ARKANSAS NUCLEAR ONE - UNIT 2

INADEQUATE CORE COOLING (ICC)

MONITORING SYSTEM

FINAL DESIGN DESCRIPTION

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ANO-2 ICC FINAL DESIGN DESCRIPTION

1.0 INTRODUCTION

1.1 Acronyms

1.1.1 RLI - Radcal Level Instrument
1.1.2 RGT - Radcal Gamma Thermometer
1.1.3 AP&L - Arkansas Power and Light
1.1.4 TEC - Technology for Energy Corporation
1.1.5 CET - Core Exit Thermocouple
1.1.6 ICC - Inadequate Core Cooling
1.1.7 ATC - Absolute Thermocouple
1.1.8 RVLMS - Reactor Vessel Level Monitoring System
1.1.9 LOCA - Loss of Coolant Accident
1.1.10 SMM - Subcooling Margin Monitor
1.1.11 SPDS - Safety Parameter Display System

1.2 Overview of the Inadequate Core Cooling (ICC) Monitoring System

1.2.1 System Purpose

In the ANO-2 installation, the Radcal Gamma Thermometer (RGT)-based monitoring system is designed to detect void formation in the reactor vessel head and upper plenum region, and to track reactor coolant inventory in accordance with NRC requirements.

1.2.2 System Components

The RGT-ICC system is based on a set of redundant reactor vessel level probes which contain RGT sensors located at selected intervals within the probes. The probe design is based on proven technology nearly identical to that used for local power measurements in several European reactors. Each probe consists of a cable pack of sheathed, single-difference thermocouples (one for each sensor), absolute thermocouples, and a heater cable; an annular core rod which contains the cable pack; and a jacket tube that surrounds the core rod/cable pack assembly.

Above-core monitoring will be accomplished with probes that traverse the entire region to be monitored. These probes are inserted through the vessel head by way of existing instrument ports. See Figure 1.

The portion of the RLI's that traverse the region above the core are surrounded by an instrument guide tube which serves as a manometer tube. The manometer tube is used to hydraulically isolate the coolant adjacent to the RLI's, thereby enabling the RGT to sense collapsed liquid level. Hydraulic isolation for narrow-range (dome vs. plenum) monitoring is to be accomplished by insertion of hydraulic isolators in the manometer tube.

ANO-2 presently has 42 radially distributed core exit thermocouples (CET's). These CET's are part of and are located in the upper portion of the in-core neutron detectors. See Figure 2 for the location of the incore assemblies. Additional details are provided in section 4.1.

Two channels of subcooling margin measurement and indication were installed in 1980 and were addressed in AP&L letter of January 18, 1980 (ØCANØ18Ø22). The details of the Subcooling Margin Monitoring (SMM) are included in section 4.1.

Data acquisition, storage and display functions are provided by a TEC 600 Data Acquisition and Control Unit. This is a modular, dedicated data acquisition and processing system with data display peripherals.

1.2.3 System Functions

The referenced sensors are located in the Radcal rods in positions designed to provide:

- advance warning of approach to inadequate core cooling (ICC);
- tracking of coolant inventory during an ICC event;
- tracking of coolant inventory during accident recovery;
- coolant temperature measurements at core exit and at the top of the upper dome region;

2.0 BACKGROUND

2.1 NUREG-0737 REQUIREMENTS

Following the March 1979 accident at Three Mile Island, the Nuclear Regulatory Commission (NRC) identified the need for additional instrumentation to detect inadequate core cooling (ICC). All power reactor licensees were required, by Orders issued in October 1979, to review and upgrade existing instrumentation. These Orders are to assure that information regarding the reactor coolant subcooling margin and core exit temperature over an elevated temperature range (detected by core exit thermocouples) is available to operators in the control room. Interim measures were taken to enhance the capabilities of these features.

The NRC also proposed that the possible need for additional instrumentation be studied to provide an unambiguous, easy-to-interpret indication of ICC, and that such instrumentation be provided, if it is found to be necessary. Design requirements and qualification criteria for additional instrumentation are specified in NUREG-0737, Clarification of TMI Action Plan Requirements. On October 21, 1980, power reactor licensees were required, pursuant to 10 CFR 50.54(f), to provide a report detailing their planned instrumentation system for monitoring ICC.

Following analysis of the information provided by the licensees, meetings with industry groups, and independent studies by the NRC staff, the Commission determined that during a small-break loss of coolant accident (LOCA), there is a period before the core has boiled dry (as indicated by core exit themocouples). During this time, control room operators would have insufficient information to clearly identify void formation in the reactor vessel head or to track coolant inventory in the vessel and primary system. Although the Subcooling Margin Monitor (SMM) gives early indication of a problem, it does not indicate whether the condition is improving or deteriorating.

The Commission concluded that the addition of a reactor coolant inventory system would improve the ability of plant operators to diagnose an approach to inadequate core cooling and to assess the adequacy of actions taken to restore core cooling. The benefit would be preventive in nature: The instrumentation would help operators to avoid a degraded or melted core in the event that voids and saturation conditions in the reactor coolant system occur as a result of insufficient cooling events or accident conditions.

In addition, the Commission concluded that the addition of a reactor coolant inventory system, coupled with upgraded incore thermocouple instruments and a subcooling margin monitor (SMM), would constitute an ICC instrumentation package that could significantly reduce the probability of incorrect operator diagnosis and resultant actions. It would be effective for events such as steam generator tube ruptures, loss of instrument bus or control system upsets, pump seal failures, or overcooling events originating from disturbances in the secondary coolant system. For low probability events involving coincidental multiple faults or more rapidly developing small-break LOCA conditions, the ICC instrumentation could also reduce the probability of incorrect operator diagnosis and resultant errors that could lead to a degraded core condition.

2.2 License Modification

Subsequent to the foregoing studies, the Commission determined that an instrumentation system for detection of ICC is required for operation of pressurized water reactors. Therefore, by Order for Modification of License dated December 10, 1982, (0CNA128211), Arkansas Power & Light Company (AP&L) was required to "...install an ICC instrumentation system consisting of subcooling margin monitors, core exit thermocouples and a reactor coolant inventory tracking system, all of which conform to the design parameters specified in NUREG-0737, Item II.F.2."

