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December 16, 1994

U.S. Nuclear Regulatory Commission
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Project Directorate I-3
Division of Reactor Projects - I/II
Washington, D.C. 20555

Subject: Annual Report of Facility Changes, Tests, and Experiments Conducted Without Prior Commission Approval
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Johnson:

The subject report is hereby submitted as required by 10 CFR 50.59(b). The enclosed report contains descriptions and summaries of the safety evaluations conducted in support of changes to the facility and procedures described in the UFSAR and special tests, from August 1993 through July 1994, performed under the provisions of 10 CFR 50.59.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy
Vice President
Nuclear Operations

RCM/tjn
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1994 REPORT

OF

FACILITY CHANGES, TESTS AND EXPERIMENTS

CONDUCTED WITHOUT PRIOR APPROVAL

FOR AUGUST 1993 THROUGH JULY 1994

SECTION A	COMPLETED ENGINEERING WORK REQUESTS (EWR) AND TECHNICAL STAFF REQUESTS (TSR)
SECTION B	COMPLETED STATION MODIFICATIONS (SM)
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SECTION E	PROCEDURE CHANGES
SECTION F	COMPLETED SPECIAL TESTS (ST) AND EXPERIMENTS

R.E. GINNA NUCLEAR POWER PLANT
DOCKET NO. 50-244
ROCHESTER GAS AND ELECTRIC CORPORATION

DATED DECEMBER 16, 1994

SECTION A - COMPLETED ENGINEERING WORK REQUESTS (EWRs) AND TECHNICAL STAFF REQUESTS (TSRs)

This section contains a description of modifications in the facility as described in the safety analysis report, and a summary of the safety evaluation for those changes, pursuant to the requirements of 10 CFR 50.59(b).

The basis for inclusion of an EWR or TSR in this section is closure of the completed modification package in the Document Control Department.

TSR 89-30
SI CHECK VALVE 1828 TEST CONNECTION

THIS TECHNICAL STAFF REQUEST (TSR) ADDRESSES THE INSTALLATION OF A PERMANENT TEST CONNECTION ON THE BLIND FLANGE DOWNSTREAM OF SI VALVE 2816 FOR QUARTERLY FUNCTIONAL TEST OF SI CHECK VALVE 1828.

REVISION 1 OF THE DESIGN CRITERIA ADDED DRAWING C-381-359, SHEET 9, REVISION 0 TO THE REFERENCES AND REVISED THE DESIGN TEMPERATURE AND PRESSURE FROM 300°F AND 210 PSIG TO 500°F AND 150 PSIG. THIS REVISION WAS NEVER PORC APPROVED.

REVISION 2 OF THE DESIGN CRITERIA REVISED/DELETED REVISION LEVEL FOR P&ID AND A-PROCEDURES AND CORRECTED CODE CLASS REFERENCE FROM THE P&ID TO NON ASME CLASS.

A REVIEW HAS BEEN PERFORMED OF THE DESIGN BASIS EVENTS. THE EVENT RELATED TO THIS MODIFICATION IS A LOSS OF COOLANT ACCIDENT (LOCA).

THE MODIFICATION WILL NOT INCREASE A) THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT WILL NOT BE INCREASED BECAUSE THE CONNECTION WILL BE USED DURING TESTING ONLY, THE CONNECTION WILL BE CAPPED AND ISOLATED FROM THE SYSTEM DUE TO THE NORMAL CLOSED POSITION OF VALVE 2816 AND THE CONNECTION WILL BE INSTALLED TO MEET ALL APPLICABLE REQUIREMENTS OF THE SAFETY INJECTION SYSTEM, OR; B) THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS REPORT WILL NOT BE CREATED BECAUSE REDUNDANT METHODS OF ISOLATING THE BRANCH LINE STILL EXIST. ORIGINALLY ISOLATION OF THE BRANCH LINE CONSISTED OF THE ISOLATION VALVE AND A BLIND FLANGE AND THE PROPOSED MODIFICATION CONSISTS OF THE ADDITION OF A CAPPED TEST CONNECTION WELDED TO A BLIND FLANGE THAT WILL FUNCTION IN A MANNER SIMILAR TO THAT OF THE ORIGINAL BLIND FLANGE, OR; C) THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION IS NOT REDUCED BECAUSE THE TEST CONNECTION WILL BE INSTALLED TO BOTH ASME CLASS 2 AND SEISMIC CATEGORY I REQUIREMENTS AND THE TEST CONNECTION HAS BEEN DETERMINED NOT TO SIGNIFICANTLY INCREASE STRESSES WITHIN THE PIPING THAT COULD POTENTIALLY LEAD TO FAILURE BECAUSE OF THE SLIGHT WEIGHT INCREASE. THE WEIGHT INCREASE HAS ALSO BEEN ADDRESSED IN THE SEISMIC ANALYSIS OF THE LINE AND HAS BEEN DETERMINED NOT TO ADVERSELY EFFECT PREVIOUS SEISMIC ANALYSIS OF THE LINE.

BASED ON THE ABOVE ANALYSIS:

1) THE MARGIN OF SAFETY DURING NORMAL OPERATION AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION WILL NOT BE DECREASED BECAUSE THE MODIFICATION DOES NOT ALTER THE INTENDED FUNCTION OF THE SAFETY INJECTION SYSTEM, NOR DOES IT REMOVE THE REDUNDANT ISOLATION OF THE BRANCH LINE AS ORIGINALLY DESIGNED.

2) ALSO, STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF CONSEQUENCES OF ACCIDENTS ARE ADEQUATE BECAUSE THE MODIFICATION HAS BEEN EVALUATED TO NOT SIGNIFICANTLY ALTER PREVIOUS SEISMIC ANALYSIS, AND INSTALLATION OF THE SYSTEM NOR WILL IT ADVERSELY EFFECT ANY SAFETY RELATED EQUIPMENT WITHIN THE AREA OF MODIFICATION.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR 2504</u>
<u>CONTAINMENT PURGING</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION OF THE CONTAINMENT PURGING SYSTEM. THE 48-INCH INBOARD CONTAINMENT ISOLATION VALVES AT PENETRATIONS IN CONTAINMENT WILL BE REMOVED AND REPLACED WITH BLIND FLANGES. A NEW MINI-PURGE SYSTEM WILL BE INSTALLED ADDING A BLOWER TO SUPPLY CLEAN AIR TO THE CONTAINMENT DURING ALL PHASES OF PLANT OPERATION.

REVISION 4 TO THIS DESIGN CRITERIA AND SAFETY ANALYSIS INCORPORATES CHANGES FROM REVISION 3 TO CORRECT THE FOLLOWING:

1) REDUCE THE MINI-PURGE FLOW RATE FROM 2000 CFM TO 1500 CFM.

2) EXISTING 6-INCH DEPRESSURIZATION VALVES WILL BE REPLACED WITH NEW 6-INCH VALVES INSTEAD OF THE ORIGINAL DESIGN FOR 8-INCH VALVES.

3) TO INCORPORATE REVISION LEVEL CHANGES.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR. THE EVENTS RELATED TO THIS MODIFICATION ARE AS FOLLOWS:

1) PRIMARY SYSTEM PIPE RUPTURE
2) PIPE BREAKS OUTSIDE THE CONTAINMENT BUILDING
3) FUEL HANDLING ACCIDENT
4) INTERNAL AND EXTERNAL EVENTS SUCH AS MAJOR AND MINOR FIRES, FLOODS, STORMS, OR EARTHQUAKES.

CONTAINMENT ISOLATION INTEGRITY AT PENETRATIONS 204 AND 300 IS ASSURED BY THE DOUBLE "O" RING SEALED BLIND FLANGES LOCATED INSIDE CONTAINMENT. THE BLIND FLANGES AND "O" RINGS ARE DESIGNED TO WITHSTAND LOCA CONDITIONS IN CONTAINMENT.

CONTAINMENT ISOLATION INTEGRITY OF PENETRATIONS 132 AND 309 IS ASSURED BY INSTALLATION OF RAPIDLY CLOSING AND TIGHTLY SEALED VALVES. THESE VALVES WILL OPERATE DURING LOCA LOADS AND ENVIRONMENTAL CONDITIONS APPLICABLE TO THEIR LOCATION.

REPLACEMENT OF EXISTING OUTBOARD CONTAINMENT ISOLATION VALVES DOES NOT ALTER THEIR LOCATION OR ORIENTATION. THEREFORE, THE CONSEQUENCES OF PIPE BREAKS OUTSIDE THE CONTAINMENT BUILDING ARE NOT INCREASE BY THIS MODIFICATION.

NEW AUTOMATIC INBOARD AND OUTBOARD CONTAINMENT ISOLATION VALVES INSTALLED AT PENETRATIONS 132 AND 309 HAVE SIMILAR DESIGN CHARACTERISTICS AS EXISTING VALVES. THEREFORE, THE CONSEQUENCES OF A FUEL HANDLING ACCIDENT WILL NOT BE INCREASED BY THIS PORTION OF THE MODIFICATION.

THE BLIND FLANGES WILL BE REMOVED AT PENETRATIONS 204 AND 300 ONLY WHEN CONTAINMENT INTEGRITY IS NOT REQUIRED. THE EXISTING AUTOMATIC OUTBOARD ISOLATION VALVES WILL PREVENT RADIOACTIVE RELEASES WHEN REFUELING INTEGRITY IS REQUIRED. THEREFORE, THE CONSEQUENCES OF A FUEL HANDLING ACCIDENT WILL NOT BE INCREASED BY THIS PORTION OF THE MODIFICATION.

THIS MODIFICATION PREVENTS SAFE SHUTDOWN EQUIPMENT FROM BEING DISABLED, AND WIRING AND CABLE WILL MEET IEEE-383-1924 FLAME TEST REQUIREMENTS. THEREFORE, THE CONSEQUENCES OF MAJOR OR MINOR FIRES ARE NOT INCREASED BY THIS MODIFICATION.

THE MODIFICATION NEITHER AFFECTS NOR IS AFFECTED BY ANY FLOOD OR STORM PREVIOUSLY EVALUATED.

NON-SAFETY RELATED PORTIONS OF THIS MODIFICATION ARE SEISMICALLY SUPPORTED TO PREVENT SAFE SHUTDOWN EQUIPMENT FROM BEING DISABLED. PIPE SUPPORTS WITHIN PENETRATION BOUNDARIES SHALL BE DESIGNED TO MONITOR STRUCTURAL INTEGRITY. THE CONSEQUENCES OF AN EARTHQUAKE WILL NOT DEGRADE THE CAPABILITY OF THIS MODIFICATION TO PERFORM ITS INTENDED SAFETY FUNCTION.

BASED UPON THE ABOVE ANALYSIS:

1) STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

2) MARGIN OF SAFETY DURING NORMAL OPERATING AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-3797</u>
<u>MICROPROCESSOR REACTOR CONTROL ROD POSITION INDICATION SYSTEM</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE UPGRADE OF THE ANALOG REACTOR CONTROL ROD POSITION INDICATION (ARPI) SYSTEM AT GINNA STATION. THE ARPI SYSTEM IS AN ORIGINAL PLANT INSTALLATION AND BECAME DIFFICULT TO MAINTAIN. TECH. SPEC. REQUIREMENTS TO BE AT HOT SHUTDOWN WITHIN SIX HOURS WITH TWO INOPERABLE RPIs MADE MAINTENANCE AT POWER VERY DIFFICULT. THE ARPI SYSTEM REQUIRED APPROXIMATELY 24 HOURS OF CRITICAL PATH TIME TO CALIBRATE DURING REFUELING OUTAGES AND CONSTANT I&C COVERAGE DURING PLANT STARTUPS TO PERFORM RPI ALIGNMENTS RESULTING FROM A NON-LINEAR SYSTEM FEEDING A LINEAR INDICATOR.

THE NEW DIGITAL MICROPROCESSOR REACTOR CONTROL ROD (MRPI) SYSTEM HAS SEVERAL ADVANTAGES: DIGITAL DETECTOR DESIGN FOR IMPROVED PERFORMANCE; REDUCED STARTUP MAINTENANCE WITH ELIMINATION OF STARTUP CALIBRATION; AND PHYSICAL AND FUNCTIONAL REDUNDANCY OF BOTH INSIDE AND OUTSIDE CONTAINMENT ELECTRONICS.

REVISION 0 OF THE DESIGN CRITERIA AND SAFETY ANALYSIS WAS PRESENTED AND APPROVED BY PORC ON MAY 5, 1986, PORC NUMBER 6.1.0-86-068-002.

REVISION 1 OF THE DESIGN CRITERIA AND SAFETY ANALYSIS WAS PRESENTED AND APPROVED BY PORC ON AUGUST 8, 1989, PORC NUMBER 6.1.0-89-131-001.

REVISION 1 OF THIS DESIGN CRITERIA ADDRESSED THE INSTALLATION OF FANS TO THE DATA ACQUISITION CABINET INSIDE CONTAINMENT. THE AMBIENT TEMPERATURE OF THE CONTAINMENT BUILDING IS WITHIN THE DESIGN TEMPERATURE LIMITS OF THE DATA ACQUISITION CABINET, THEREFORE, THE FANS ARE CONSIDERED UNNECESSARY AND WILL BE REMOVED UNDER REVISION 2.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE:

1) MAJOR AND MINOR FIRES,
2) A SEISMIC EVENT,
3) A CONTROL ROD DROP EVENT.

