UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 6 TO

FACILITY OPERATING LICENSE NO. R-108

DOW CHEMICAL COMPANY

DOCKET NO. 50-264

1.0 INTRODUCTION

By letter dated October 16, 1990, and supplemented on November 19, 1990, Dow Chemical Company, (the licensee) submitted a request to amend Appendix A of Facility Operating License No. R-108. The requested amendment would (1) allow installation of the microprocessor based instrumentation and control system, and (2) change the timing for issuance of the Technical Specification (TS) required annual report.

2.0 INSTALLATION OF MICROPROCESSOR BASED INSTRUMENTATION AND CONTROL SYSTEM

The licensee plans to install, in parallel to their existing control console, the new digital microprocessor based instrumentation and control system. The transfer of control from the old to the new system (including scram) is planned to be via a series of gradual steps accompanied by tests which demonstrate the reliability of the new equipment while maintaining the proven performance of the existing control system. Upon completion of all testing, the new console will be used to control (except for the hardwired trip functions) both the safety and nonsafety aspects of reactor operation, and the old analog console will be disconnected. The primary functions of the new system remain the same as the old system: to monitor critical parameters and provide a scram signal when needed, to provide information to the operator; and to provide control for the reactor operation.

2.1 Hardware and Systems Assessment

This portion of the review focused on the areas of potential vulnerability or susceptibility of the new control console which might compromise its ability to present accurate information to the operator and to provide scram signals when required. No assessment was made of the reliability of the nonsafety-related controls. Issues investigated included single failure, environmental qualification, seismic qualification, surge withstand capability (SWC), electromagnetic interference (EMI), failure modes and effects, reliability, error detection, and independence.

The Safety System Scram Circuit consists of one analog nuclear power monitor channel (NPP-1000) and one microprocessor based nuclear power channel (NM-1000). Both the NPP-1000 and NM-1000 will provide a direct scram function at high power and the NM-1000 will also provide a scram on short reactor period. There is also a watchdog scram function that is included in the modification to assure proper operation of the microprocessor functions. All other scram functions (which include the high voltage setpoints on the two neutron detectors, as well as manual scram, keyswitch scram, and loss of power scram) will be identical to the current system.

Similar instrumentation and control systems have been accepted by the NRC in Amendment No. 19 for the Armed Forces Radiobiology Research Institute (AFRRI) dated July 23, 1990 and in Amendment No. 29 for General Atomics (GA) dated October 4, 1990. During the review, the licensee described the new system including licensing, engineering, testing and training aspects. The staff also had benefit of material from the previously mentioned reviews of AFRRI and GA.

The primary review criteria for instrument and control systems for research reactors are presented in ANSI/ANS 15.15 (1978) "Criteria for the Reactor Safety Systems of Research Reactors." The staff performed this evaluation also using criteria which apply to current vintage nuclear power plants. However, due to the inherent reactivity insertion safety feature of the TRIGA reactor design, minimal decay heat generation and the attendant minimal probability of fuel damage; the staff has concluded that these power plant criteria may serve as guidelines, but strict adherence to the power plant criteria is generally not warranted.

2.1.1 Environmental and Seismic Qualification

The new control system is to be installed in the control room and the reactor room. The system is to be constructed in standard commercial enclosures suitable for a mild environment. The operations at GA and AFRRI and testing at the Dow Chemical Company, to date, have not revealed any temperature or humidity problems that could be of concern to the components of the new system. Further, the staff considers the reactor room to be a mild environment when compared to power plant requirements and therefore, the entire system can be considered to be in a mild environment.

Though there have been no requirements promulgated for seismic qualification testing of research reactor control equipment, the staff considered the equipment for general ruggedness. The licensee indicated that the equipment is mounted in a commercial quality fashion which should prevent any significant movement of components within the console and racks. The primary concern remaining would be relay contact chatter which could prevent a scram when required. The safety system scram circuits for this system are conservatively designed to scram on failure (which includes contact chatter) and therefore the staff concludes that any further testing is not warranted.

Based on the above, the system is acceptable from the standpoint of environmental and seismic considerations.