2.3 Implementation Activities

In response to the Order for Modification of License, AP&L developed a system conceptual approach to meet the NUREG-0737, Item II.F.2 design parameters and subsequently presented their approach to the NRC staff on March 31, 1983. Based on a favorable response, AP&L complied with Sections III.2 and III.3 of the Order on April 15, 1983, by providing detailed schedules for engineering, procurement, and installation of the inventory tracking system, and provided an Item II.F.2 conformance report for all components of the ICC instrumentation system (2CAN048306). In addition, each of eight questions posed by the NRC regarding the proposed system were answered in AP&L letter dated May 4, 1983 (0CAN058301).

The NRC documented by letter dated August 9, 1983 (2CNA088302), that the proposed Arkansas Nuclear One - Unit 2 (ANO-2) system provides a satisfactory basis for AP&L to proceed with the final design engineering and development program. This letter included nine additional questions regarding the system, together with a list of milestones relating to installation, testing and calibration, procedure development, operator training, turning on the system, and an AP&L Implementation Letter.

By letter dated August 26, 1983 (ØCANØ88314), AP&L provided a program plan which proposed the approach for developing a Radcal Gamma Thermometer (RGT)-based ICC Monitoring System. By letter dated September 12, 1983 (2CANØ983Ø6), AP&L provided responses to the additional nine questions contained in NRC's August 9, 1983 letter. The AP&L letter indicated that additional information on certain questions would be supplied at a later date. On December 13, 1983, AP&L letter (2CAN1183Ø3) provided further information regarding the August 9, 1983 NRC questions, provided an updated milestone schedule, and notified the NRC of the earlier-than-planned third ANO-2 refueling outage. This in turn mandated an earlier fourth refueling outage (2R4) which was the outage in which the in-vessel hardware would be installed. The earlier outage would require expediting final design, equipment delivery, emergency operating procedure development, and review and approval by the NRC.

Since the issuance of the Order for Modification of License for Arkansas Nuclear One, Units 1 and 2, AP&L has submitted several milestone schedules to the NRC staff. These efforts were an earnest attempt to meet the intent of the Order, to ensure sufficient and timely interaction with the NRC staff to allow approval of the ANO-2 ICC Monitoring System, and to ensure subsequent installation during the fourth refueling outage (2R4).

During the fourth ANO-2 refueling outage (2R4), AP&L removed two incore detector strings and installed two RGT probes in-core locations E-8 and N-8. The new instruments are an integral part of the ICC Monitoring System. These locations are identified in Figure 2.

The Phase I Confirmatory Test Program conducted a series of tests to provide data for licensing support and to provide plant-specific design data to be incorporated in the Phase II system hardware and software development.

The confirmatory tests were completed in June 1984. The preliminary test results were presented to the NRC staff on July 12, 1984, documented by letter dated August 30, 1984 (2CANØ88412), and discussed again in a meeting on October 17, 1984. Subsequent to the meeting, a revision to the Final Test Program Report was provided by AP&L in letter dated March 15, 1985 (ØCANØ385Ø2).

The NRC concluded that the proposed ICC Monitoring System would provide plant operators with a clear indication of approach to ICC conditions and a valuable indication of the effects of recovery measures. The NRC concurred by letter dated January 30, 1985 (ØCNAØ18529) that the test data demonstrated the "proof in principle" of the RGT design concept. Hence, AP&L was authorized to proceed with the final design and implementation of the ICC Monitoring System.

Although the start date for the ANO-2 fourth refueling outage (2R4) was advanced to mid-March 1985 because of enhanced unit performance during the previous operating cycle, AP&L remained committed to installing the ANO-2 ICC Monitoring System during 2R4. Many activities that would normally await approval by the NRC were pursued in anticipation of NRC approval. Several meetings between representatives of AP&L, TEC, and the NRC staff were held to facilitate communications and timely approval of the ICC system's conceptual and preliminary design. Based on those actions, fabrication of equipment proceeded to the point where hardware delivery and installation were accomplished in the Spring of 1985.

2.4 Schedule for Completion of ICC Modifications

In our letter of April 15, 1983 (2CANØ483Ø6), AP&L indicated that the in-containment cabling and connectors were environmentally qualified. After reviewing the qualification reports in greater detail, however, it was determined that all the requirements for environmental qualifications were not addressed. Therefore, a qualification program for the in-containment cabling and connectors for the CET's was undertaken. Since the cabling and connectors have passed previous testing, we are confident that they will pass current testing and thereby permit installation during 2R5. This was addressed in AP&L letter dated March 18, 1986 (2CANØ386Ø7).

3.0 DESIGN REQUIREMENTS

3.1 Functional Requirements

The functional requirements of the ANO-2 Inadequate Core Cooling (ICC) Monitoring System are to:

- Satisfy the NRC requirements given in NUREG-0737, Item II.F.2 for ICC monitoring. This is accomplished through the measurement of reactor coolant inventory and temperature above the core.

- Satisfy the NRC Order for Modification of License, dated December 10, 1982, to install an ICC Monitoring System for detecting and monitoring ICC conditions.

3.2 Operational Requirements

The operational requirements of the ANO-2 ICC Monitoring System are to:

- Provide capability for measurement of coolant inventory and temperature above core.

- Provide processing of signals from AP&L's existing core exit thermocouples.

- Provide capability for data transmission to AP&L's Emergency Facility Computer (i.e., Safety Parameter Display System).

- Provide capability for measurement of subcooled margin via subcooled margin monitors (SMM).

3.3 Separation Requirements

The redundant, identical sensors and instruments of the ICC Monitoring System are physically and electrically separated and channeled in accordance with the requirements of NUREG-0737 and other qualification documents.