NEW WIRING AND CABLE MAY BE REQUIRED FOR THIS MODIFICATION WHICH COULD ADD TO THE FIRE LOADING OF THE PLANT. THEREFORE, THE DESIGN CRITERIA REQUIRES THAT ALL SUCH CABLE MEET THE IEEE-383-1974 FLAME TEST REQUIREMENTS. BECAUSE OF THIS THERE WILL BE NO SIGNIFICANT INCREASE OF FIRE LOADING CAUSED BY THIS MODIFICATION.

THIS MODIFICATION HAS BEEN REVIEWED TO ENSURE THAT FAILURE OF ANY ELECTRICAL CABLE INSTALLED AS A PART OF THIS MODIFICATION WILL NOT RESULT IN THE DISABLING OF VITAL EQUIPMENT NEEDED TO SAFELY SHUT DOWN THE PLANT DURING POSTULATED FIRES.

THE ROD POSITION INDICATION (RPI) SYSTEM IS DESIGNATED NON SEISMIC CATEGORY I, HOWEVER, THE DESIGN CRITERIA REQUIRES THAT THOSE PORTIONS OF THE SYSTEM WHOSE FAILURE COULD CAUSE DAMAGE TO SAFETY RELATED EQUIPMENT WILL BE DESIGNED TO MEET THE REQUIREMENTS OF REGULATORY GUIDE 1.29, SECTION C.2.

DURING A MAJOR SEISMIC EVENT THE REACTOR WOULD BE TRIPPED. A NON-SEISMIC RPI SYSTEM MAY NOT PROVIDE RELIABLE POSITION INDICATION OR ROD BOTTOM INDICATION DURING OR AFTER THE SEISMIC EVENT. THIS IS ACCEPTABLE BECAUSE THE MAJOR CONCERN IS THAT REACTOR POWER BE DECREASING AND REMAIN SUBCRITICAL. THIS FUNCTION IS FULFILLED BY THE SEISMIC NUCLEAR INSTRUMENTATION SYSTEM (NIS). THE NIS WOULD MONITOR THE POWER DECREASE, VERIFYING REACTOR HAS TRIPPED AND THE SOURCE RANGE WOULD PROVIDE INDICATION THAT THE CORE IS REMAINING SUBCRITICAL.

ADDITIONAL ACCEPTABILITY OF A NON-SEISMIC RPI SYSTEM IS OBTAINED FROM REG. GUIDE 1.97. BASED ON THIS GUIDE THE RPI SYSTEM IS CATEGORY 3 WHICH DOES NOT REQUIRE ANY QUALIFICATION.

THE MRPI DATA CABINET WAS SEISMICALLY ANCHORED BEFORE INSTALLATION OF THE COOLING FANS, THEREFORE, REMOVING THE COOLING FANS WILL NOT DEGRADE THE SEISMIC ANCHORAGE OF THE DATA CABINET.

SECTION 10.6 OF THE DESIGN CRITERIA REQUIRES THAT THE NEW MRPI SYSTEM BE DESIGNED TO INTERFACE WITH THE EXISTING TURBINE RUNBACK LOGIC, THEREFORE, A CONTROL ROD DROP EVENT WILL NOT AFFECT THE TURBINE RUNBACK RESPONSE.

REMOVAL OF THE EXISTING COOLING FANS DOES NOT CHANGE THE CONCLUSIONS OF THE SAFETY ANALYSIS.

THEREFORE, IT HAS BEEN DETERMINED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND FOR THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

EWR-3990
DIESEL GENERATOR BUILDING MODIFICATION

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE STRUCTURAL UPGRADE OF THE DIESEL GENERATOR BUILDING TO WITHSTAND TORNADO WIND LOADING, TORNADO MISSILES, FLOODING, AND TO ENSURE THAT THE BUILDING REMAINS 1) UNDAMAGED DURING AND AFTER AN OPERATING BASIS EARTHQUAKE (OBE) AND 2) FUNCTIONAL AFTER A SAFE SHUTDOWN EARTHQUAKE (SSE) TO ENSURE SAFETY FUNCTIONS FOR THE BUILDING AND ITS CONTENTS ARE MET.

THIS LATEST REVISION LEVEL OF THE DESIGN CRITERIA AND SAFETY ANALYSIS ADDS AN ADDITIONAL FIREWATER SYSTEM LINE EXTENSION TO PROVIDE PROTECTION VIA A SPRINKLER SYSTEM FOR THE AUXILIARY OPERATIONS OFFICE TO BE LOCATED ON THE OPERATING FLOOR OF THE TURBINE BUILDING. THUS, THE PREVIOUS PORC APPROVAL OF THIS EWR AT MEETING 90-120 ON 8/29/90 IS APPENDED BY THIS REVIEW.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND THE EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE WIND, SNOW AND TORNADO LOADINGS, FLOODING, SEISMIC, FIRES AND LOSS OF AC POWER.

ALL THE FIRE PROTECTION FEATURES REQUIRED TO MEET THE CONDITIONS OF THE GINNA LICENSE, TO ASSURE COMPLIANCE WITH 10 CFR 50 APPENDIX R, OR TO MAINTAIN EQUIVALENT LEVELS OF PROTECTION FROM FIRES WILL BE MAINTAINED DURING AND FOLLOWING THE STRUCTURAL UPGRADE MODIFICATIONS. CHANGES TO THE FIRE PROTECTION FEATURES OF THE PLANT ARE DESCRIBED IN SECTION 1.1.1 OF THE DESIGN CRITERIA. THE CHANGES AND ADDITIONS TO THE FIRE PROTECTION FEATURES WILL EXTEND, ENHANCE OR MAINTAIN THEIR FIRE PROTECTION CAPABILITY. THE DOCUMENTATION OF THIS JUSTIFICATION WILL BE PROVIDED IN THE APPENDIX R CONFORMANCE VERIFICATION. THEREFORE, THE FIRE PROTECTION FEATURES WILL NOT BE DEGRADED BY THE MODIFICATIONS.

THE POTENTIAL FOR FLOODING THE CONTROL ROOM AS A RESULT OF A FIRE WATER SPRINKLER LINE BREAK OR A FIRE WATER SYSTEM ACTUATION HAS BEEN ADDRESSED PREVIOUSLY WITH THE INSTALLATION OF THE CONTROL ROOM PRESSURE WALL FIRE WATER SPRAY CURTAIN. THE INSTALLATION OF FIRE WATER CAPABILITIES FOR THE NEW OPERATIONS OFFICE AREA WOULD NOT IMPOSE ADDITIONAL FLOODING POTENTIAL AS THE OPERATION OF THIS OR ANY OTHER EXISTING FIRE WATER SUPPRESSION SYSTEM WOULD RESULT IN THE ANNUNCIATION OF A FIRE WATER PUMP START IN THE CONTROL ROOM. THIS ANNUNCIATION WOULD ALERT THE APPROPRIATE PERSONNEL TO THE CONDITION SO THAT ACTIONS WOULD BE TAKEN TO DISCONTINUE FIRE WATER FLOW AS REQUIRED.

THE POTENTIAL FOR THE INSTALLATION OF FIRE WATER SYSTEM PIPING FOR THE OPERATIONS OFFICE AREA TO RESULT IN AN INCREASE OR DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM IS CONSIDERED TO BE NEGLIGIBLE. THE FIRE WATER PIPING TO BE INSTALLED IS NOT IN THE IMMEDIATE AREA OF SECONDARY SYSTEM COMPONENTS INCLUDING THE TURBINE STOP VALVES WHICH COULD BE ADVERSELY AFFECTED BY THE FIRE WATER SYSTEM. ALSO ACTUATION OF ANY FIRE WATER SYSTEM WOULD BE ANNUNCIATED TO THE CONTROL ROOM VIA THE OPERATION OF EITHER THE ELECTRIC OR DIESEL FIRE PUMPS WHICH WOULD ENSURE APPROPRIATE OPERATOR ACTION. APPROPRIATE OPERATOR ACTION WOULD ENSURE THAT THE SYSTEM IS ISOLATED UPON EXTINGUISHMENT OF A POTENTIAL FIRE SO THAT THE IMPACT OF ANY FIRE WATER SYSTEM ACTIVATION ON OTHER PLANT SYSTEMS WOULD BE MINIMIZED.

THE PROPOSED MODIFICATION INCLUDING THE CAPABILITY TO PROVIDE FIRE WATER TO THE OPERATIONS OFFICE WOULD NOT INCREASE THE PROBABILITY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE ACCIDENTS ANALYZED IN CHAPTER 15 OF VOLUME VIII OF THE UFSAR ARE ONLY REMOTELY RELATED TO THE SYSTEMS EFFECTED BY THIS MODIFICATION (POSSIBLY SECOND OR THIRD LEVEL BACK UP SYSTEMS). THERE IS NO WAY THE PROPOSED CHANGES WILL HAVE ANY AFFECT ON THE PROBABILITY OR FREQUENCY OF THE OCCURRENCE OF SUCH AN ACCIDENT. THE SAFETY RELATED EQUIPMENT THAT THE UPGRADED DIESEL GENERATOR BUILDING IS DESIGNED TO PROTECT IS EQUIPMENT USED IN THE MITIGATION OF ACCIDENTS AND IS NORMALLY NOT OPERATING. THIS EQUIPMENT, WHEN OPERATING DOES NOT CREATE THE POTENTIAL TO CAUSE THE ACCIDENTS DESCRIBED IN THE UFSAR.

ALSO, THE PROPOSED FIRE WATER SYSTEM MODIFICATIONS DOES NOT EFFECT ANY ACCIDENT MITIGATION SYSTEMS. THUS, THE UFSAR ACCIDENT CONSEQUENCES WILL NOT BE CHANGED BY THIS MODIFICATION.

EXTENDING THE FIRE WATER PIPING UNDER THIS REVISION 5 OF THE SAFETY ANALYSIS DOES NOT IMPACT SAFETY RELATED EQUIPMENT AS THIS PORTION OF THE MODIFICATION IS LOCATED IN THE TURBINE BUILDING. ALSO THE NEW PIPING IS REQUIRED TO BE INSTALLED IN ACCORDANCE WITH NFPA CODE FOR DESIGN, MATERIAL AND CONSTRUCTION REQUIREMENTS.

THUS, THIS MODIFICATION NEITHER INCREASES THE CONSEQUENCES, NOR DOES IT REDUCE THE MARGINS OF SAFETY FOR:

1) EQUIPMENT REQUIRED TO FUNCTION DURING AND FOLLOWING OBE, SEE, FLOODING, WIND, AND SNOW LOADS, INCLUDING TORNADO EVENTS.

2) FIRE PROTECTION FEATURES.

3) EQUIPMENT REQUIRED TO EFFECT A SAFE SHUTDOWN OF THE PLANT, FOLLOWING A LOSS OF OFF-SITE AC POWER.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

EWR-4492
CONTAINMENT EQUIPMENT AND PERSONNEL HATCH MODIFICATION

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE REBUILD OF THE EXISTING EQUIPMENT AND PERSONNEL AIR LOCK MECHANICAL DRIVE SYSTEMS. THE "REBUILDS" WILL USE THE SAME STYLE AND QUALITY OF PARTS USED IN THE ORIGINAL CONSTRUCTION. THE MATERIALS ARE AN "UPGRADE". AN ADDITION WILL PROVIDE ADDITIONAL SUPPORT TO THE MAIN HINGE PIN.

REVISION 1 OF THE DESIGN CRITERIA AND SAFETY ANALYSIS WERE PRESENTED AND APPROVED BY THE PORC ON 1/17/90, MEETING #90-006.

THE PURPOSE OF REVISION OF THE DESIGN CRITERIA IS TO 1) CLASSIFY THE ADDITIONAL MAIN HINGE BEARING SUPPORT AS NON-SEISMIC AND NOT SUBJECT TO NUCLEAR QUALITY ASSURANCE REQUIREMENTS.

THE PURPOSE OF REVISION OF THE SAFETY ANALYSIS IS TO 1) CHANGE REVISION OF DESIGN CRITERIA IN REFERENCE SECTION AND 2) PROVIDE SAFETY ANALYSIS FOR THE NON-SEISMIC MAIN HINGE PIN MODIFICATION.

A REVIEW HAS BEEN MADE OF ALL EVENTS IN THE GINNA STATION UFSAR AND THE EVENTS REQUIRING ANALYSIS BY THE U.S. NRC REGULATORY GUIDE 1.70. THE EVENTS RELATED TO THESE UPGRADES AND MODIFICATIONS ARE SEISMIC, FIRES AND CONTAINMENT INTEGRITY.

THE MODIFICATIONS TO THE AIR LOCK SYSTEMS WILL BE OF THE SAME DESIGN AS THE ORIGINAL, HOWEVER, THE COMPONENT MATERIALS WILL BE "UPGRADED". THE UPGRADES, THEREFORE, WILL NOT DEGRADE THE CAPABILITY OF THE SYSTEM TO PERFORM AS DESIGNED DURING OR AFTER A SEISMIC EVENT.

