessive exposure of personnel.

4.4 Environmental Program

An environmental monitoring program has been carried on since initial reactor startup. Since initial startup, the Radiological Health Section of the Georgia Department of Public Health has had fixed filter particulate air samplers operating at approximately 1000 feet from the reactor and has had film badges placed at 50 locations outside the reactor perimeter. For 5 MWt operations, the current program will be expanded to include additional film badges and thermoluminescent dosimeters placed in nearby buildings, inlet ventilation ducts and in those sectors most likely to receive the maximum whole body dose from cloud passage of Ar-41 as predicted by the meteorological conditions for the GTRR site. Special monitoring programs using ionization chambers will be made under specific meteorological conditions and known radioactivity releases from reactor operation.

5.0 CONDUCT OF OPERATIONS

5.1 Organization and Administration

The structure of the operating and engineering organization responsible for the GTRR will not be changed as a result of the proposed modifications and future operation at power levels up to 5 MWt. The use of the reactor will continue to be determined by the Chief of the Nuclear Sciences Division with the approval of the Director of the Engineering

Experiment Station and the President of the Georgia Institute of Technology. The GTRR is administered through the Head of the Nuclear Research Center. The Reactor Supervisor is responsible for safe operation and maintenance of the reactor.

All experiments and associated procedures must be reviewed and approved by the Reactor Supervisor prior to use. Procedures and proposed reactor experiments also must be reviewed by the Nuclear Safeguards Committee. This Committee is composed of seven engineers or scientists appointed by the President of Georgia Tech and the Charter of the Committee includes review of all operating, maintenance, health physics and emergency procedures.

The technical competence of the operating organization has been demonstrated to be satisfactory by its performance during operations at 1 MWt. Plans are included for assuring that the reactor operating personnel are trained for operations at the 5 MWt power level. In addition to their participation in the plant modifications, a series of formal training sessions will be employed to review basic principles, to explain the new emergency cooling system and instrument modifications, and to review the new procedural and design limitations and Technical Specifications. We have concluded that the operating organization and administrative controls, including provisions for experiments, procedures, operator training and safety review, are adequate for the proposed operations at 5 MWt.

5.2 Quality Assurance

The Quality Assurance (QA) program for the GTRR modifications and operations at 5 MWt is described in the October 30, 1972 submittal. The responsibility to define and implement the QA program is assigned to the GTRR Reactor Supervisor. He is responsible and has final authority in all quality assurance matters. The Nuclear Safeguards Committee will have prime responsibility for conducting comprehensive annual audits of the quality assurance program to determine the adequacy and effectiveness. While this organizational arrangement does not provide independence between operations and QA, we believe that it is acceptable for a facility and staffing of the GTRR complexity. We have concluded that the QA program associated with the construction modification of the GTRR is adequate and sufficient quality controls have been established to assure proper compliance with 10 CFR 50 Appendix B.

5.3 Emergency Plan

The applicant has outlined steps to be taken in emergencies. These include organizational responsibilities including coordination with offsite agencies, protective measures that would be taken in the event of a severe radiological emergency, and arrangements made for medical support. The planning also incorporates periodic reviews and updating, training, and drills in order to maintain the effectiveness of the plan.

We conclude that the licensee's plans for coping with emergencies in the unlikely event of an accident are adequate to protect the public health and safety.

5.4 Security Program

The applicant has submitted a proprietary document, pursuant to 10 CFR 2.790, which describes the principal features of a security program designed to deny, inhibit, or detect unauthorized access to the facility and to the containment, including periods when the facility is not occupied. Means for protecting fuel storage facilities and for monitoring the reactor top and beam port floor areas also are included.

We have concluded that the described security program provides reasonable protection for the GTRR.

6.0 ACCIDENT ANALYSIS

The applicant has analyzed the potential hazards resulting from accidents that could occur at the GTRR. These include loss of electric power, failure of the primary or secondary coolant pumps, failure of a single component in the nuclear or process instrumentation systems, failure of a shim safety rod to drop on scram, and failure of regulating rod such that the rod continues to withdraw without operator action. In addition, the applicant has analyzed postulated events, including simultaneous withdrawal of the regulating rod and a shim safety rod, failure of a shim safety rod by movement beyond the normal insertion limit in the core, reactor operation at the 5 MWt level with the shim safety rods at different elevations, and a refueling accident.

In the event of a break in the primary system piping, a LOCA could occur. The proposed emergency cooling system and primary coolant system isolation valves would act to prevent fuel melting. If the LOCA involved failure of the reactor vessel or failure of a nozzle on this vessel, the emergency cooling system would supply adequate coolant, initially from the stored D_2O followed by light water from the city water supply. We have concluded that the proposed ECCS is designed to provide adequate core cooling in the event of a LOCA at the GTRR.