3.4 Environmental Requirements

3.4.1 Control Room Environment

The control room portion of the ICC Monitoring System is designed to operate in the following environmental conditions:

	Normal	Accident Conditions
Temperature, F	75±10	105
Pressure, psia	14.7	14.7
Relative Humidity (noncondensing) percent	20-90	20-90
Radiation Integrated Dose, R	Negligible	Negligible

3.4.2 Reactor Containment Environment

The containment portion of the ICC Monitoring System is designed to operate in the following environmental conditions:

	Normal	Accident Conditions
Temperature, F	120	See Fig. 3
Pressure, psia	12.7-16.7	See Fig. 3
Relative Humidity, percent	90	100
Radiation Integrated Dose, R	2.5×10^6	1.1×10^8
Chemical Spray		
pH	na	See Fig. 3
Boric Acid, ppm	na	See Fig. 3
Sodium Hydroxide, ppm	na	See Fig. 3

3.4.3 Reactor Internal Environment

The in-reactor portion of the ICC Monitoring System is designed to operate in the following environmental conditions:

	Normal	Accident Conditions
Temperature, F	620	2300
Pressure, psia	2265	3000
Radiation Integrated Dose, ncm^2	3×10^{22}	--

Water Chemistry	Normal
pH (600 F) calculated	6.8 - 7.8
pH (70 F)	4.8 - 8.5
Boric Acid, ppm	0-14000
Lithium (Li), ppm	0.5-2.0
Chlorides, ppm	Max. 0.10
Florides, ppm	Max. 0.10

3.4.4 Design Life

All equipment, materials and components are designed for a period of 40 years of operation in the normal and accident environments specified above. Components for which a 40-year life expectancy could not be reasonably assured are designed to permit replacement. Technology for Energy Corporation (TEC) has prepared a list of all such components, identifying the expected life of these components, together with a test/replacement schedule for the system, to satisfy the service life design basis of 40 years.

Performance specifications for the individual hardware components are provided in Appendix 1 of Reference 11.1.3.

4.0 DESIGN FEATURES

4.1 Description

The ICC Monitoring System is based on the use of the Reactor Vessel Level Monitoring System (RVLMS), Core-Exit Thermocouples (CET's), and Subcooling Margin Monitor (SMM).

4.1.1 Reactor Vessel Level Monitoring System (RVLMS)

The RVLMS consists of redundant Radcal Level Instruments (RLI's) which contain axially distributed Radcal Gamma Thermometer (RGT) sensors and temperature sensors (ATC's). Associated cabling and signal processing hardware and software (DAS) are also part of the RVLMS.

4.1.1.1 Radcal Level Instruments (RLI's)

4.1.1.2 Radcal Gamma Thermometers (RGT's) Level Sensors

4.1.1.3 Temperature Sensors (ATCs)

The measurement of coolant temperature in the reactor vessel is based on absolute thermocouples (ATC's) placed in the RLI as shown in Figure 1. The absolute thermocouples are placed away from the high-heat output sections of the heater cable.

4.1.1.4 Data Acquisition System (DAS)

There is a data acquisition system (DAS) for each channel to acquire, process and display the ICC monitoring system data. The DAS includes a display containing the following information:

- Reactor coolant inventory above the core;
- Reactor coolant temperature at the core exit and the vessel head

Each DAS is installed in the control room in a Class 1E cabinet and includes an integral Class 1E display. The primary display the non-Class 1E Safety Parameter Display System (SPDS) is provided for operator use in the control room.

4.1.2 Core Exit Thermocouples (CET's)

The CETs, 21 for each channel of the ICC Monitoring System, are grouped in four channels, as follows.

CET Channel	Number of CETs	ICC channel
I (Red)	11	I
II (Green)	10	II
III (Yellow)	10	I
IV (Blue)	11	II

The Channel I and III CETs are processed by the Channel I ICC DAS, and the Channel II and IV CETs are processed by the Channel II ICC DAS.

The CET's are considered a complete system from the sensors to the monitoring and display devices. The signals from the CET's are inputs to the SPDS primary display in the control room. The displayed temperature range is 0°-2300°F. See Figure 4.

The cables for each channel are routed in separate, safety related raceways to the signal processing and isolation equipment. The equipment is mounted in seismically qualified cabinets located in the control room. All equipment for one channel is physically and electrically separated from the other channel.

4.1.3 Subcooling Margin Monitor (SMM)

Two channels of saturation margin measurement and indication were installed in 1980, as discussed in AP&L letter dated January 18, 1980 (ØCANØ18Ø22).

For indication of temperature margin to saturation, the calculator selects the highest temperature input. The calculator utilizes the pressure input as a pointer to locate the corresponding saturation temperature in steam tables resident in the calculator memory. The process temperature is subtracted from the saturation temperature and the difference is then available for recording and display. The pressure margin to saturation calculation is done in a similar manner.

The pressure input for each subcooled margin calculator is derived from redundant, safety-grade, wide-range (0-3000 psig) pressurizer pressure transmitters. These transmitters also provide the pressure signals to the Plant Protection System. The two temperature inputs for each calculator are from redundant, safety-grade wide-range (150-750°F) T_{Hot} RTD's in each loop. Redundancy requirements are satisfied by separate and redundant subcooled margin monitoring channels.

4.2 Response Characteristics

The ICC Monitoring System has the following response characteristics.

4.2.1 Data Acquisition System (DAS)

4.2.2 Radcal Level Instruments (RLI's)

In the RLI, sensitivity of the output signal to a change in water level is maximized within the limits of the heater cable power output and overheating considerations. The final signal will at least double when changing from covered to uncovered, and will be at least halved when changing from uncovered to covered. Sufficient change in signal to indicate the loss or recovery of water level at an ICC monitoring display will occur within 60 seconds of the actual loss or recovery of water level at that sensor.

4.2.3 Temperature

The temperature range of output for which calibration is maintained is 0 F to 1500 F. (The temperature sensors will survive to 2300 F and the displayed range reflects this capability; however, decalibration has been shown to occur above 1500 F for sheathed but unprotected small-diameter type K thermocouples. This is not considered to be a problem, because the NRC, in Regulatory Guide 1.97, Table 3, Note 3, states that type K thermocouples meet the requirements for core exit thermocouples. The decalibration has been shown to be significantly reduced if the thermocouples are protected. The thermocouples in the Radcal Level Instrument are protected by the jacket and core tubes. When the temperature returns to less than 1500 F, the Radcal Level Instrument has been shown to exhibit no decalibration effects.) The temperature range quoted (0 F to 1500F) is the range over which the calibration holds.

4.3 Data Acquisition System (DAS) Design Features

Data acquisition, storage and display functions are provided by a TEC 600 Data Acquisition and Control Unit. This is a modular, dedicated data acquisition and processing system with data display peripherals.