THE MODIFICATIONS WILL CONSIST OF NON-FLAMMABLE MECHANICAL COMPONENTS AND, THEREFORE, WILL NOT INCREASE THE FIRE LOADING IN THE PLANT. THE MODIFICATION WILL NOT DEGRADE ANY EXISTING FIRE BARRIERS.

COMPONENT PARTS PENETRATING THE CONTAINMENT PRESSURE BOUNDARY SHALL BE REPLACED WITH EQUAL OR UPGRADED MATERIAL COMPONENTS. THESE COMPONENTS WILL BE THE SAME DESIGN AS THE ORIGINAL MECHANISMS AND, THEREFORE, NOT DEGRADE THE CONTAINMENT PRESSURE BOUNDARY.

THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY, PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT WILL NOT BE INCREASED BY THE PROPOSED ADDITION.

THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS REPORT WILL NOT BE CREATED BY THE PROPOSED ADDITION.

EWR-4526
DIESEL FUEL OIL SYSTEM

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION TO THE EMERGENCY DIESEL GENERATORS FUEL OIL SYSTEM.

THE DIESEL FUEL OIL SYSTEM MODIFICATIONS WILL INCLUDE; THE INSTALLATION OF A NEW FILTERING DESIGN FOR THE TRANSFER PUMPS, REPLACEMENT OF THE TRANSFER PUMPS, INSTALLATION OF IMPROVED SYSTEM INSTRUMENTATION AND CONTROL, MODIFICATION OF THE PIPING ASSOCIATED WITH THE FILL AND BYPASS SOLENOID VALVES TO ALLOW REMOVAL OF THE EXISTING VALVES FOR MAINTENANCE, AND THE ADDITION OF ISOLATION VALVES TO THE RECIRCULATION PIPING. THE MODIFICATIONS WILL CONSIDER THE PIPING MATERIALS TO BE USED FOR THE TRANSFER SYSTEM, IN TERMS OF ITS APPROPRIATENESS FOR THE INTENDED SERVICE. DEFINITIVE FUEL OIL REQUIREMENTS WILL BE ESTABLISHED. THE EXISTING CONTROL CONFIGURATION ASSOCIATED WITH THE TRANSFER PUMPS WILL BE ANALYZED FOR THE POTENTIAL OF A FIRE IN ONE DIESEL ROOM RENDERING BOTH TRANSFER PUMPS INOPERABLE. MODIFICATIONS TO CORRECT ANY DEFICIENCIES RELATED TO THIS CONFIGURATION WILL BE MADE AS REQUIRED.

A REVIEW HAS BEEN MADE OF THE DESIGN BASIS EVENTS TO DETERMINE THOSE RELATED TO THE PROPOSED MODIFICATION. THE EVENTS ASSOCIATED WITH THIS WORK ARE:

A) FIRES

B) SEISMIC EVENTS

C) LOSS OF OFFSITE POWER

FIRE BARRIERS WILL NOT BE DEGRADED AND MATERIAL USED WILL MEET CRITERIA EQUAL TO OR GREATER THAN THOSE PRESENTLY INSTALLED. THE POTENTIAL MAY EXIST FOR A FIRE IN ONE EDG ROOM RENDERING BOTH TRANSFER PUMPS INOPERABLE DUE TO AN EXISTING CONTROL CONFIGURATION FOR THE PUMPS. AN ANALYSIS OF THIS CONDITION IS INCLUDED AS PART OF THE MODIFICATION WORK SCOPE. MODIFICATIONS TO CORRECT ANY DEFICIENCIES RELATED TO THIS CONFIGURATION WILL BE MADE AS REQUIRED AND WILL BE REVIEWED AS PART OF THE APPENDIX R CONFORMANCE VERIFICATION. THEREFORE, THE MODIFICATIONS WILL NOT SIGNIFICANTLY ALTER THE AREA FIRE LOADING, THE SOURCES OF FIRE INITIATION, NOR THE ACCEPTABILITY OF THE CONSEQUENCES OF A FIRE.
THE PIPING AND PIE SUPPORT MODIFICATIONS WILL BE EVALUATED, IN

REGARD TO A SEISMIC EVENT, TO CRITERIA IDENTICAL TO THE SEISMIC UPGRADE PROGRAM. THIS WILL ENSURE THAT MODIFICATIONS WILL BE DESIGNED SO AS TO EQUAL OR IMPROVE THE SYSTEM'S CAPABILITY TO WITHSTAND A SEISMIC EVENT.

POWER FOR VITAL ELECTRICAL LOADS IS SUPPLIED BY THE EMERGENCY DIESEL GENERATORS IN THE EVENT OF A LOSS OF OFFSITE POWER.

THE PURPOSE OF THE MODIFICATION IS TO INCREASE THE RELIABILITY OF THE FUEL OIL SYSTEM. THEREFORE, THE MODIFICATION WILL NOT AFFECT THE PREVIOUS LOSS OF OFFSITE POWER ANALYSIS.

BASED UPON THE ABOVE ANALYSIS:

1) STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

2) MARGIN OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION

<u>EWR-4613</u>
<u>BORIC ACID ADDITION SYSTEM - 1/2" TIE-INS</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION TO THE BORIC ACID ADDITION SYSTEM.

SAFETY ANALYSIS STEP 3.2 SECOND STATEMENT IS CORRECT ALONE, BUT THE WORD "INCREASED" IN THE STATEMENT USED IN THIS STEP SHOULD BE "DECREASED". ENGINEERING WILL SUBMIT A CORRECTION TO THE SAFETY ANALYSIS. BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-4617</u>
<u>HANDRAIL UPGRADE CONTAINMENT STRUCTURE</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE UPGRADING OF VARIOUS HANDRAILS INSIDE CONTAINMENT. THE PRESENT HANDRAIL ANCHORAGES ARE EITHER INSUFFICIENT OR NON-FUNCTIONAL DUE TO BEING LOOSE OR STRIPPED. THIS MODIFICATION WILL EVALUATE AND UPGRADE THE HANDRAILS LOCATED WITHIN THE CONTAINMENT STRUCTURE AS RECOMMENDED BY THE SAFETY COMMITTEE.

PRE-PORC REVIEW OF REVISION 0 RESULTED IN THE FOLLOWING COMMENTS:

1) PARAGRAPH 2.5 OF THE SAFETY ANALYSIS SHOULD BE CORRECTED TO CALL OUT UFSAR SECTION 3.8.3 VS. SECTION 3.8.4.

2) USE OF PLURAL/SINGULAR OF HANDRAIL(S) IS CONFUSING.

REVISION 1 OF THIS DESIGN CRITERIA ADDRESSES THE PROPER UFSAR SECTION FOR CONTAINMENT INTERNAL STRUCTURES AND CLARIFIES THE USE OF THE WORD HANDRAILS IN THE TEXT.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR, AND THE EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE OPERATING BASIS AND SAFE SHUTDOWN EARTHQUAKES AND FIRE.

THE HANDRAILS WILL BE DESIGNED TO WITHSTAND OBE AND SSE LOADS. SMALL HANDRAILS AT THE CONCRETE STAIRS ON THE OPERATING FLOOR AND THE HANDRAILS AT THE TRANSFER TUBE WILL NOT BE DESIGNED TO WITHSTAND OBE AND SSE LOADS, BUT WILL BE ANALYZED FOR THE POSSIBILITY OF THE HANDRAILS BECOMING MISSILES, SHOULD THEY BECOME DISLODGED.

THIS MODIFICATION WILL BE REVIEWED PER ENGINEERING PROCEDURE QE-326 TO ENSURE COMPLIANCE WITH THE APPLICABLE PROVISIONS OF 10CFR50 APPENDIX R AND THE FACILITY OPERATING LICENSE.

THUS, THIS MODIFICATION NEITHER INCREASES THE CONSEQUENCES, NOR DOES IT REDUCE THE MARGINS OF SAFETY FOR, 1) EQUIPMENT REQUIRED TO FUNCTION DURING AND FOLLOWING A SEISMIC EVENT, 2) POSE A THREAT TO THE REACTOR COOLANT SYSTEM BOUNDARY, OR 3) AFFECT THE LEVELS OF PROTECTION FROM FIRES DURING AND FOLLOWING MODIFICATIONS TO THE CONTAINMENT HANDRAILS.

BASED UPON A REVIEW OF THE UFSAR AND TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-4643</u>
<u>STATION 13A RELAY COMMUNICATIONS</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION TO THE STATION 13A PROTECTIVE RELAY COMMUNICATIONS SYSTEM WHICH PROVIDES PROTECTION FOR CIRCUITS 908, 911, 912, AND 913. THIS COMMUNICATIONS SYSTEM IS NECESSARY FOR THE HIGH SPEED TRIPPING CAPABILITY REQUIRED TO MAINTAIN STABILITY OF THE GINNA TURBINE GENERATOR.

THE PRESENT COMMUNICATIONS SYSTEM IS LEASED FROM TELEPHONE COMPANIES AND IS USED FOR PRIMARY RELAY COMMUNICATIONS. AFTER 1/1/89, WHICH IS THE DEADLINE SET FOR THE AT&T DIVESTITURE, THE EXISTING TERMINAL EQUIPMENT WILL NO LONGER BE SUPPORTED AND LEASED LINE CHARGES WILL INCREASE DRAMATICALLY.

THE RECOMMENDED METHOD OF PROVIDING RELAY COMMUNICATIONS IS TO EMPLOY THE FIBER OPTIC SYSTEM WHICH IS BEING INSTALLED IN 1987 AND 1988. THIS SYSTEM WILL OFFER THE FLEXIBILITY TO ATTAIN THE REQUIRED SEPARATE ROUTING OF PRIMARY AND SECONDARY COMMUNICATIONS WITH THE INHERENT RELIABILITY OF FIBER OPTICS.

SECONDARY RELAY COMMUNICATIONS, WHICH DO NOT NOW EXIST ON THE GINNA INTERCHANGE SYSTEM, WILL BE INSTALLED THROUGH THE USE OF TERMINAL NODE CONTROLLERS AT STATIONS 13A, 204, 124, 42, 121, AND 122. THIS WORK WILL BE PERFORMED PRIOR TO WORK ON THE PRIMARY RELAY COMMUNICATIONS.

PRIMARY RELAY COMMUNICATIONS, WHICH PRESENTLY CONSISTS OF AUDIO TONES OVER LEASED TELEPHONE LINES, WILL BE CONNECTED TO FIBER OPTIC MULTIPLEXERS BEING INSTALLED AS PART OF THE 1987 AND 1988 FIBER OPTIC SYSTEM. THE TELEPHONE COMPANY OWNED TONE EQUIPMENT AND LEASED LINES WILL THEN NO LONGER BE REQUIRED.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND THE EVENTS REQUIRING ANALYSIS BY USNRC REGULATORY GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE LISTED AS FOLLOWS:

1) LOSS OF EXTERNAL LOAD
2) LOSS OF OFFSITE POWER
3) SEISMIC EVENTS
4) FIRE

THE FIRST EVENT CONSIDERED IS THE LOSS OF EXTERNAL LOAD.

LOSS OF EXTERNAL LOAD HAS BEEN ANALYZED IN THE GINNA UFSAR. THIS MODIFICATION DOES NOT AFFECT THE RESULTS OF THAT ANALYSIS.

DURING CONSTRUCTION, THERE IS THE POSSIBILITY OF INADVERTENT DE-ENERGIZATION OF ONE OF THE 115 KV LINES WHEN THE PLANT IS AT FULL POWER. THIS EVENT HAS BEEN ANALYZED IN THE GINNA UFSAR. PROCEDURES WILL BE INSTITUTED TO REDUCE THE POSSIBILITY OF AN INADVERTENT LOSS OF A 115 KV CIRCUIT.

AFTER COMPLETION, THIS MODIFICATION WILL NOT INCREASE THE PROBABILITY OF A LOSS OF EXTERNAL ELECTRICAL LOADS.

THUS, THIS MODIFICATION NEITHER INCREASES THE CONSEQUENCES, NOR DOES IT REDUCE THE MARGINS OF SAFETY FOR A LOSS OF EXTERNAL LOAD.

THE SECOND EVENT CONSIDERED IS LOSS OF OFFSITE POWER.

WORK ON OFFSITE POWER SOURCES, CIRCUIT 767 FROM NO. 6 TRANSFORMER AND CIRCUIT 751, THE 34 KV CIRCUIT FROM STATION 204, WILL NOT BE REQUIRED AS PART OF THIS MODIFICATION. IN ADDITION, WORK WILL BE SCHEDULED AROUND OUTAGES OF THE DIESEL GENERATORS AT GINNA.

THUS THIS MODIFICATION NEITHER INCREASES THE CONSEQUENCES, NOR DOES IT REDUCE THE MARGINS OF SAFETY FOR A LOSS OF OFFSITE POWER.

THE THIRD EVENT CONSIDERED IS SEISMIC EVENTS.

STATION 13A IS NOT NEEDED FOR SAFE SHUTDOWN DURING OR AFTER AN EARTHQUAKE.