2.1.2 Electromagnetic Interference (EMI)

The staff reviewed the susceptibility of the new equipment to EMI due to the potential for common mode interference which could disable more than one system at a time. For the new system, industrial-type isolators are generally used which prevent conducted EMI from being transmitted between the control and safety mechanisms. The neutron flux signal cabling is shielded to reduce the impact of radiated EMI. Previous experience with similar equipment provided by several different vendors at other facilities, and analysis of the installation of this new system has indicated that if EMI causes any disturbance in the system it will most likely cause a scram which is a conservative condition at this type of reactor. Based on the above, the staff concludes that EMI should not prevent a scram when required, EMI would most likely result in a scram which is a safe condition, and the design is therefore acceptable.

2.1.3 Power Supplies

The power supplies for the system are buffered to reduce the possible impact of minor power line fluctuations. The scram circuits for the new system are designed to scram when power is lost to them. The NPP-1000 are analog devices and will respond to power fluctuations similar to the existing analog equipment. The digital NM-1000 nuclear power channel uses a battery backed-up random access memory (RAM) to store data during loss of power. In addition to self-diagnostics, the NM-1000 has a watchdog timer circuit which puts the NM-1000 in a tripped condition and scrams the reactor if power fluctuations prevent proper software operation. The NM-1000 is also tested to verify that the system returns to proper operation following restoration of power. The staff finds this acceptable.

2.1.4 Failure Modes and Effects

In the previous reviews the staff considered a Scram Circuit Safety Analysis. This analysis identified the various ways in which the reactor safety system could fail. These include:

1) Physical System Failure (wire breaks, shorts, ground fault circuits)
2) Limiting Safety System Setting Failure (failure to detect)
3) System Operable Failure (loss of monitoring)
4) Computer/Manual Control Failure (automatic and manual scram).

This analysis was based on a fault tree approach which predicted failure to scram for various failure modes. Failures attributable to the unique failure modes of the software of the NM-1000 were considered. The analysis concluded that a failure of all safety systems and therefore failure to scram was extremely unlikely. The staff concludes that the failure modes and effects of the new system were acceptably addressed.

2.1.5. Independence, Redundancy and Diversity

The staff reviewed the data link between the safety channels and the nonsafety systems. The safety channels provide direct scram inputs and are also hardwired directly to independent indicators on the control console. The operators are provided with information from both the analog NPP-1000 power monitor and the digital NM-1000 monitor. The information is displayed on both direct wired bar graphs and on a graphic CRT. In addition, the safety channels provide inputs to the Non-Class 1E Data Acquisition Computer (DAC) through isolators. The isolators used have not been tested for maximum credible faults which the staff requires for power plant use, but have been tested by the manufacturers to standard commercial criteria. The DAC is then connected via redundant high speed serial data trunks to the Non-Class 1E Control System Computer (CSC) which interfaces with the operator by controls, a keyboard and CRT displays.

The scram circuit is essentially unchanged in that it maintains the fail safe design using the same automatic and manual contacts which open to remove power to the control rod magnets. This system has also added the computer watchdog scram and substituted the digital NM-1000 power and period scrams for similar analog scrams. The use of both analog and digital neutron monitoring, and the watchdog scram function provides additional diversity and redundancy to the scram system. The system as installed meets most of the requirements of IEEE-279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" and IEEE 379-1977 "Application of the Single-Failure Criteria to Nuclear Power Generating Station Class 1E Systems."

Based on the above, the independence, redundancy and diversity of the new system is acceptable.

2.1.6 Testing and Operating History

Extensive testing of the new system has been done at both GA and AFFRI. A significant number of design changes took place during the testing and phase-in of the new system. The staff has reviewed the problems discovered during testing of the system and has concluded that the resolutions appear acceptable. The staff also agrees with the licensee that long-term operability and safety is enhanced due to installation of equipment which has spare parts readily available. An additional improvement is the self diagnostics feature which allows continuous on-line testing and reduces the possibility of undetected failures.

2.2 Software Assessment

2.2.1 Verification and Validation Plan

The staff requires an approved verification and validation (V&V) plan for software which performs a safety function or provides safety related information to the operators.