The primary display device is the existing non-Class 1E SPDS. The backup display device is a Class 1E qualified TEC 601 Plasma Display Unit. The display has four 16-character lines for display of alphanumeric data. Two lines are used for display of input or computed information. The other two lines are used for operator dialog. Currently diplayed data are dynamically updated. In addition to display callup, either display device allows the operator to perform the following;

- Modification of selected calibration constants (under password control);
- Modification of setpoints (under password control);
- Heater control.

The calibration constants are stored in non-volitile, electrically eraseable, progammable memory (EEPROM). Hence, although constants can be deliberately changed, they are not lost when system power is turned off or interrupted.

Data processing includes the following functions:

- Estimating collapsed liquid level using above-core RGT differential thermocouple data.
- Calculating absolute coolant temperature at the core exit and at the vessel head.
- Alarm checking and generation.

The installation provides two passive 20 milliampere RS232 serial ports, which supply a data link at 2400 baud to the SPDS.

Data processing for display and output consists of the following:

4.3.1 DAS Calculated Values

4.3.1.1 Level Calculation, General

4.3.1.2 Level Calculations, Slow and Fast Sensors

4.3.1.3 Level Estimation

4.3.1.4 Calculation of Absolute Temperature

Range: 0 F to 2300 F

4.3.2 Calibration

4.4 DAS Operation

4.4.1 Operating Modes

The ICC Monitoring System operates in several modes:

- Initialization and Startup;
- Normal Operation
- Local Command Interpreter; and
- System Reconfiguration and Device Editing (non-monitoring mode).

The Initialization and Startup mode begins when system power is turned on. The system executes a set of self-diagnostics that check the processor and memory. The system then starts the ICC monitoring software and begins the averaging and history processes.

Another mode of the ICC Monitoring System is System Reconfiguration and Device Editing. This occurs when a new device table is created or an old device is being changed or deleted. During this mode, all scanning of inputs and serial communication with the SPDS is suspended. Some limited changes to devices do not require entering the Device Edit mode.

Sensor setpoint adjustments and removal of a device from scan do not stop ICC monitoring operations. Furthermore, the 15-minute averages will run continuously and will not be stopped or started by any system mode other than "power off."

4.4.2 Power Supply Control

4.4.3 Sensor Functional Tests and Calibration

ICC functions may be stopped or started by assuming control of the system after entering the correct system password from the keypad/display. The following setup and control system functions can be accomplished after stopping the ICC monitoring process:

- Modify selected sensor and thermocouple calibration constants.
- Modify sensor thresholds and selected alarm limits.
- Turn off scanning of sensors or thermocouples (in the event of their failure)
- RLI heater on/off status.
- RLI heater on/off control to check reference calibration constants and to confirm RGT calibration with zero heater power.

The system can perform functional tests of the RLIs by manually changing the heater cable current and comparing actual changes in sensor signals with expected changes.

4.4.4 Validation

The system checks and validates all input signals. This validation consists of checking for short or open circuits. Invalid signals are identified and eliminated from data processing and display.

The operator is provided with a clear indication of invalid sensors and can manually flag an input signal as being invalid.

The DAS is capable of executing startup software self-diagnostics for the detection of hardware and software failures.

5.0 CONSTRUCTION FEATURES

5.1 Fabrication

5.1.1 Quality Assurance

Fabrication of all components was accomplished in accordance with the requirements of the TEC quality assurance program. Each component was fabricated to achieve the reliability and design lifetime specified in the design requirements. The manufacturer used all good practices and standards considered appropriate for fabrication of nuclear components.

5.1.2 Materials

Materials for the seal plug and parts welded thereto are described in detail in References 11.1.9, 11.1.10, and 11.9.

5.1.3 Welding

Seal-plug-to-Radcal welding was performed as described in References 11.1.9, 11.1.10, and 11.9.

5.1.4 Cleanliness

Cleanliness control during manufacture and testing of the RLIs was accomplished in conformance with TEC Specification 125-SP-03.

5.2 Electrical System

5.2.1 Installation

Connectors and cabling to transmit the instrument signals and heater power between the reactor vessel and the control room were field installed and terminated by AP&L to existing electrical penetrations in the containment boundary.

In accordance with NUREG 0737, Item II.F.2, all electrical equipment is Class 1E and includes these features:

- LOCA-resistant connectors;
- Radiation-resistant insulated thermocouple extension wires for the in-containment section between connectors; and
- Two independent channels with non-common fault redundance.

5.2.2 Power Source

5.2.3 Arrangement

5.2.4 Signal Conditioning

The signal conditioning includes the following functions:

- Reference junctions for absolute thermocouple signals;
- Amplification of the thermocouple signals from the millivolt range to the volt range;
- Continuous monitoring of all functioning thermocouple signals.

5.2.5 Performance Specifications

Performance specifications for the individual components of the signal conditioning system are provided in Appendix 1 of Reference 11.1.3.

Experience with in-core installations in French PWRs has shown that the signals can be transmitted to the control room without intervening amplification or other signal conditioning. Signal noise has been in the 1% to 2% range. The French experience is corroborated by the Halden Project experience.

At Halden, gamma thermometer signals are transmitted 150-200 meters from the reactor, inside a mountain, to the open air control room building. Experience shows that from the standpoint of signal quality and noise immunity, RGT signals can be transmitted in ordinary multilead cables with twisted pairs and a common sheath.

The following examples describe various kinds of signals encountered. Two types of thermocouple signals are involved: Difference and Absolute.

For the difference thermocouples, no reference junction is needed, and copper lead wires may be used. All the difference thermocouples from one rod are transmitted in the same multi-lead cable.

For the absolute thermocouples, chromel-alumel leads must be used and led to a reference junction outside the containment. The heater cables are powered from a power supply in the control room. The transmission cables must be capable of carrying currents in the range of one to five amperes. The heaters are supplied from separate, independent power supplies for each channel.

Because the heater cable is grounded to the sheath at the tip of the Radcal probe, and the sheath is in galvanic contact with the sheath of the other thermocouples, precautions are taken to avoid unintended ground loops.

5.2.6 Connectors

Each RLI has its own multipin connectors, which are connected by way of flexible conduit and multi-pair cable to the reactor containment penetration.