THEREFORE, THIS MODIFICATION IS NON-SEISMIC.

THE FOURTH EVENT CONSIDERED IS FIRE.

THIS MODIFICATION DOES NOT INVOLVE WIRING OR FIRE BARRIERS AT GINNA.

THEREFORE, THIS MODIFICATION HAS NO EFFECT ON FIRES AT GINNA.

IT HAS BEEN DETERMINED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN DETERMINED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

EWR-4671
LOOP LEVEL UPGRADE

THIS ENGINEERING WORK REQUEST (EWR) WHICH WAS REVISED TO DELETE THE AUTOMATIC CORRECTION REQUIREMENT OF MCB INDICATOR LI-432A FOR BIASES DUE TO VARIANCES IN RHR FLOW. THIS ENHANCEMENT WILL BE CONSIDERED FOR INSTALLATION UNDER A FUTURE EWR.

THE CHANGE TO THIS MODIFICATION DID NOT INCREASE THE PROBABILITY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR. IT WILL PROVIDE A REDUNDANT LOCAL SITEGLASS INDICATION OF LOOP WATER LEVEL. THE MODIFICATION IS BEING INITIATED TO ENHANCE THE LEVEL INDICATION DURING REDUCED INVENTORY OPERATION BY ADDING REDUNDANCY AND RELIABILITY (GL-88-17). ACCIDENTS DURING THE REDUCED INVENTORY MODE OF OPERATION ARE NOT REQUIRED AS PART OF UFSAR ANALYSIS. THE 3/8 INCH DIAMETER LINE BREAK HAS BEEN PREVIOUSLY EVALUATED IN SECTION 6.3.3 OF REF. 2.2 FOR THE ASME CODE RECLASSIFICATION DESCRIBED IN THE DESIGN CRITERIA. CONSEQUENTLY, THIS MODIFICATION DOES NOT AFFECT ANY ACCIDENTS EVALUATED IN THE UFSAR.

THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR WILL NOT BE INCREASED BECAUSE THE PROPOSED MODIFICATION DOES NOT INVOLVE ANY CHANGES WHICH COULD RESULT IN ANY INCREASED RADIOLOGICAL DOSES TO THE PUBLIC FOR ACCIDENTS EVALUATED IN THE UFSAR. THE MODIFICATION INVOLVES INDICATION OF THE LOOP LEVEL PARAMETER FOR CONTROL ROOM OPERATORS DURING REDUCED INVENTORY OPERATION ONLY.

THE PROBABILITY OF OCCURRENCE OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE UFSAR WILL NOT BE INCREASED BECAUSE THE TUBING, TRANSMITTERS, SIGNAL PROCESSING, AND INDICATORS WILL BE DESIGNED AND INSTALLED TO MEET THE EQUIVALENT (OR BETTER) REQUIREMENTS TO WHICH THE "B" LOOP LEVEL INSTRUMENT LOOP WAS DESIGNED.

THE CONSEQUENCES OF MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN UFSAR ARE NOT INCREASED BY THIS MODIFICATION BECAUSE THE PROPOSED INSTRUMENTATION TUBING AND VALVES PERFORM A SIMILAR FUNCTION AS THOSE ON THE EXISTING "B" LOOP LEVEL INSTRUMENT LOOP.

THE POSSIBILITY OF AN ACCIDENT OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE UFSAR IS NOT INCREASED SINCE THE DESIGN AND FUNCTION OF THE PROPOSED CHANGES ARE MADE TO ENHANCE THE LEVEL INDICATION THAT PRESENTLY EXISTS BY ADDING A REDUNDANT SYSTEM. THE ADDITION OF THE "A" RCS LOOP LEVEL INDICATION IS DESIGNED TO REDUCE THE POSSIBILITY OF A LOSS OF RHR COOLING DURING REDUCED INVENTORY OPERATION BY PROVIDING MORE ACCURATE LEVEL INDICATION, A REDUNDANT INSTRUMENT LOOP TO REDUCE THE POTENTIAL FOR LEVEL ERRORS, AND INDICATION ON THE PPCS FOR ENHANCED OPERATOR TRENDING AND AWARENESS.

THE POSSIBILITY OF A DIFFERENT TYPE OF MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY THAN ANY PREVIOUSLY EVALUATED IN THE UFSAR IS NOT INCREASED BECAUSE THE TUBING AND INSTRUMENT LOOP DESIGN AND OPERATION IS FUNCTIONALLY EQUIVALENT TO (OR BETTER) THE EXISTING "B" LOOP LEVEL.

THE PROPOSED MODIFICATION DOES NOT REDUCE ANY MARGIN OF SAFETY AS DEFINED IN THE BASIS OF ANY TECHNICAL SPECIFICATION. AS STATED IN ATTACHMENT 5 OF REFERENCE 2.6, THERE ARE NO TECHNICAL SPECIFICATIONS THAT COVER OPERATION WHILE IN THE REDUCED INVENTORY CONDITION. THE LICENSING SUBMITTAL FOR GL-88-17, REFERENCE 2.6, DEFINES THE LICENSING BASIS FOR LEVEL INDICATION WHILE IN THE REDUCED INVENTORY CONDITION. REFERENCE 2.7 COVERS THE THERMAL-HYDRAULIC ANALYSIS, WHICH RG&E ENDORSED, FOR LOSS OF RHR COOLING WHILE THE RCS IS PARTIALLY FILLED. PLANT SPECIFIC ANALYSIS, REFERENCE 2.8, WAS ALSO PREPARED AND IS APPLICABLE. THIS PROPOSED MODIFICATION INVOLVES DESIGN AND INSTALLATION OF A LEVEL INDICATION SYSTEM AND IMPROVED ACCURACY FOR THE EXISTING "B" LOOP SYSTEM. THESE CHANGES DO NOT AFFECT THESE REFERENCED ANALYSES SINCE LEVEL INDICATION IS NOT A VARIABLE PARAMETER IN THE ANALYSIS ASSUMPTIONS. CONSEQUENTLY, THE MARGIN OF SAFETY ESTABLISHED BY THESE ANALYSES IS NOT REDUCED.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-4675</u>
<u>RHR RECIRCULATION</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION TO THE RHR PUMP RECIRCULATION DESIGN CRITERIA AND SAFETY ANALYSIS.

THE MODIFICATION INVOLVES THE UPGRADING OF THE RHR PUMP RECIRCULATION SYSTEM. THE PURPOSE OF THE UPGRADE IS TO PROVIDE GREATER ASSURANCE THAT THE MINIMUM FLOW REQUIREMENTS, IN TERMS OF PUMP PROTECTION, OF EACH OF THE TWO RHR PUMPS CAN BE ACHIEVED WHENEVER ONE OR BOTH PUMPS ARE IN OPERATION. MODIFICATIONS WILL ALSO INCLUDE THE EVALUATION OF THE EXISTING FLOW INDICATOR UTILIZED FOR RHR SYSTEM TESTING, INCLUDING POSSIBLE REPLACEMENT AND/OR RELOCATION OF THE INDICATOR.

A REVIEW HAS BEEN MADE OF THE DESIGN BASIS EVENTS TO DETERMINE THOSE RELATED TO THE PROPOSED MODIFICATION. THE EVENTS ASSOCIATED WITH THIS WORK ARE:

A) FIRES
B) SEISMIC EVENTS
C) DECREASE IN REACTOR COOLANT INVENTORY

THE FOLLOWING ASSESSMENT IS MADE:

THE PROBABILITY AND CONSEQUENCES OF A FIRE HAVE BEEN ADDRESSED IN SECTION 27.0 OF THE MODIFICATION DESIGN CRITERIA. AS DESCRIBED IN THE CRITERIA; FIRE BARRIERS WILL NOT BE DEGRADED, NONCOMBUSTIBLE AND HEAT RESISTANT MATERIALS WILL BE USED WHENEVER PRACTICAL, AND ELECTRIC CABLES AND SPLICES UTILIZED WILL MEET IEEE FLAME TEST REQUIREMENTS. IN ADDITION, THE MODIFICATIONS WILL BE REVIEWED AGAINST THE ASSUMPTIONS OF 10CFR50 APPENDIX R. DEVIATIONS WILL BE ANALYZED TO ASSURE CONTINUED COMPLIANCE WITH APPENDIX R. THEREFORE, THE MODIFICATIONS WILL NOT SIGNIFICANTLY ALTER THE AREA FIRE LOADING, THE SOURCES OF FIRE INITIATION, NOR THE ACCEPTABILITY OF THE CONSEQUENCES OF A FIRE.

NEW AND EXISTING SAFETY RELATED PIPING AND PIPE SUPPORTS INVOLVED IN THE MODIFICATION WILL BE EVALUATED, IN REGARD TO A SEISMIC EVENT, TO CRITERIA CONSISTENT WITH THE SEISMIC UPGRADE PROGRAM. THIS WILL ENSURE THAT MODIFICATIONS WILL BE DESIGNED SUCH THAT THE SYSTEM'S CAPABILITY TO WITHSTAND A SEISMIC EVENT IS NOT DEGRADED.

THE RHR SYSTEM IS PART OF THE ECCS WHICH IS REQUIRED TO MITIGATE THE EFFECTS OF AN INCIDENT WHICH RESULTS IN A DECREASE IN REACTOR COOLANT INVENTORY. THE INTENT OF THE MODIFICATION IS TO INCREASE THE RELIABILITY OF THE SYSTEM THROUGH PROVIDING GREATER ASSURANCE THAT THE RHR PUMPS ARE PROTECTED FROM POSSIBLE DAMAGE. ADDITIONALLY, SECTIONS 1.3.2 AND 15.1 OF THE DESIGN CRITERIA REQUIRES THAT THE MODIFICATIONS NOT HAVE AN ADVERSE AFFECT ON THE PERFORMANCE CHARACTERISTICS OF THE RHR SYSTEM FOR ALL REQUIRED MODES OF OPERATION, SUCH THAT THE DELIVERY CURVE SHOWN IN FIGURE 1 CAN STILL BE ACHIEVED. AN ANALYSIS, PER SECTION 15.2, WILL BE PERFORMED TO SUPPORT THIS CONCLUSION. SECTION 23.4 REQUIRES THAT THE PERFORMANCE OF THE RHR SYSTEM BE VERIFIED FOLLOWING THE MODIFICATIONS. THEREFORE, THE MODIFICATIONS WILL NOT AFFECT ANY PREVIOUS ANALYSIS FOR A DECREASE IN REACTOR COOLANT INVENTORY.

BASED ON THE ABOVE ANALYSIS, IT HAS BEEN DETERMINED THAT:

A) THE MARGINS OF SAFETY DURING NORMAL OPERATION AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

B) THE STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

PRELIMINARY SAFETY EVALUATION

THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY, PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT ARE NOT INCREASED.

THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A TYPE DIFFERENT FROM ANY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT HAS NOT BEEN CREATED.

THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION IS NOT REDUCED.

THE PROPOSED MODIFICATION DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION, BUT IT DOES REQUIRE A TECHNICAL SPECIFICATION CHANGE. THE ACCEPTABLE LEVEL OF PERFORMANCE FOR THE RHR PUMPS LISTED IN SECTION 4.5.2.1 WILL BE REVISED TO REFLECT A HIGHER RECIRCULATION FLOW RATE.

EWR-4770
BORIC ACID STORAGE TANK LEVEL INSTRUMENTATION

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION OF BORIC ACID STORAGE TANK LEVEL INSTRUMENTATION.

THIS MODIFICATION WILL HEAT TRACE EACH OF THE FOUR BUBBLER TUBE SENSING LINES FROM THE LEVEL TRANSMITTERS TO THE BAST'S. THE INCREASE IN NITROGEN TEMPERATURE (TO 150° F) WILL PREVENT THE PRECIPITATION OF BORON AT THE END OF THE BUBBLER TUBES AND THE FALSE INDICATION OF BAST LEVEL HIGHER THAN ACTUAL. N_2 TEMPERATURE WILL BE MEASURED AND PROVIDED TO A TEMPERATURE CONTROLLER AND HEAT TRACE RECORDER (14A).

THE HEAT TRACE WILL CONSIST OF FOUR INDEPENDENTLY REGULATED SECTIONS, ONE FOR EACH BUBBLER HIGH PRESSURE SENSING LINE. THE HEAT TRACE AND THEIR CONTROLLERS SHALL BE DESIGNATED NON-CLASS 1E, HOWEVER, THE ATTACHMENT OF THE HEAT TRACE AND INSULATION TO THE BUBBLER SENSING LINES WILL BE ANALYZED TO ENSURE THE INTEGRITY OF THE EXISTING TUBING AND TUBING SUPPORTS. AS PART OF EWR 3881, MECHANICAL ENGINEERING SHALL BE RESPONSIBLE FOR THIS ANALYSIS.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE (1) MAJOR AND MINOR FIRES, (2) A SEISMIC EVENT.

NEW WIRING AND CABLE WILL BE REQUIRED FOR THIS MODIFICATION WHICH COULD ADD TO THE FIRE LOADING OF THE PLANT, THEREFORE, THE DESIGN CRITERIA REQUIRES THAT ALL SUCH CABLE MEET THE IEEE-383-1974 FLAME TEST REQUIREMENTS. BECAUSE OF THIS THERE WILL BE NO SIGNIFICANT INCREASE OF FIRE LOADING CAUSED BY THIS MODIFICATION.