2.2.1.1 Verification of Software Design

The NM-1000 software development by GA was reviewed by the staff to determine the acceptability of the V&V plan. The staff compared the GA V&V plan to Regulatory Guide 1.152 "Criteria for Programmable Digital Computer System in Safety Systems of Nuclear Power Generating Stations." Extensive testing of the GA microprocessor based instrumentation and control system at GA, AFFRI, and McClellan Air Force Base installations of similar systems proved the software to function as designed. GA also performed extensive V&V testing of the software, and discrepancies and operating problems. The staff has concluded that based on this previous review and experience, and the virtually identical nature of the applicable software, the verification of the new system is acceptable.

2.2.1.2 Validation Testing

The licensee's initial testing program was designed to show that the new system is capable of performing the assigned tasks. In addition to the initial testing program, the self-diagnostics feature allows continuous on-line testing, and the routine surveillance tasks normally performed by the reactor operators help to assure continued operation and rapid discovery of problem areas. Finally, a comprehensive procedure has been devised to provide testing and assurance that the new system will operate properly before relinquishing the scram circuits of the old system. The staff's review finds this testing program acceptable.

2.3 Maintenance and Surveillance

An extensive program is in place to evaluate the operation of the components of the control system. This program includes regularly-scheduled surveillance and test procedures which include those items required by the Technical Specifications as well as a number of other items. Major daily checks of all scram systems, channel tests of a number of information systems, and operation of the reactor at a specified low power are used to document the operability of the systems and to evaluate the operating characteristics of the reactor itself. Other tasks performed at monthly, semi-annual, and annual intervals include calibrations of important systems (including the control rods, the power levels, and the radiation-sensing instruments) as well as a number of non-safety-related parameters. These procedures, which have been in place for a number of years, assured proper maintenance of the present control system, and can be expected to provide continued assurance of proper operation and detection of problem areas with the proposed system. Based on this experience, as well as the self diagnostic features of the new console system, the staff concluded that the maintenance and testing program was acceptable for the new console.

2.4 Training of Operators

Reactor operations personnel have participated in training sessions at the General Atomics facility in San Diego. About 1/2 of the 40-hour training course was devoted to classroom training, covering details of the operation of the console, the structure and operation of the NM-1000 and NPP-1000 neutron channels, the software, the safety circuits, and the computers. The remainder of the time was used mostly in hands-on operation of the Dow console during its final stages of testing and adjustment - as a test stand operation, not connected to a reactor or neutron detectors - operating the computers, following the testing sequences, and going through the NM-1000 and NPP-1000 circuits in great detail, generally reinforcing the classroom instruction.

Finally, each of the operations personnel manipulated the controls and performed several operations on the GA 250 kW reactor (operations of this reactor as part of training programs for non-GA personnel have been approved by the GA Safety Committee and were performed under the direction of a licensed SRO of the GA facility). Operations included startup, intermediate reactivity changes, and shutdown. Although the GA instrumentation and control system is not identical to the proposed Dow installation, the steady-state operations are quite similar. The Dow people performed such operations as would be performed at the Dow facility, including startup, approach to power using relatively long periods consistent with the Dow Technical Specifications, manual operation at power, and automatic-mode steady-state operation. Particular attention was paid to observation and control of the period, which in the proposed system is generated by the digital NM-1000 circuit, as compared to the current Dow system where the period information is derived from the analog wide-range log power channel.

The staff concluded that this training was acceptable to assure operator understanding of the new instrumentation and control system.

3.0 CHANGE IN ANNUAL REPORT DUE DATE

In order to provide additional assurance that surveillance requirements are accomplished in a timely manner, the licensee has requested a change for the due date of the TS required annual report. The change would require issuance of the annual report in the first quarter of calendar year 1991 and annually, in the first quarter of each year thereafter. This change is an administrative change in the date of reporting to more directly coincide with annual surveillance requirements. This change does not impact on content of the annual report. Therefore, the staff concludes this change is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

4.1 Installation of Microprocessor Based Instrumentation and Control System

This amendment involves changes in a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.2 Change in Annual Report Due Date

This amendment involves changes in the category of recordkeeping, reporting, and administrative procedures and requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The staff has also concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

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Dated: December 13, 1990