Electrical connectors are provided (inside containment) in the electrical cable that transmits signals from, and power to, the RLI in each channel. The electrical connectors are provided to facilitate removal of the reactor pressure vessel head during refueling.

The bottom end of each connector is located above the match point between the RLI and the instrumentation flange.

The connectors are sized so that the connector half attached to the RLI will pass through a 0.75 inch diameter circle to permit the seal plug drive nut and the instrumentation flange to be removed over them.

The connectors are designed and constructed to facilitate making and breaking the connection.

5.2.7 Transient Shield

A transient shield is included whose functions are to seal the flexible conduit that contains the RLI signal and power conductors and to provide an environmentally protected enclosure in which to make the electrical transition from the bundle of individual pairs that go from the RLI (in the flexible conduit) to the softline cable that runs to the containment penetration. The transient shield is environmentally qualified for Class 1E service for the in-containment environmental conditions.

5.2.8 Signal Cabling

5.3 Mechanical Construction

5.3.1 Instrument Nozzle and Closure Flange and Seal Plug

5.3.2 Hardware

5.4 Installation Details

5.4.1 Sensor Location

5.4.2 Radcal Level Instrument (RLI's)

5.4.3 Guide Tubes

6.0 TESTING PROGRAM

6.1 Prototype Testing

TEC and AP&L designed and conducted an extensive experimental program on the RVI to verify ICC monitoring capability, and to provide licensing support and design data for the system hardware for ANO-1 and ANO-2. The test program was conducted at the Oak Ridge National Laboratory (ORNL), using two atmospheric air/water test facilities and the pressurized water Forced Convection Test Facility (FCTF). The air/water facilities were used to provide manometer tube design data and basic sensor response parameters.

The FCTF is a typical reactor simulation facility that has been used in several NRC programs. It has both blowdown and reflood capability with sufficient control and instrumentation systems to perform the tests required to simulate a reactor under small loss of coolant conditions.

The detailed tests and analyses are presented in Section 6.2.

6.1.1 Overall Test Objectives

6.1.2 Air/Water Test Objectives

6.1.3 Upper Head Test Objectives

6.1.4 In-Core Test Objectives

6.1.5 Overall Conclusions

6.2 Post Fabrication Testing

6.2.1 Overall Testing

After fabrication, all sensors were functionally tested to determine that the sensor response was correct. Actual response was compared to expected results based on analyses and/or earlier testing.

DAS hardware and software were functionally tested to determine correct acquisition, processing, and display of the sensor signals. This testing included insertion of known input values to simulate sensor output signals. DAS output signals were compared to expected values. Channels I and II DASs were compared with each other to ensure that the same output signals occurred on each DAS for the same input signals.

The DAS System was functionally tested in November, 1985 after installation at ANO-2 in accordance with a factory acceptance test procedure. The results of the tests verified that the system conforms to the functional requirements.

The RGT sensors were calibrated after fabrication to determine the sensor thermal sensitivity. This sensitivity was combined with the analytically determined coupling coefficient to produce overall instrument factors. The RGT sensor calibration simulated gamma heating by volumetric (or Joule) electric heating, and also by using the integral heater cable.

6.2.2 Calibration Tests

6.2.2.1 Constancy of Calibration

At the Savannah River Project (SRP) gamma thermometers have exhibited constant calibrations with a fast fluence equivalent to five and one-half years in a PWR (with 1.5% calibration accuracy). Gamma thermometers at the Halden heavy-boiling water reactor (HBWR) have held constant calibration accuracy over seven years of irradiation with no observable changes in signal relative to the power in surrounding fuel. At HBWR the fuel loading is so variable and heterogeneous that the uncertainty abbreviations associated with this observation is larger than at SRP and is estimated by the applicants to be plus or minus 5%.

A highly documented, well-controlled exposure test of RGT specimens has been undertaken at a test reactor facility. It has been calculated that the 10 specimens being irradiated there will change calibration less than 5% after a fast neutron exposure of $6.2E^{21}$ neutrons per cm^2 (equivalent to three years in a PWR).

6.2.2.2 Accurate Out-of-Reactor Calibration

The sensitivity of RGT signals to heating of the sensors can be measured in a laboratory by direct electrical heating or by time constant determination. In practice to date, a variation of plus or minus 1.5% in mean sensitivity has been demonstrated in French PWRs. Individual chambers have shown high linearity of signal (correlation coefficient greater than 0.9999 to the best fit straight line) in both room temperature and high-temperature (300 degree C coolant) electrical heating calibration.

6.2.2.3 Large Signals

6.2.2.4 In-Core Recalibration

6.2.2.5 RGT Calibration

7.0 INSTALLATION, TESTING, AND MAINTENANCE

7.1 Installation

7.2 Installation Tests

7.3 Future Modifications

8.0 TRAINING AND PROCEDURE REVISIONS

8.1 Training

Training programs for the system have been conducted utilizing both operation and maintenance manuals. Each training course reviews and uses appropriate manuals to ensure that trainees are familiar with them. Training courses simulate expected plant and system activities, including failure modes (to the extent possible or practical). On completion of the courses, evaluation forms were used to ascertain the trainees' comprehension of the course materials.

Three separate training programs have been conducted; for system operators, for personnel responsible for system hardware maintenance, and for personnel responsible for system software maintenance. These courses were initiated immediately following factory acceptance testing.

Additional training will be conducted after completion of EOP revisions and necessary operating procedure modifications.

8.1.1 Operator Training

This course provided a general system overview from an operations standpoint. Block-diagram level discussions were held regarding system configuration, system startup, and alarms. These discussions provided the information required to operate the system from any operator device.

8.1.2 Hardware Maintenance Training

This course began with an overview of the system configuration and communication logic of the system. Included were sessions on calibration, display functions, control functions, and diagnosing of system and/or memory problems.

8.1.3 Software Maintenance Training

This course provided an overview of the and system software. System software configuration and database philosophy were covered, which included system startup, device tables, data acquisition, and diagnostics.

8.2 Control Room Design Review (CRDR)

In accordance with the Order for Modification of License, the task analysis portion of the CRDR has been completed.