AN APPENDIX R CONFORMANCE REVIEW SHALL BE PREPARED TO DETERMINE IF THE PROPOSED MODIFICATION IMPACTS APPENDIX R COMPLIANCE.

THIS MODIFICATION WILL BE REVIEWED TO ENSURE THAT FAILURE OF ANY ELECTRICAL CABLE INSTALLED AS PART OF THIS MODIFICATION WILL NOT RESULT IN THE DISABLING OF VITAL EQUIPMENT NEEDED TO SAFELY SHUT DOWN THE PLANT DURING POSTULATED FIRES.

THE PROPOSED HEAT TRACE AND THEIR CONTROLLERS ARE DESIGNATED NON-CLASS 1E, NON-SEISMIC CATEGORY 1, HOWEVER, THE ATTACHMENT OF THE HEAT TRACE AND INSULATION TO THE BUBBLER SENSING LINES SHALL BE SEISMICALLY DESIGNED TO ENSURE THAT THE SEISMIC QUALIFICATION OF THE EXISTING SENSING LINES IS NOT DEGRADED. IN ADDITION, NEW HEAT TRACE TEMPERATURE CONTROLLERS AND ELECTRICAL CONDUIT SHALL BE SEISMICALLY MOUNTED PER USNRC REG. GUIDE 1.29 SECTION C.2. THEREFORE, THIS MODIFICATION WILL NOT DEGRADE THE CAPABILITY OF ANY SAFETY RELATED EQUIPMENT TO PERFORM ITS INTENDED SAFETY FUNCTION.

THEREFORE, THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND FOR THE MITIGATION OF THE CONSEQUENCES HAVE NOT BEEN AFFECTED.

IN ACCORDANCE WITH THE PROVISIONS OF 10CFR50.59, THIS MODIFICATION DOES NOT PRESENT AND UNREVIEWED SAFETY QUESTION BECAUSE:

1) THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT IS NOT INCREASED.

   THE PURPOSE OF THIS MODIFICATION IS TO IMPROVE THE RELIABILITY OF, AND DECREASE THE REQUIRED MAINTENANCE FOR THE BAST LEVEL INDICATION SYSTEM. WHILE THE COMPONENTS AND INSTRUMENTATION ADDED BY THIS MODIFICATION ARE NOT CLASSIFIED SEISMIC CLASS 1E, ALL COMPONENTS WILL BE MOUNTED SO AS NOT TO DEGRADE THE SEISMIC CLASSIFICATION OF EXISTING SYSTEMS. THE APPENDIX R CONFORMANCE REVIEWS WILL INSURE COMPLIANCE WITH APPENDIX R ANALYSIS. THE FAILURE OF NON-1E COMPONENTS WILL NOT DEGRADE THE CAPABILITY OF ADJACENT OR CONNECT 1E SYSTEMS.

2) THE POSSIBILITY FOR AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS IS NOT CREATED.

   AS DISCUSSED ABOVE, THIS MODIFICATION WILL INCREASE THE RELIABILITY OF THE BAST LEVEL INDICATION SYSTEM WHILE MAINTAINING THE DESIGN CAPABILITIES OF INTERFACING SYSTEMS. THEREFORE, THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE IS NOT CREATED.

3) THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION IS NOT REDUCED.

TECHNICAL SPECIFICATION TABLE 4.1 INCORPORATES THE REQUIREMENT TO ROD THE BAST BUBBLER TUBES WEEKLY. RODDING THE TUBES INSURES THAT THE LEVEL INDICATION SYSTEM REMAINS OPERABLE. IF COMPLETELY SUCCESSFUL, THIS MODIFICATION WILL PREVENT THE PRECIPITATION OF BORON AT THE ENDS OF THE TUBES MAKING THE RODDING UNNECESSARY. HOWEVER, A TECHNICAL SPECIFICATION CHANGE WILL NOT BE SUBMITTED UNTIL PREVENTION OF PRECIPITATION IS VERIFIED. THEREFORE, GIVEN THE DESIGN REQUIREMENTS FOR THE MODIFICATION DISCUSSED ABOVE, THE MODIFIED LEVEL INDICATION SYSTEM WILL BE AT LEAST AS RELIABLE AND ACCURATE AS THE CURRENT CONFIGURATION. THE MARGIN OF SAFETY IS NOT REDUCED.

<u>EWR 4773B</u>
<u>ADVANCED DIGITAL FEEDWATER CONTROL - PHASE B</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES:

1) MODIFYING THE EXISTING FEEDWATER CONTROL VALVES (FCVS) FAIL OPEN/CLOSED LOGIC TO ALLOW THE MAIN AND BYPASS (FCVS) TO OPERATE IN AUTOMATIC CONTROL DURING TURBINE TRIP CONDITIONS AND THROUGHOUT PLANT STARTUP.

GINNA CURRENTLY HAS LOGIC FOR FAIL-OPEN OR FAIL-CLOSED OF THE MAIN AND BYPASS FCVS UPON TURBINE TRIP (OR THE TURBINE NOT BEING LATCHED) WITH THE MAIN FCVS IN AUTOMATIC CONTROL. THE FCVS ARE FAIL-OPEN UPON TURBINE TRIP WITH TAVG > 554½F AND FAIL-CLOSED UPON TURBINE TRIP WITH TAVG < 554½F. PRESENTLY, THE MAIN FCVS MUST BE PLACED IN MANUAL CONTROL FOR FEEDWATER CONTROL MODULATION WITH THE TURBINE TRIPPED OR BEFORE THE TURBINE IS LATCHED DURING STARTUP.

THE PROPOSED LOGIC MODIFICATION WILL DELETE THE EXISTING FAIL-OPEN LOGIC AND REPLACE THE FAIL-CLOSED LOGIC WITH ACTUATION UPON REACTOR TRIP.

THE FOLLOWING WILL BE PERFORMED:

A) REMOVE THE EXISTING FAIL OPEN LOGIC FOR SOLENOID S3 FOR BOTH THE MAIN AND BYPASS FEEDWATER CONTROL VALVES.

B) REMOVE THE EXISTING TURBINE TRIP LOGIC FOR SOLENOID S2 FOR BOTH MAIN AND BYPASS VALVES AND REPLACE WITH REACTOR TRIP LOGIC.

EWR 4773 WILL INTERFACE WITH EWR 4894. EWR 4894 WAS CREATED TO ADDRESS THE CONCERNS OF NRC INFORMATION NOTICE (IN) 88-24, FAILURES OF AIR-OPERATED VALVES AFFECTING SAFETY-RELATED SYSTEMS.

EWR 4894 WOULD HAVE REPLACED SOLENOID VALVES 14900S3, 14901S3, 14902S3 AND 14903S3. THE EXISTING SOLENOID VALVES ARE RATED FOR 60 PSI, HOWEVER THE INSTRUMENT AIR AS SUPPLIED IS 100 PSI TO MEET THE FW BYPASS OPERATORS MINIMUM REQUIRED PRESSURE FOR ACTUATION.

EWR 4894 WILL BE USED AS THE MECHANISM TO REMOVE THE FOLLOWING SOLENOIDS:

14900S3 (VALVE V4271)
14901S3 (VALVE V4269)
14902S3 (VALVE V4272)
14903S3 (VALVE V4270)

2) MODIFYING THE EXISTING SAFETY INJECTION (SI) LOGIC TO ALLOW THE BYPASS FCVS TO BE OPERATED DURING PERFORMANCE OF FUNCTIONAL RESTORATION PROCEDURE FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

RELAYS SIAF1 AND SIAF2 LOCATED IN THE SAFEGUARDS INITIATION RACKS WILL BE DISABLED. THIS WILL ALLOW OPERATION OF THE FEEDWATER BYPASS VALVES FOLLOWING MANUAL RESET OF SAFEGUARDS INITIATION. FOLLOWING AN SI RESET, A REOCCURRING SI SIGNAL WILL REINITIATE FEEDWATER ISOLATION.

3) DISABLING THE AUTO FUNCTION OF THE A AND B SECONDARY CHEMICAL ADDITION PUMP CONTROL SWITCHES. THESE SWITCHES ARE PRESENTLY HELD IN THE MANUAL POSITION SINCE THE AUTO OPERATION WAS DISABLED DURING THE ADFCS INSTALLATION.

REVISION 0 OF THIS DESIGN CRITERIA AND SAFETY ANALYSIS WAS NOT APPROVED BY PORC SINCE THE WESTINGHOUSE SAFETY EVALUATION WAS NOT YET RECEIVED. REVISION 1 INCORPORATED MINOR COMMENTS AND TYPOGRAPHICAL ERRORS.

A REVIEW HAS BEEN MADE OF EVENTS ANALYZED IN THE GINNA STATION UFSAR AND EVENTS REQUIRING ANALYSIS BY USNRC REGULATORY GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE (1) MAJOR AND MINOR FIRES, (2) EVALUATION OF MARGIN TO EXCESSIVE COOLDOWN, (3) EVALUATION OF TURBINE TRIP TRANSIENTS FOR LEVEL CONTROL WITH THE ADFCS IN AUTOMATIC AND MODIFIED OPEN/CLOSE LOGIC AND (4) DISABLING OF FEEDWATER ISOLATION FUNCTION FOR SI FOR RELAYS SIAF1 AND SIAF2 AND (5) A SEISMIC EVENT.

NEW FIELD WIRING AND CABLE WILL BE REQUIRED FOR THIS MODIFICATION THAT COULD ADD TO THE FIRE LOADING OF THE PLANT, THEREFORE, THE DESIGN CRITERIA REQUIRES THAT ALL SUCH CABLE MEET THE IEEE-383-1974 FLAME TEST REQUIREMENTS. BECAUSE OF THIS THERE WILL BE NO SIGNIFICANT INCREASE OF FIRE LOADING CAUSED BY THIS MODIFICATION.

WESTINGHOUSE WILL PERFORM AN ANALYSIS (SEE ATTACHED) THAT SUPPORTS THE REPLACEMENT OF THE TURBINE TRIP LOGIC WITH REACTOR TRIP LOGIC FOR THE FEEDWATER FCV ISOLATION FUNCTION AND REMOVAL OF THE FEEDWATER FCV FAIL OPEN FUNCTION. THIS ANALYSIS WILL BE REVIEWED AND APPROVED BY RG&E. THE EXISTING AND PROPOSED CHANGES ARE NOT SAFETY-RELATED FOR THE ADDITION OF A REACTOR TRIP RELAY AND DELETION OF THE FEEDWATER FCV FAIL OPEN LOGIC.

AN ASSESSMENT BY RG&E WILL BE PERFORMED UPON RECEIPT OF THE ABOVE DESCRIBED WESTINGHOUSE ANALYSES. WHEN THE RESULTS ARE CONFIRMED TO BE ACCEPTABLE, HARDWARE MODIFICATION WILL BE PERMITTED.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE REPLACEMENT OF THE TURBINE TRIP LOGIC WITH REACTOR TRIP LOGIC FOR THE FEEDWATER FCV ISOLATION FUNCTION AND THE REMOVAL OF THE FEEDWATER FCV FAIL OPEN FUNCTION WILL BE EVALUATED AND WILL SATISFY EXISTING ACCIDENT EVALUATION CRITERIA.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE ON ANALYSIS WILL SHOW THAT FEEDWATER ISOLATION RESULTING FROM REACTOR TRIP LOGIC INSTEAD OF TURBINE TRIP LOGIC AND REMOVING THE FEEDWATER FCV FAIL OPEN FUNCTION WILL MEET ALL EXISTING CRITERIA FOR UFSAR CHAPTER 15 ACCIDENTS.

FEEDWATER ISOLATION (FWI) SIGNALS MAY INADVERTENTLY INTERFERE WITH OPERATOR ACTIONS DURING PERFORMANCE OF EMERGENCY OPERATING PROCEDURES. DURING THE PERFORMANCE OF FUNCTIONAL RESTORATION PROCEDURE FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, STEPS 5 AND 9 DIRECT OPERATORS TO RESTORE FEEDWATER OR CONDENSATE FLOW TO THE STEAM GENERATORS VIA THE BYPASS FCVS. THIS IS NOT POSSIBLE WITH THE PRESENT FWI LOGIC IF ANY SI ACTUATION SIGNAL IS PRESENT, REGARDLESS IF SI IS RESET OR NOT.

FWI FROM AN SI IS INITIATED FOR THE A TRAIN BY RELAYS SI10X AND SIAF1 LOCATED IN SAFEGUARDS INITIATION (SI) RACKS. THE B TRAIN HAS SIMILAR LOGIC AND IS INITIATED BY SI20X AND SIAF2. THE SI10X AND SI20X RELAYS ARE RESET WHEN THE MASTER SI RELAYS ARE RESET. THE SIAF1 AND SIAF2 RELAYS, HOWEVER, ARE ACTUATED DIRECTLY BY AN SI INITIATION LOGIC SIGNAL AND CANNOT BE RESET WHEN THE INITIATING SIGNAL IS STILL PRESENT. SINCE REDUNDANT FWI INITIATION SIGNALS ARE ALREADY PROVIDED FOR EACH OF THE FW FCVS BY TRAIN A (RELAY SI10X) AND TRAIN B (RELAY SI20X), RELAYS SIAF1 AND SIAF2 WILL BE REMOVED FROM THE FWI CONTROL CIRCUITRY.