The analyses of reactivity insertion accidents indicate that fuel melting would not result from accidents identified above involving rod withdrawals. The spectrum of reactivity insertion accidents analyzed included flooding of a beam tube and dropping of a fuel element into a just-critical core. These events involve a maximum reactivity insertion of approximately 2.5% delta k/k. These analyses are based on correlation with data obtained at the SPERT facilities. The analyses indicate that fuel temperatures would not exceed the clad melting temperature. Thus, we have concluded that no significant fuel melting would result from these reactivity insertion accidents.

The above described spectrum of accidents has been analyzed by the licensee to determine that the GTRR can be operated safely. On the basis of these analyses, a design basis accident (DBA) was established for the GTRR. The DBA is defined as the accident that is analyzed to result in the most severe consequences to the health and safety of the public.

The DBA for the GTRR is derived from a loss-of-coolant flow caused by blockage of a fuel element while operating at 5 MWt. Although a scram would most likely occur, we have assumed that scram is delayed sufficiently to cause the equivalent of one fuel element to be melted. Similarly with the reactor operating at 5 MWt, the inadvertent closure of a primary coolant isolation valve will result in loss-of-coolant flow; again, an assumed delay in the scram would result in fuel damage. We have concluded that the DBA at the GTRR would not exceed in severity release of the fission products associated with one fuel element situated in D_2O coolant within the reactor vessel.

In our evaluation of the consequences of the loss-of-flow accident, we have assumed a complete meltdown of one fuel element containing the maximum fission product inventory. The assumed fission product release fractions for MTR-type fuel with high fuel burnup under fuel melt conditions are 100 percent for noble gases, 70 percent for iodines, 10 percent for cesiums and less than 0.1 percent for other radioisotope families. Since the fission products would be released through the D_2O coolant, we assumed an iodine removal factor of 2 for the 2.5 feet of coolant above the reactor core. Since a pressure buildup in containment would not result from this DBA, it would be conservative to assume a leakage. We conservatively have assumed a leakage rate of 0.5 percent per day which is based on the Technical

Specification leakage rate limit of 1 percent per day at 2 psig over-pressure. We assumed a containment plateout factor of 2 for the iodines released from the reactor pool.

We have evaluated the GTRR meteorological data for 1970 and determined the appropriate dispersion factors for the DBA. The controlling dispersion mechanism within the exclusion area of 150-meter radius will be the mechanical mixing within the containment building and associated laboratory building wake. We have determined the appropriate X/Q value at 150 meters to be 7.7×10^{-3} sec/m^3 at the 5 percentile level which is equivalent to a Pasquill Type F stability class with a wind speed of 0.35 m/sec and a building wake dispersion correction factor equivalent to 7.5. The appropriate X/Q value at the low population distance (400 meters) for the same meteorological conditions is 4×10^{-3} sec/m^3.

We calculate a 2-hour whole body dose at the 150-meter exclusion distance following the DBA to be less than 0.9 Rem which includes the direct containment shine gamma dose of approximately 0.8 Rem and the finite cloud passage beta and gamma of less than 0.1 Rem. We calculate a 2-hour thyroid inhalation dose at the same distance of 4 Rem. We have evaluated the 24-hour whole body and thyroid inhalation doses at the low population zone distance (LPZ). The 24-hour thyroid inhalation dose at the LPZ could be 14 Rem without allowance for reduction in leak rate or wind meander. The 24-hour whole body dose at the LPZ would be approximately 0.2 Rem. These calculated doses are less than 5 percent of the guideline doses given in 10 CFR 100 for reactor siting and are a factor of 4 to 10 greater than the permissible annual doses for an individual in the general population as given in 10 CFR 20 for radiation protection in unrestricted areas. We have concluded that such doses do not represent a significant risk to the health and safety of the public.

7.0 TECHNICAL SPECIFICATIONS

With letter dated June 23, 1972, the licensee has submitted "Draft Proposed Technical Specifications for the Georgia Tech Research Reactor" dated May 26, 1972. These proposed specifications include safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features and administrative controls that are important to safety. The specifications are supported by analyses presented in the Safety Analysis Report and in the bases for each specification. The proposed Technical Specifications, as submitted, will be modified to satisfy regulatory requirements and will be incorporated in the license for operation.

We have concluded that the Technical Specifications, as modified, will provide appropriate limits and controls for safe operation of the GTRR.

8.0 CONCLUSION

Based on our review of the application, we have concluded that there is reasonable assurance that the GTRR can be modified and operated at 5 MWt as proposed without endangering the health and safety of the public.

FOR THE ATOMIC ENERGY COMMISSION

Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
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Date: December 19, 1972

AEC letter dated May 2, 1973, that issues the Construction Permit

ENCLOSURE 8