8.3 Human Factors Engineering (HFE)

NUREG-0737, Item II.F.2 (ICCs and CETs) requires that: Types and locations of displays and alarms should be determined by performing a human-factors analysis, taking into consideration:

- the use of this information by an operator during both normal and abnormal plant conditions;
- integration into the EOP;

- integration into operator training; and
- other alarms during emergency and need for prioritizing alarms.

HFE reviews have been conducted in order to comply with the requirements based upon present project status. Additional reviews will be conducted as necessary.

8.4 Emergency Operating Procedure (EOP)

EOP Technical Guidelines addressing ICC monitoring are currently being developed. After NRC approval of the guidelines, the station EOP will be modified to provide guidance for utilizing RVLMS.

9.0 QUALITY ASSURANCE

9.1 Arkansas Power and Light Company (AP&L)

AP&L's quality assurance program, as documented in the Quality Assurance Manual, APL-TOP-1A, Revision 7, complies with its referenced Regulatory Guides.

9.2 Technology for Energy Corporation (TEC)

The RVLMS was designed, manufactured, and tested in accordance with the TEC Quality Assurance Program. This program addresses the requirements of ANSI N45.2-1977, subsequent applicable daughter standards, and 10 CFR50, Appendix B. The implementing procedures used in TEC's quality assurance program are contained in the TEC QA Manual. The procedures comply with ANSI Standard N45.2 insofar as they are applicable to the project requirements. TEC's quality assurance program applies to all contracts or QA purchase orders designated by TEC's customers as having quality assurance requirements.

TEC applied the referenced QA program during manufacture of the test specimens and during all phases of design, manufacture, testing and equipment qualification. Furthermore this QA program was implemented during the manufacturing and testing of all components of the deliverable RVLMS to AP&L.

The Quality Assurance Manual, Corporate Engineering Procedure CEP-106, entitled Software Design and Control, and lower-tier departmental procedures were implemented for the control of all software and programming work. Qualification plans and procedures for all verification and validation requirements and hardware qualification efforts have been, and will continue to be documented. Qualification test reports were provided as part of the final documentation.

All shipping, packaging and handling has been and will be performed in accordance with ANSI N45.2.2 Level B. Further details are in TEC's Quality Assurance Plan and Quality Assurance Manual. The TEC Quality Assurance Manual was submitted to AP&L for review prior to initiation of equipment fabrication.

9.3 Safety Classification

The ICC Monitoring System is classified as Class 1E and satisfies the requirements of IEEE 323-1974, IEEE 344-1975, and IEEE 384-1981, except for the SPDS, which is classified as non-Class 1E. The system is classified as Seismic Category I, except for the SPDS.

Primary pressure retaining parts and parts welded thereto are Class I components, in the context of Section III of the ASME Boiler and Pressure Vessel Code, with the exception of those items described in Sections 5.1.2 and 5.1.3 of this document.

9.3.1 Core Exit Thermocouples

ANO-2 has 42 remaining core exit thermocouples located at the top of the in-core instrumentation assemblies. The sensors, electrical connectors, and signal processing equipment are currently non-Class 1E.

The CET instrumentation system is projected to be upgraded during 2R5. As indicated in AP&L letter dated September 12, 1983, (2CANØ983Ø6), the ANO-2 incore instrument cables and connectors were procured as prefabricated cable assemblies. The procurement specification required that the assemblies be manufactured and tested in accordance with IEEE Standards 323-1974, 344-1971, and 383-1974. Bendix Corporation supplied the prefabricated assemblies and developed test procedures to demonstrate that the materials and assemblies complied with the procurement specifications.

9.3.2 Test Results

Radiation, LOCA and seismic tests were performed by Wyle Laboratories in Huntsville, Alabama. The results of this testing are documented in Test Report #43117 according to Test Plan #541/3996/CP and show that the cable assemblies are capable of withstanding, without compromise of structural or electrical integrity, the prescribed simulated environments.

The incore instrument assemblies, which contain the CET's have been designed to withstand the operating environment inside the reactor. A qualification program for the in-containment cabling and connectors for the CET's is presently underway. Based upon the fact that the cabling and connectors passed the previous testing, we have a high degree of confidence that they will pass this testing and therefore allow installation during 2R5.

10.0 CONFORMANCE ANALYSIS

The following analysis is presented in a format consistent with that contained in NUREG-0737, Item II.F.2, Attachment 1.

10.1 Reactor Vessel Level Monitoring System (RVLMS)

The RLI will give an early warning of the approach to inadequate core cooling (ICC). The sensors are axially located to provide optimum resolution in the areas of most concern. The system was intended to incorporate a segmented manometer tube that would differentiate the upper head region and the plenum region.

In the upper head region, the stilled liquid level could be monitored with the reactor coolant pumps running; while, in the plenum region this would not be possible because of turbulent flow conditions. However, because of difficulties encountered during the effort to machine the required porting in the guide/manometer tubes, level differentiation between the plenum and the upper head is not possible. See Section 7.2 for details regarding results of the installation tests.

Each RLI contains multiple sensors distributed axially along its length. The uppermost sensor is an absolute thermocouple located just under the reactor vessel head. Except for two other absolute thermocouples located at core exit and entrance, the remaining sensors are differential heated thermocouples of various configurations for local sensing.

10.1.1 Advantages

The advantages of using RLI's for level measurement are:

- No density compensation is required.
- There is a large and easily detectable signal change with a phase change of coolant surrounding the sensor.
- The RLI is simple in design and sturdy in construction.
- The signal transmission system design is straightforward.

Placing an absolute thermocouple in the RLI just under the reactor head provides a unique benefit. The sensor monitors temperature at the metal-to-liquid interface of the reactor head, which is influenced by the sensible heat stored in the reactor head. This temperature measurement will provide valuable information concerning the influence of this sensible heat on coolant conditions at the metal-to-liquid interface. Use of these temperature measurements by operators may help to prevent inadvertent steam bubble formation during natural circulation cooldown and during repressurization after a small break LOCA.

10.1.2 Environmental Qualification

The RLI's signal transmission system and signal processing system up to and including the isolators, are seismically and environmentally qualified as described in Section 9.3. The SPDS is non-Class 1E, but is backed by a Class 1E TEC-601 Plasma Display Unit and Keypad.