THIS WILL ALLOW THE BYPASS FEEDWATER FCVS TO BE OPERATED DURING THE PERFORMANCE OF FUNCTIONAL RESTORATION PROCEDURE FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE PROBABILITY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE PROBABILITY OF AN ACCIDENT IS NOT AFFECTED BY THE REMOVAL OF THESE TWO SI RELAYS. THE PROBABILITY OF AN OCCURRENCE OF AN ACCIDENT IS NOT APPLICABLE TO THE REMOVAL OF THE SIAF1 AND SIAF2 RELAYS.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THERE WILL REMAIN TWO TRAINS FOR SI ACTUATION AND EACH PROVIDING AN ISOLATION SIGNAL TO THE FW FCVS (MAIN AND BYPASS). REMOVAL OF THE SIAF1 AND SIAF2 RELAYS DOES NOT CHANGE THE SAFEGUARD ACTUATION SYSTEM'S FUNCTION OF INITIATING ISOLATION OF THE FEEDWATER SYSTEM IN THE EVENT OF AN SI SIGNAL.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE PROBABILITY OF OCCURRENCE OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE CHANGE PROVIDES A RESPONSE TO A PLANT RECOVERY PROCEDURE. THE CHANGE WILL ALLOW THE FEEDWATER CONTROL SYSTEM TO PROVIDE WATER TO THE STEAM GENERATORS IN THE EVENT OF AN SI SIGNAL OCCURRENCE WITH A SUBSEQUENT SI RESET. THIS IS IN ACCORDANCE WITH PLANT PROCEDURE FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK. FOLLOWING AN SI RESET, A REOCCURRING SI SIGNAL WILL REINITIATE FEEDWATER ISOLATION.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE EXISTING REDUNDANCY OF THE SI SYSTEM WILL BE MAINTAINED. THE CHANGE WILL ALLOW THE FEEDWATER CONTROL SYSTEM TO PROVIDE WATER TO THE STEAM GENERATORS IN THE EVENT OF AN SI SIGNAL OCCURRENCE WITH A SUBSEQUENT SI RESET. THIS IS IN ACCORDANCE WITH PLANT PROCEDURE FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK. FOLLOWING AN SI RESET, A REOCCURRING SI SIGNAL WILL REINITIATE FEEDWATER ISOLATION.

THE PROPOSED MODIFICATION WOULD NOT CREATE THE POSSIBILITY OF AN ACCIDENT OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE PROPOSED CHANGE WILL AFFECT ONLY THE RESET CAPABILITY OF THE FWI VALVES RESULTING FROM AN SI RESET.

THE PROPOSED MODIFICATION WOULD NOT CREATE THE POSSIBILITY OF A DIFFERENT TYPE OF MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY THAN ANY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE CHANGE WILL ALLOW THE FEEDWATER CONTROL SYSTEM TO PROVIDE WATER TO THE STEAM GENERATORS IN THE EVENT OF AN SI SIGNAL OCCURRENCE WITH A SUBSEQUENT SI RESET. THIS IS IN ACCORDANCE WITH PLANT PROCEDURE FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK. FOLLOWING AN SI RESET, A REOCCURRING SI SIGNAL WILL REINITIATE FEEDWATER ISOLATION.

REMOVAL/DISABLING OF THE A AND B SECONDARY CHEMICAL ADDITION PUMP AUTO/MANUAL CONTROL SWITCHES.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE PROBABILITY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE PROBABILITY OF AN ACCIDENT IS NOT AFFECTED BY THE REMOVAL OF THE AUTO FUNCTION FOR THE A AND B SECONDARY CHEMICAL ADDITION PUMPS. THESE PUMPS ARE CLASSIFIED NON-SAFETY RELATED.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THERE IS NO ACCIDENT EVALUATION REQUIRED FOR THIS SYSTEM.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-4833</u>
<u>SAS/PPCS NERP MODIFICATIONS</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE MODIFICATION TO PROVIDE A COMMUNICATIONS LINK BETWEEN THE TECHNICAL SUPPORT CENTER (TSC) AND GINNA SIMULATOR BUILDING.

REVISION 0 OF THIS DESIGN CRITERIA AND SAFETY ANALYSIS WAS NOT PRESENTED TO PORC. REVISION 1 OF THIS DESIGN CRITERIA AND SAFETY ANALYSIS PRESENTED AND APPROVED BY PORC ON 1/4/90, INCORPORATED ADDITIONAL CONDUIT TO BE INSTALLED TO EXTEND THE SUBWAY SYSTEM TO THE SECURITY GUARDHOUSE AND TO THE BROOKWOOD TRAINING CENTER. PORC RESTRICTED WORK SCOPE TO ALLOW TRENCHWORK, CONDUIT INSTALLATION AND CABLE PULLING UNTIL PLANT COMMENTS CONCERNING TEST SPECIFICATIONS WERE RESOLVED.

REVISION 2 OF THIS DESIGN CRITERIA AND SAFETY ANALYSIS WAS NOT PRESENTED TO PORC. REVISION 2 RESOLVED ALL PLANT COMMENTS THAT CONCERNED REVISION 1.

REVISION 2 ALSO EXPANDED THE SCOPE OF THIS EWR TO INCLUDE INSTALLATION OF CONDUIT FROM MANHOLE 1B TO A NEW MANHOLE G SO THAT A FIBER OPTIC CABLE FROM STATION 13A TO THE GINNA TRAINING CENTER CAN BE INSTALLED. ONLY THE INSTALLATION OF THE ADDITIONAL CONDUIT AND NEW MANHOLE ARE INCLUDED IN THE SCOPE OF THIS WORK.

THE PURPOSE OF REVISION 3 IS DUE TO A CHANGE TO ELECTRICAL REQUIREMENTS, SECTION 17.1, AND FIRE PROTECTION REQUIREMENTS, SECTION 27.1.

DUE TO THE MATERIALS NECESSARY FOR MANUFACTURING THE MULTI-CONDUCTOR CABLE, IEEE-383 1974 FLAME TEST RATING CANNOT BE OBTAINED. THE APPENDIX R CONFORMANCE REVIEW WILL ALSO BE REVISED TO DOCUMENT THE INCREASED COMBUSTIBLE LOADING EFFECT.

THE SAFETY ASSESSMENT SYSTEM (SAS) AND PLANT PROCESS COMPUTER SYSTEM (PPCS) WERE INSTALLED IN 1986 AND ARE DESIGNED TO PROVIDE REAL-TIME PLANT DATA TO OPERATORS IN THE CONTROL ROOM AND PERSONNEL IN THE TECHNICAL SUPPORT CENTER (TSC), THE EMERGENCY OFFSITE FACILITY (EOF) AND THE ENGINEERING SUPPORT CENTER (ESC). THE SAS AND PPCS SYSTEMS ARE NOW CONSIDERED THE PRIMARY MEANS OF PROVIDING THE TSC, EOF, AND ESC WITH GINNA PLANT DATA DURING AN UNUSUAL EVENT. HOWEVER, DURING THE ANNUAL GINNA EMERGENCY DRILLS, THE SAS AND PPCS SYSTEMS ARE NOT CAPABLE OF PROVIDING DATA CONSISTENT WITH THE ACCIDENT SCENARIO BEING EXERCISED SINCE ONLY ACTUAL GINNA REAL-TIME DATA IS DISPLAYED. THIS MODIFICATION WILL PROVIDE THE CAPABILITY TO SUPPLY GINNA SIMULATOR DATA TO THE SAS AND PPCS VIDEO DISPLAY TERMINALS AND PRINTERS LOCATED IN THE TSC, EOF, AND ESC. DATA ORIGINATING FROM THE SIMULATOR, AND CONSISTENT WITH THE EMERGENCY EXERCISE, WILL THEN BE USED IN SUPPORT OF THE NUCLEAR EMERGENCY RESPONSE PLAN TRAINING AND EXERCISES. IN ORDER TO PROVIDE THIS CAPABILITY THE FOLLOWING MODIFICATIONS ARE PROPOSED.

IN THE TSC, TWO PERIPHERAL SWITCH RACKS WILL BE INSTALLED. ONE RACK WILL ACCOMMODATE THE PERIPHERAL SWITCHES FOR THE SAS SWITCHABLE DEVICES, THE SECOND RACK WILL ACCOMMODATE THE PPCS SWITCHABLE DEVICES. TO MAINTAIN INDEPENDENCE OF POWER SOURCES, THE SAS AND PPCS RACKS WILL BE POWERED FROM THEIR RESPECTIVE POWER DISTRIBUTION PANELS.

THE SAS SIMULATOR (SASS) SOFTWARE SYSTEM AND THE PPCS SIMULATOR (PPCSS) SOFTWARE SYSTEM WILL BE MODIFIED TO IDENTIFY THE ADDITIONAL PLANT PERIPHERAL DEVICES.

THE ROUTE OF THE FIBER OPTIC CABLE FROM THE SIMULATOR TO THE TSC WILL REQUIRE THE INSTALLATION OF UNDERGROUND CONDUIT AND MANHOLES. THE LOCATION OF THE NEW SUBWAY SYSTEM WILL BE CHOSEN SUCH THAT IT CAN BE USED TO PERMANENTLY ROUTE POWER CABLES BETWEEN HI MAST LIGHTING POLES #7 AND #8. THE ORIGINAL POWER CABLES FOR THESE LIGHTING POLES WERE INSTALLED AS DIRECT BURIED CABLES. A FAULT HAS DEVELOPED IN THESE CABLES, AND AS A TEMPORARY MEANS OF REPAIR, A SET OF CABLES WERE INSTALLED BETWEEN POLES #7 AND #8 ABOVE GROUND IN CONDUIT. THE PERMANENT REPLACEMENT OF THE HI MAST LIGHTING CABLES WAS REQUESTED AS PART OF EWR 5004. SINCE IT WILL BE LOGICAL AND ECONOMICAL TO REPLACE THE CABLES WHEN THE SUBWAY SYSTEM IS INSTALLED, THE CABLE REPLACEMENT PORTION OF EWR 5004 WILL ACTUALLY BE PERFORMED AS PART OF THIS MODIFICATION.

THE NEW SUBWAY SYSTEM WILL ALSO BE EXTENDED TO PROVIDE A ROUTE FOR ADDITIONAL COMMUNICATION CABLES THAT ARE REQUIRED TO SUPPORT EWR 4858, GINNA TRAINING CENTER. THEREFORE, NEW CONDUIT WILL BE INSTALLED TO EXTEND THE SUBWAY SYSTEM TO THE SECURITY GUARDHOUSE AND TO THE BROOKWOOD TRAINING CENTER.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA UFSAR AND THE EVENTS REQUIRING ANALYSIS BY THE USNRC REGULATORY GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE:

(1) SEISMIC EVENT
(2) MAJOR AND MINOR FIRES
(3) INDUSTRIAL SECURITY

THIS MODIFICATION DOES NOT INSTALL ANY NEW OR MODIFY ANY EXISTING EQUIPMENT NECESSARY FOR THE SAFE SHUTDOWN OF THE PLANT. NEW EQUIPMENT WILL NOT BE LOCATED IN ANY AREA THAT CONTAINS SAFETY RELATED EQUIPMENT. THE MODIFICATION IS, THEREFORE, DESIGNED AS NON-SEISMIC.

THIS MODIFICATION INVOLVES THE INSTALLATION OF NEW EQUIPMENT IN A GINNA FIRE AREA. AN APPENDIX R EVALUATION IS, THEREFORE, REQUIRED.

IN ACCORDANCE WITH THE DESIGN CRITERIA, THE UNDERGROUND SUBWAY SYSTEM AND THE INSTALLATION OF CONDUIT FOR THE HI MAST LIGHTING POLES #7 AND #8 WILL BE UNDERGROUND INSIDE AND OUTSIDE OF THE PROTECTED AREA FENCE. NO UNDERGROUND SAFETY RELATED ELECTRICAL CIRCUITS WITHIN THE PROTECTED AREA FENCE BOUNDARY WILL BE AFFECTED. THIS MODIFICATION DOES NOT INVOLVE INSTALLATION OF NEW EQUIPMENT IN VITAL AREAS, AND THEREFORE DOES NOT AFFECT SECURITY OPERATIONS.

THEREFORE, THIS MODIFICATION DOES NOT DEGRADE THE CAPABILITY OF ANY SAFETY SYSTEM TO PERFORM ITS FUNCTION. THE ASSUMPTIONS AND CONCLUSIONS OF EXISTING ANALYSES ARE UNCHANGED. NO NEW TYPES OF EVENTS ARE POSTULATED.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-4894</u>
<u>SOV REPLACEMENT FOR THE MAIN AND BYPASS FW CONTROL VALVES</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THREE ISSUES TO THE MAIN AND BYPASS FW CONTROL VALVE SOV'S:

1. MAXIMUM OPERATING PRESSURE DIFFERENTIAL (MOPD) RATING OF THE SOV'S ARE NOT CONSISTENT WITH THE INSTRUMENT AIR SUPPLY SYSTEM OPERATING PRESSURE.