10.1.3 Single Failure Analysis

For upper head level measurement, two RLI's have replaced two in-core instrument assemblies. Electrically, the two RLI's are separate, and two channels of diverse cable routing are maintained all the way to the Class 1E DAS unit. The non-Class 1E display (SPDS) is supplied (through isolators) from the DAS.

10.1.4 Class 1E Power Source

All portions of the system are provided with Class 1E power with the exception of the SPDS computer (with its own inverter power supply and battery backup).

10.1.5 Availability Prior to an Accident

Because the RVLMS is composed of two redundant channels, one channel may be temporarily removed from service for surveillance testing, calibration or maintenance while the other channel remains in service, thus insuring system availability.

10.1.6 Quality Assurance
See Section 9.1

10.1.7 Continuous Indication

Reactor vessel level information will be continuously provided to the DAS and the SPDS along with other inputs. Any information provided to the SPDS is available to be displayed in a human-factored color graphics console at operator request. The information is also displayed on the DAS Keypad/Display Unit's four line, 16 character per line display. Currently displayed data are dynamically updated.

10.1.8 Recording of Instrument Outputs

Recording of reactor vessel level information is performed by the trending capabilities of the SPDS computer. The trend information is stored by the computer and is available for display on demand. The DAS trend buffer maintains a constantly updated series of recent averages that are also available on demand.

10.1.9 Identification of Instruments

The DAS is a dedicated, modular unit for which the control panel serves only ICC monitoring functions. As such, it is clearly identifiable for the purpose of obtaining reactor vessel level information. The TEC 601 unit is useful for monitoring and assessing accident conditions.

10.1.10 Isolation

The RLI s, signal transmission system and display/control unit are all Class 1E components. Where isolation is required, as in the output to the SPDS, qualified isolation is provided.

10.1.11 Channel/Monitor/Sensor Check and Test

Periodic validation of sensor signals, monitors and channels is possible during operation. For the purpose of surveillance testing or repair, one channel of the DAS may be taken out of service for a short period without affecting the operation or the validity of signals in the other channel.

By using the heater current indication, step changes in heater current added to the existing sensor input signal should result in a proportionally identical increase in output signal on both channels. Hence, channel signals can be checked for identical response, and, if necessary (under password control), constants can be changed during plant operation to correct discrepancies.

The DAS is capable of executing startup software self-diagnostics for the detection of hardware and software failures.

Except for the in-containment components, the RVLMS can be repaired during plant operation. There is no in-containment signal conditioning equipment.

See Section 4.4.4

10.1.12 Access Control

The DAS maintains a System Password List with provision for 20 passwords. The first password in the list is the System Manager password. Timed passwords are provided to enable maintenance personnel access for a specific period. A timed password is valid until the expiration date and time entered for that password. Reference 11.1.5 provides a detailed discussion of the password system.

10.1.13 DAS Adjustments

Access to setpoints, calibration constants and test points for the RVLMS is under password control.

10.1.14 Invalid Indications

When the DAS is turned on, but not yet initialized, it begins its sensor scanning and trend buffer updating but is not considered to be in operation until it is initialized by means of a reset button. During the time between power-on and operation, the DAS will not display any false values.

During operation, the DAS removes from service and discontinues value displays for any failed sensor or shorted or open circuit. Such failure is clearly flagged. The operator, under password control, can remove from service any failed sensor.

10.1.15 Direct Input

Signals from the RGT sensors are input directly into the DAS without prior signal amplification or noise filtering. A reference junction is required for the absolute thermocouples.

See Sections 4.2.2, 4.3.1, and 6.2.2.3.

10.1.16 Normal Use

The RVLMS may be used by the operators during the filling and venting phases of reactor startup, as well as the cooldown and draindown phases of reactor shutdown. The absolute thermocouples provide temperature information during normal operation.

The temperature range for maintenance of calibration of the sensors is 0 F to 1500 F. Thus, they will provide reliable indication of water temperature in the plenum and head regions.

See Section 4.2.3.

10.1.17 Periodic Testing

The RVLMS is designed to accommodate periodic testing. As part of a Class 1E system, the ICC instruments will be subjected to normal surveillance testing in accordance with the ANO-2 Technical Specifications.

10.1.18 Emergency Operating Procedure

Emergency Operating Procedure Technical Guidelines are under development for guidance on use of RVLMS. Upon NRC approval of the guidelines, the station EOP will be updated.

10.2 Subcooling Margin Monitor (SMM)

Two channels of saturation margin measurement and indication are installed and operational in ANO-2. Each channel of SMM consists of one dedicated digital calculator, two wide-range (150 F to 750 F) RTD temperature sensors, two wide-range (0-3000 psig) pressurizer pressure transmitters, one strip chart recorder, and a digital indicator. These components meet seismic and environmental qualifications in compliance with IEEE 323 and IEEE 344.

The ANO-2 plant computer has backup display capability for subcooling margin. In addition, steam tables are provided in the control room for use by the operators to manually determine saturation margin .

10.3 Core Exit Thermocouples (CETs)

NUREG-0737, Item II.F.2 states; The instrumentation must be evaluated for conformance to Appendix B, 'Design and Qualification Criteria for Accident Monitoring Instrumentation,' as modified by the provisions of items 6 through 9.

Significant upgrading has been and is being undertaken to make this a Class 1E system up to and including the backup display device and suitable isolators for the non-Class 1E display. See Section 9.3.1.

10.3.1 Arrangement

Figure 2 shows the locations of the in-core instrument assemblies. Figure 4 diagrams the CET Monitoring System. The core exit thermoccuples are part of the in-core instruments.

10.3.2 Primary Display

The SPDS computer is the primary display. Its color graphic capabilities provide a core map with CET temperatures.

10.3.3 Temperature Range

The CETs have a range of 0 F to 2300 F. The readouts conform to this range. Readings that are out-of-range high or low cause an alarm on the alarm CRT of the computer. This alerts the operator of possible instrument failure. Out-of-range readings are rejected by the computer prior to calculating the average of the five highest readings.

10.3.4 Backup Display

The backup display is a TEC 601 Plasma Display Unit and Keypad that has input signals from at least 16 CETs. Individual thermocouples can be selected manually. Operability of the backup unit is checked by periodic surveillance and calibration. The range of this display is also 0 F to 2300 F.

10.3.5 Use of the Display

Use of the primary display (SPDS) will be based on the ICC tab of the Emergency Operating Procedure (EOP). The backup display will be used when the primary display is unavailable. The primary display provides the core exit temperature information that the operator needs to cope with an ICC event. This information will be called for by the EOP at the time it is required.

The CETs may be used to provide backup indication for reactor outlet RTDs (hot leg temperature).

10.3.5.1 Training

The training program will include procedure training that specifies which indications to monitor and what actions to take based on the indications.

10.3.6 Environmental Qualification

With the exception of the CETs and in-containment cables and connectors, all parts of the CET Monitoring System are environmentally qualified. See Section 2.4

10.3.7 Single Failure Analysis

The system is single failure proof up to and including the final electrical isolation devices in the signal processing equipment. From this point, both channels feed the common SPDS display. The SPDS was designed to approach 99% availability. The CET inputs to the backup display are derived from one of the channels.

10.3.8 Class 1E Power Source

Each channel from the CETs up to and including the final electrical isolator in the signal processing equipment, and the backup display, is powered from its own, independent, Class 1E power source. The SPDS is powered from a highly reliable non-Class 1E inverter that is fed from a safety bus with battery backup.

10.3.9 Availability Prior to an Accident

CET information is available continuously. Because of the redundant channel design, a channel can be out of service for calibration or maintenance for short periods while the other channel remains in service.

10.3.10 Quality Assurance

See Section 9.1.

10.3.11 Continuous Indication

CET temperature information is available from the SPDS computer display which the operator can select and display continuously on demand. The operator must manually select the desired CET location to monitor on the backup display. Because all CET's monitor temperature over the same range, there are no gaps in the measurement.

10.3.12 Recording of Instrument Outputs

Recording of CET temperature information is performed only on the SPDS, using its trending capabilities. Trend information is stored by the computer and is available for display on demand.

10.3.13 Identification of Instruments

The SPDS, as the primary display system, is recognized as being useful for monitoring and assessing accident conditions. The backup display is also available for use as an accident monitoring device.

10.3.14 Isolation

See Section 9.3

11.0 REFERENCES

11.1 Technology for Energy Corporation (TEC)

The following major documents were submitted to AP&L for approval. (TEC submitted additional documents considered by TEC to be necessary for complete documentation.) TEC maintains a permanent, auditable file of all documentation of the Project.

11.2 Nuclear Regulatory Commission (NRC)

11.2.1 NUREG 0737, Clarification of the TMI Action Plan Requirements

11.2.2 Supplement 1 to NUREG 0737, Requirements for Emergency Response Capability

11.2.3 Appendix B to 10CFR 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

11.2.4. 10CFR50.49, Environmental Qualification of Electrical Equipment for Nuclear Power Plants

11.2.5 Regulatory Guide 1.100, Seismic Qualification of Electrical Equipment for Nuclear Power Plants

11.2.6 Regulatory Guide 1.26, Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants

11.2.7 Regulatory Guide 1.63, Electric Penetration Assemblies in Containment Structures for Water Cooled Nuclear Power Plants

11.2.8 Regulatory Guide 1.75, Physical Independence of Electrical Systems

11.2.9 Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants

11.2.10 Regulatory Guide 1.97, Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Env Conditions During and Following an Accident

11.3 American Society of Mechanical Engineers (ASME)

11.3.1 ASME Boiler and Pressure Vessel Code, Section III

11.4 American National Standards Institute (ANSI)

11.4.1 ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants

11.4.2 ANSI N45.2.2, Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Plants

11.4.3 ANSI N45.2.4, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing Requirements of Instrumentation and Electri cal Equipment During the Construction of Nuclear Power Generating Stations

11.4.4 ANSI N45.2.6, Qualification of Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants

11.4.5 ANSI N45.2.8, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants

11.4.6 ANSI N45.2.9, Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants

11.4.7 ANSI N45.2.10, Quality Assurance Terms and Definitions

11.4.8 ANSI N45.2.12, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants

11.4.9 ANSI N45.2.13, Supplementary Quality Assurance Requirements for Preparation of Procurement Documents for Nuclear Power Plants

11.4.10 ANSI N45.2.14, Quality Assurance Program Requirements for the Design and Manufacture of Class 1E Instrumentation and Electric Equipment for Nuclear Power Generating Stations

11.4.11 ANSI N45.2.22, Supplementary Requirements for Inspection of Dimensional Characteristics

11.4.12 ANSI N45.2.23, Qualifications of Quality Assurance Program Audit Personnel for Nuclear Facilities

11.4.13 ANSI N512, Protective Coatings (paints) for the Nuclear Industry

11.5 Institute of Electrical and Electronic Engineers (IEEE)

11.5.1 IEEE 323, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

11.5.2 IEEE 344, Guide for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

11.5.3 IEEE 383, Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations

11.5.4 IEEE 384, Criteria for Independence of Class 1E Equipment and Circuits

11.6 Arkansas Power and Light Company

Letter ME-85-122, April 4, 1985 (to TEC), Response and Documentation, Evaluation of Off-Specification Chemical Content of SFA 5.9 ER308 Welding Wire

12.0 FIGURES

13.0 TABLES

Figure 1 - ANO 2 REACTOR VESSEL
AND ICC DETECTORS

Figure 2 - ANO-2 ICC DETECTOR LOCATIONS

- location of core exit thermocouples
- E8 and N8 identify location of RGT probes.

Figure 3. Environmental Conditions for Qualification of In-Containment Equipment

Figure 4

ANO-2 CORE EXIT THERMOCOUPLE (CET) MONITORING SYSTEM

Figure 5. Electrical System Arrangement

Figure 6

".1 SQ.FT. BREAK LOCA"

TABLE 1

ICC HARDWARE LIST

TABLE 2

PERFORMANCE OF SENSOR TYPES

TABLE 3

PERFORMANCE OF GAS-GAP LENGTHS

TABLE 4

PERCENT EFFICIENCIES OF THE UPPER PLENUM MANOMETER GEOMETRIES
(WITH STANDARD DEVIATIONS)

TABLE 5

PERCENT EFFICIENCIES OF THE DOME MANOMETER GEOMETRIES
(WITH STANDARD DEVIATIONS)

Table 6

Calculation of Measurement Error