2. IMPROVE THE CURRENT CLOSURE (STROKE) TIMES OF THE MAIN FEEDWATER CONTROL VALVES TO INCREASE THE MARGINS BETWEEN THE ACTUAL OPEN TO CLOSE STROKE TIMES AND THOSE TIMES REQUIRED TO SATISFY THE ASSUMPTIONS OF THE SYSTEM DESIGN BASIS.

3. ELIMINATION OF THE S3 SOV'S IN SUPPORT OF EWR 4773 "ADVANCED DIGITAL FEEDWATER CONTROL SYSTEM (ADFCS) PHASE B" MODIFICATIONS.

THE PHYSICAL MODIFICATION ASSOCIATED WITH ISSUE #1 INVOLVES THE REPLACEMENT OF THE EXISTING S1 AND S2 SOV'S FOR FEEDWATER BYPASS CONTROL VALVES 4271 AND 4272 WITH SOLENOID VALVES THAT HAVE MOPD RATINGS GREATER THAN THE EXISTING SYSTEM CONDITIONS. THE NEW SOV'S WILL MEET THE REQUIRED SYSTEM SPECIFICATIONS AND ELIMINATE THE OPERATIONAL ISSUE ASSOCIATED WITH THE SOV (MOPD) RATINGS.

THE PHYSICAL MODIFICATION ASSOCIATED WITH ISSUE #2 INVOLVES THE REPLACEMENT OF THE S1 AND S2 SOV'S ON THE MAIN FW CONTROL VALVES 4269 AND 4270, WITH VALVES THAT ARE CONSTRUCTED WITH LARGER DIAMETER INTERNAL PORTS THAT WILL PERMIT FASTER VALVE CLOSING TIMES.

THE PHYSICAL MODIFICATION ASSOCIATED WITH ISSUE #3 INVOLVES THE REMOVAL OF THE EXISTING S3 SOV'S AND THE REWORK OF ALL ASSOCIATED INSTRUMENT AIR TUBING FOR THE MAIN AND BYPASS FW CONTROL VALVES 4269, 4270, 4271 AND 4272. THE PHYSICAL ELECTRICAL WORK SCOPE ASSOCIATED WITH THE REMOVAL OF THE VALVES WILL BE INCLUDED WITHIN THE DESIGN SCOPE OF EWR 4773B.

A REVIEW HAS BEEN MADE OF ALL THE DESIGN BASIS EVENTS REQUIRING ANALYSIS BY NRC REGULATORY GUIDE 1.70 AND THE GINNA STATION UFSAR. THE EVENTS RELATED TO THIS MODIFICATION ARE: (1) SEISMIC EVENT, (2) MAJOR AND MINOR FIRES, AND (3) FEEDWATER SYSTEM MALFUNCTIONS/STEAMLINE BREAKS/LARGE AND SMALL BREAK LOCA'S.

THE ELECTRICAL AND CONTROL LOGIC CHANGES ASSOCIATED WITH THIS MODIFICATION ARE WITHIN THE SCOPE OF EWR 4773B. THE DESIGN BASIS EVENTS RELATED TO THESE PORTIONS OF THE MODIFICATION THAT REQUIRE ANALYSIS ARE IDENTIFIED IN THE SAFETY ANALYSIS FOR EWR 4773B.

ALL SOLENOID VALVES AND THEIR ELECTRICAL CONNECTIONS REQUIRE QUALIFICATION TO SEISMIC CATEGORY I REQUIREMENTS AS SPECIFIED IN SECTION 3.1 AND 3.2 OF EWR 4894 DESIGN CRITERIA. IN THE EVENT OF AN EARTHQUAKE, THE SOLENOID VALVE AND CABLE RESTRAINT DESIGN REQUIREMENTS WILL ASSURE THAT THE SOLENOID VALVES ARE NOT ACCELERATED OR STRESSED BEYOND THEIR DESIGN VALUES SO THAT THEIR ASSOCIATED PROCESS VALVES ARE ABLE TO ASSUME THEIR DESIGN FAIL-SAFETY POSITION OR OPERATE POST-ACCIDENT.

ALL SAFETY CLASS 3 TUBING ASSOCIATED WITH THIS MODIFICATION SHALL BE QUALIFIED AS SEISMIC CATEGORY I TO THE EXTENT THAT VALVE CLOSURE IS ASSURED (IE. ACTUATOR EXHAUST LINE TUBING SHALL MAINTAIN STRUCTURAL INTEGRITY TO ENSURE THE LINE DOES NOT KINK AND BLOCK THE VENT PATH).

THE NEW SOLENOID VALVES SHALL BE DESIGNED TO OPERATE WITHOUT HINDRANCE FROM THE POSTULATED ADVERSE ENVIRONMENT AS REQUIRED BY 10CFR50.49. THE UPGRADED VALVES SHALL ENSURE THE OPERABILITY OF THEIR ASSOCIATED PROCESS VALVES WHEN EXPOSED TO AN ACCIDENT ENVIRONMENT.

THE SCOPE OF THIS EWR MODIFICATION INVOLVES THE REMOVAL/REPLACEMENT OF EXISTING SOV'S ALONG WITH THE REWORK OF EXISTING PIPING/TUBING AND ELECTRICAL CONNECTIONS. NO ADDITIONAL COMBUSTIBLES WILL BE UTILIZED THAT WOULD AFFECT ANY FIRE LOADINGS. THEREFORE, THESE TYPE MODIFICATIONS WILL NOT DEGRADE THE LEVEL OF FIRE PROTECTION/DETECTION AND ALTERNATE SHUTDOWN CAPABILITIES EXISTING IN THE PLANT AND WILL NOT REQUIRE AN APPENDIX R CONFORMANCE REVIEW.

THE SCOPE OF THE PROPOSED MODIFICATION WILL ONLY IMPROVE THE CLOSURE TIME PERFORMANCE OF THE FW CONTROL VALVES. THEREFORE, THE ASSUMPTIONS CONSIDERED FOR THE FW VALVE OPERATIONS DURING RELATED DESIGN BASIS EVENTS (I.E., FW SYSTEM MALFUNCTION/STEAMLINE BREAKS/LARGE AND SMALL BREAK LOCA'S) REMAIN VALID.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR 4937</u>
<u>HEAT TRACE UPGRADE</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE REPLACEMENT OF THE FOLLOWING NON TECH. SPEC., ASBESTOS INSULATED HEAT TRACE CIRCUITS:

P12	S12	P29	S29	P30
S30	P34	S34	P54	S54
P56	S56	P67	S67	

THESE CIRCUITS WILL BE REPLACED WITH SELF REGULATING HEAT TRACE CABLE WHICH DOES NOT REQUIRE THERMOSTAT CONTROLS AND IS DESIGNED TO MAINTAIN THE ASSOCIATED PIPING SYSTEMS ABOVE THE SOLUBILITY TEMPERATURE FOR 13% BORIC ACID.

A REVIEW HAS BEEN MADE FOR ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND THE EVENTS REQUIRING ANALYSIS BY USNRC GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE: (1) RADIOACTIVE LIQUID WASTE SYSTEM MALFUNCTION THAT RESULTS IN THE RELEASE OF RADIOACTIVE LIQUIDS, (2) FIRE EVENT, AND (3) SEISMIC EVENT.

MALFUNCTION OF RADIOACTIVE LIQUID WASTE SYSTEM

THE POTENTIAL TO CAUSE A MALFUNCTION OF THE RADIOACTIVE LIQUID WASTE SYSTEM THAT COULD RESULT IN THE RELEASE OF RADIOACTIVE LIQUIDS IS UNCHANGED. THE NEW HEAT TRACE AND INSULATING SYSTEMS OPERATING BASIS REMAIN THE SAME AS THE EXISTING SYSTEMS THAT ARE REPLACED. THIS SYSTEM DOES NOT INCREASE THE POSSIBILITY OF PIPE RUPTURE. SHOULD THE SYSTEM FAIL, THE POTENTIAL EXISTS TO PLUG A LINE DUE TO BORON CRYSTALLIZATION. NO CREDIT IS TAKEN FOR THESE NEWLY HEAT TRACED LINES TO OPERATE IN ANY UFSAR CHAPTER 15 EVENT. THE NEW SYSTEMS MAINTAIN PROCESS TEMPERATURES WITHIN OPERATING PARAMETERS BY DELIVERING ELECTRICAL POWER THROUGH A RESISTANCE TO DEVELOP HEAT. THIS IS THE SAME PRINCIPLE AS THE SYSTEMS THAT ARE BEING REPLACED. ALARMS AND VISUAL VERIFICATION THAT ARE IN EFFECT WILL REMAIN THE SAME TO VERIFY PROPER OPERATION. THIS MODIFICATION DOES NOT AFFECT THE OPERATION OF THE RADIOACTIVE LIQUID WASTE SYSTEM.

FIRES

THE RESPONSE OF THE RADIOACTIVE LIQUID WASTE SYSTEM AND BORIC ACID SYSTEM TO THE FIRE EVENT REMAINS UNCHANGED. THE REQUIREMENTS OF SECTION 27 OF THE DESIGN CRITERIA ASSURE THAT THE POTENTIAL FOR FIRE HAS NOT BEEN INCREASED. ADVERSE CONSEQUENCES OF A FIRE HAVE NOT BEEN CHANGED.

EARTHQUAKES

CONSISTENT WITH IEEE STANDARD 622-1987, THE HEAT TRACING SYSTEMS SHALL BE NON-SEISMIC CATEGORY 1. THE SEISMIC CATEGORY 1 PIPING SYSTEMS TO WHICH THE HEAT TRACE AND INSULATION SYSTEMS ARE TO BE ATTACHED WILL BE ANALYZED TO ENSURE THAT SEISMIC INTEGRITY HAS NOT BEEN ADVERSELY AFFECTED. THE RESPONSE OF THE BORIC ACID SYSTEM OR RADIOACTIVE LIQUID WASTE SYSTEM TO A SEISMIC EVENT IS UNCHANGED.

THE. PROPOSED MODIFICATION WOULD NOT INCREASE THE PROBABILITY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE AS STATED IN 3.3.1, NO INCREASE IN PIPE RUPTURE IS ANTICIPATED AND NO CREDIT FOR THESE LINES, AS COULD RESULT FROM PIPE PLUGGING DUE TO BORON PRECIPITATION, IS TAKEN IN UFSAR CHAPTER 15 EVENTS.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THERE ARE NO ACCIDENTS EVALUATED IN THE SAFETY ANALYSIS REVIEW THAT MAY HAVE THEIR RADIOLOGICAL CONSEQUENCES ALTERED AS A DIRECT RESULT OF THIS MODIFICATION.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE PROBABILITY OF OCCURRENCE OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE NEW PROPOSED EQUIPMENT IS MORE RELIABLE THAN THAT BEING REPLACED. THE NON-1E STATUS OF THE HEAT TRACE IS ACCEPTABLE BECAUSE IN THE EVENT OF AN EARTHQUAKE, NO RELIANCE ON ANY AFFECTED EQUIPMENT EXISTS.

THE PROPOSED MODIFICATION WOULD NOT INCREASE THE CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THERE ARE NO ANALYZED ACCIDENT CONSEQUENCES DESCRIBED IN THE UFSAR THAT ARE AFFECTED BY MALFUNCTION OF THE MODIFIED EQUIPMENT.

THE PROPOSED MODIFICATION WOULD NOT CREATE THE POSSIBILITY OF AN ACCIDENT OF A DIFFERENT TYPE THAN PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THE POSTULATED ACCIDENT SCENARIOS GERMANE TO THIS MODIFICATION ARE COMPARABLE TO THE TYPES OF ACCIDENTS THAT HAVE BEEN EVALUATED IN THE SAFETY ANALYSIS REVIEW. (SEE UFSAR SECTION 15.7)

THE PROPOSED MODIFICATION WOULD NOT CREATE THE POSSIBILITY OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY DIFFERENT THAN PREVIOUSLY EVALUATED IN THE UFSAR BECAUSE THERE ARE NO FAILURE MODES OF A DIFFERENT TYPE THAT ARE CREATED BY THIS MODIFICATION WHEN COMPARED TO THE MODES OF FAILURES THAT HAVE BEEN EVALUATED IN THE SAFETY ANALYSIS REVIEW.

THE PROPOSED MODIFICATION WOULD NOT REDUCE ANY MARGIN OF SAFETY AS DEFINED IN THE BASIS OF ANY TECHNICAL SPECIFICATION BECAUSE THERE ARE NO MARGINS OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION THAT ARE INVOLVED WITH THIS MODIFICATION.

<u>EWR-4951</u>
<u>REPLACE TDAFW PUMP RECIRC CHECK VALVE</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE REPLACEMENT OF THE TURBINE-DRIVEN AUXILIARY FEED WATER PUMP (TDAFW) RECIRCULATION LINE CHECK VALVE V4023, AND THE BY-PASS LINE AND VALVE OF TDAFW PUMP LUBE OIL COOLER. THIS REPLACEMENT CHECK VALVE IS IN ASME B&PV CODE, SECTION III, CLASS 3 CHECK VALVE OF A DESIGN MORE SUITABLE FOR THE PLANT CONDITIONS.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70.

THE DESIGN BASIS EVENTS RELATED TO THIS PROPOSED MODIFICATION ARE:

A. FIRES
B. SEISMIC EVENT
C. LOSS OF NORMAL FEEDWATER FLOW WHICH INCLUDES:

1. LOSS OF MAIN FEEDWATER WITH OFFSITE POWER AVAILABLE.

2. LOSS OF MAIN FEEDWATER WITHOUT OFFSITE POWER AVAILABLE.

3. RUPTURE OF FEEDWATER LINE

D. RUPTURE OF A MAIN STEAM LINE

E. LOSS OF ALL AC POWER

F. LOSS-OF-COOLANT ACCIDENT

A) FIRES

AN APPENDIX R CONFORMANCE VERIFICATION SHALL BE PREPARED TO DETERMINE IF THE PROPOSED MODIFICATION IMPACTS APPENDIX R COMPLIANCE.

- NON-COMBUSTIBLE AND HEAT-RESISTANT MATERIALS SHALL BE USED.
- EXISTING FIRE BARRIERS SHALL NOT BE DEGRADED NOR SHALL THE PERFORMANCE OF ANY EXISTING FIRE PROTECTION EQUIPMENT BE AFFECTED.
- ALL NEW PENETRATIONS THROUGH FIRE BARRIERS SHALL BE SEALED WITH APPROPRIATE FIRE SEALS.
- ALL NEW PENETRATIONS THROUGH WALLS, CEILINGS OR FLOORS WHICH ARE NOT FIRE BARRIERS BUT PROVIDE A HALON FIRE SUPPRESSION SYSTEM BOUNDARY SHALL BE SEALED WITH APPROPRIATE AIR SEALS.
- MATERIALS USED IN THIS MODIFICATION SHALL NOT INCREASE THE PROBABILITY OR CONSEQUENCES OF A FIRE.

THEREFORE, THIS MODIFICATION WILL NOT INCREASE THE PROBABILITY OR CONSEQUENCES OF A FIRE.

B) A SEISMIC EVENT

NEW AND EXISTING SAFETY-RELATED PIPING (INCLUDING THE RE-ROUTING OF EXISTING PIPING, IF REQUIRED), VALVE AND SUPPORTS INVOLVED IN THIS MODIFICATION WILL BE EVALUATED IN REGARD TO A SEISMIC EVENT. EVALUATION CRITERIA SHALL BE CONSISTENT WITH THE GINNA STATION SEISMIC UPGRADE PROGRAM. THESE CRITERIA WILL ENSURE A MODIFICATION DESIGN THAT DOES NOT DEGRADE THE AUXILIARY FEEDWATER SYSTEM'S CAPABILITY TO WITHSTAND A SEISMIC EVENT.

THE DESIGN CRITERIA REQUIRES THAT ANY ADDITIONS TO THE PLANT WHOSE FAILURE COULD CAUSE DAMAGE TO SAFETY-RELATED EQUIPMENT BE RESTRAINED AND SUPPORTED IN A MANNER COMPARABLE TO SEISMIC CATEGORY I ITEMS, TO PREVENT SUCH FAILURE. THEREFORE, A SEISMIC EVENT WILL NOT RESULT IN DEGRADATION OF SAFETY-RELATED EQUIPMENT AT GINNA STATION.

C) LOSS OF NORMAL FEEDWATER FLOW (OR EVENTS IDENTIFIED IN D, E, AND F ABOVE).

NEW PIPING AND VALVE INTERFACING WITH THE AUXILIARY FEEDWATER SYSTEM WOULD BE SIZED, SPECIFIED AND INSTALLED IN ACCORDANCE WITH EXISTING PIPING CLASSIFICATIONS AND CODE DESIGNATIONS FOR DESIGN, MATERIAL AND CONSTRUCTION.

THE CHECK VALVE IS A REPLACEMENT FOR AN EXISTING VALVE AND IS REQUIRED TO MEET THE SAME PERFORMANCE CRITERIA AS THE EXISTING VALVE. THE REPLACEMENT CHECK VALVE IS NOT EXPECTED TO EXPERIENCE SIMILAR FAILURES SINCE IT IS AN IMPROVED VALVE DESIGN WITH AN INCREASE IN PRESSURE CLASS.

THE VALVE WOULD BE REQUIRED TO BE SUFFICIENTLY OPEN TO PERFORM ITS SAFETY FUNCTION WHEN THE TDAFW PUMP IS IN OPERATION, THEREBY PROVIDING RECIRCULATION FLOW.

THE VALVE CONTROLS (INCLUDING RECIRCULATION LINE) WOULD BE IDENTICAL TO THE EXISTING VALVE CONTROLS.

THE VALVE WOULD BE REQUIRED TO MEET THE TESTING REQUIREMENTS AS DEFINED IN THE GINNA INSERVICE TEST PROGRAM.

CONSEQUENTLY, IN EVENT OF LOSS OF NORMAL FEEDWATER OR ANY OTHER EVENT REQUIRING AUXILIARY FEEDWATER FLOW, THEREBY SATISFYING DESIGN BASES REQUIREMENTS, THE NEW RECIRCULATION LINE CHECK VALVE IS REQUIRED TO MEET THE SAME PERFORMANCE CHARACTERISTICS AS THE EXISTING VALVE.

THE LUBE OIL COOLING WATER BYPASS VALVE SHALL SERVE ONLY TO CONTROL THE TEMPERATURE OF THE LUBE OIL DURING COLD WEATHER OPERATION. UFSAR SECTION 10.5.4.2 DESCRIBES A TEST CONDUCTED IN WHICH THE TDAFW PUMP WAS SUCCESSFULLY OPERATED FOR ALMOST 2 HOURS WITHOUT ANY COOLING WATER. THE TDAFW PUMP IS ONLY CREDITED FOR LONG-TERM OPERATION FOR SPECIFIC FIRE AND STATION BLACKOUT SCENARIOS. THESE EVENTS WOULD NOT PRECLUDE OPERATOR ACCESS TO THE TDAFW PUMPS. IN ADDITION, OPERATOR ACTION FOR THESE EVENTS IS NORMALLY ACCEPTED AS BEING REQUIRED DUE TO THEIR NATURE. THEREFORE, INSTALLATION OF THE BYPASS LINE, AND SUBSEQUENT ISOLATION OF SW TO THE LUBE OIL COOLER, ALLOWS SUFFICIENT TIME FOR ANY OPERATOR ACTION TO REOPEN THE LINE AS REQUIRED. NECESSARY PLANT

EMERGENCY, OPERATING, AND TESTING PROCEDURES WILL BE MODIFIED TO REQUIRE OPERATORS TO REOPEN THE LINE TO THE LUBE OIL COOLER WITHIN THIS TIME FRAME.

CONCLUSION

BASED ON THE ABOVE ANALYSIS, IT HAS BEEN CONCLUDED THAT:

A. THE MARGINS OF SAFETY DURING NORMAL OPERATION AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

B. THE STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR PREVENTION OF ACCIDENTS AND THE MITIGATION FOR THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

C. THE PROPOSED MODIFICATION DOES NOT INVOLVE A CHANGE IN THE GINNA TECHNICAL SPECIFICATIONS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-4996</u>
<u>L.T.O.P. SYSTEM RELIEF VALVES 8615A AND 8615B</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE INSTALLATION OF NEW FLANGE CONNECTIONS AT RELIEF VALVES 8615A AND 8615B INLETS.

REVISION 0 OF THE DESIGN CRITERIA AND SAFETY ANALYSIS WERE PRESENTED AND APPROVED BY THE COMMITTEE ON 2/7/90, MEETING NO. 90-014.

THE PURPOSE OF REVISION OF THE DESIGN CRITERIA AND SAFETY ANALYSIS IS TO DELETE REFERENCES TO BLANKING FOR HYDROTESTING PER PLANT COMMENTS FOR REVISION 0 AND REFORM AT THE PRELIMINARY SAFETY EVALUATION TO ADDRESS THE SEVEN 10CFR50.59 QUESTIONS.

A REVIEW HAS BEEN PERFORMED OF ALL EVENTS ANALYZED IN THE GINNA STATION UPDATED FINAL SAFETY ANALYSIS REPORT. THE EVENTS RELATED TO THIS MODIFICATION ARE:

1) FIRES
2) SEISMIC EVENTS
3) EFFECT ON L.T.O.P.

THE MODIFICATION WILL NOT INCREASE THE PROBABILITY OF OR THE EFFECTS OF A FIRE SINCE THE MATERIALS USED WILL MEET CRITERIA EQUAL TO OR GREATER THAN THOSE PRESENTLY INSTALLED.

THE ADDITION OF THE TWO FLANGES AT THE INLETS OF RELIEF VALVES 8615A AND 8615B WILL BE SEISMICALLY SUPPORTED AND THEREFORE WILL NOT AFFECT SAFETY RELATED EQUIPMENT.

THE MODIFICATION DOES NOT AFFECT THE N2 SUPPLY TO THE L.T.O.P. SYSTEM. NEITHER THE QUANTITY NOR THE PRESSURE IS AFFECTED BY THE INSTALLATION OF THE FLANGES. THEREFORE, THE PERFORMANCE OF L.T.O.P. IS NOT AFFECTED BY THIS MODIFICATION.

BASED ON THE ABOVE ANALYSIS:

1) THE MARGIN OF SAFETY DURING NORMAL OPERATING AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

2) THE STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR 4998</u>
<u>S/G CONTAINMENT PENETRATION</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE INSTALLATION OF REMOVABLE FLANGES ON EXISTING CONTAINMENT SPARE PENETRATION NO. 2 FOR ROUTING SGI/M CABLING DURING THE ANNUAL INSPECTION AND OUTAGE (AI&O).

REVISION 1 OF THE DESIGN CRITERIA AND SAFETY ANALYSIS WERE PRESENTED AND APPROVED BY THE PORC ON 3/14/90, MEETING NO. 90-025.

THE PURPOSE OF REVISION OF THE DESIGN CRITERIA AND SAFETY ANALYSIS IS TO CORRECT REFERENCE TO ANSI CODE IN SECTION 25.3. THIS WAS A PORC COMMENT OF DC/SA REVISION 1.

A REVIEW HAS BEEN PERFORMED OF ALL EVENTS ANALYZED IN THE GINNA STATION UPDATED FINAL SAFETY ANALYSIS REPORT. THE EVENTS RELATED TO THIS MODIFICATION ARE:

1) FIRES
2) SEISMIC EVENTS
3) NATURAL PHENOMENA
4) MISSILE HAZARDS
5) CONTAINMENT INTEGRITY

THE MODIFICATION WILL NOT ALTER THE AREA FIRE LOADING, THE SOURCES OF FIRE INITIATION, NOR THE ACCEPTABILITY OF THE CONSEQUENCES OF A FIRE SINCE THE MATERIAL USED WILL MEET CRITERIA EQUAL TO OR GREATER THAN THOSE PRESENTLY INSTALLED.

THE MODIFICATION SHALL BE DESIGNED TO WITHSTAND SEISMIC LOADING AND TO MAINTAIN THE STRUCTURAL INTEGRITY OF THE EXISTING PENETRATION.

THE EFFECTS OF SEVERE WEATHER CONDITIONS SHALL BE CONSIDERED IN THE DESIGN OF THE PENETRATION MODIFICATION. THEREFORE, THE EXISTING STRUCTURE WILL NOT BE DEGRADED BY THE PROPOSED MODIFICATION.

THE NEED FOR MISSILE PROTECTION FOR THIS PENETRATION SHALL BE EVALUATED DURING THE DESIGN OF THE PENETRATION MODIFICATION.

THE MODIFICATION TO THE EXISTING PENETRATION SHALL BE DESIGNED AND INSTALLED SUCH THAT CONTAINMENT INTEGRITY IS NOT COMPROMISED.

BASED ON THE ABOVE ANALYSIS:

1) THE MARGIN OF SAFETY DURING NORMAL OPERATING AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

2) THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY, PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT ARE NOT INCREASED AND THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A TYPE DIFFERENT FROM ANY PREVIOUSLY EVALUATED IN SAFETY ANALYSIS REPORT HAS NOT BEEN CREATED.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

EWR-5041
INTERMEDIATE BUILDING PIPE HANGERS

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE ADDITION AND/OR MODIFICATION OF PIPE SUPPORTS WHICH SUPPORT THE PORTION OF THE 10" DIAMETER CONDENSATE MAKEUP REJECT LINE LOCATED IN THE INTERMEDIATE BUILDING SUB-BASEMENT.

THIS PIPING IS CONSIDERED NON-SAFETY RELATED, BUT WILL BE SUPPORTED AND ANALYZED PER SEISMIC CATEGORY 1 CRITERIA TO ENSURE THAT OTHER SAFETY RELATED PIPING (SERVICE WATER) WILL NOT BE ADVERSELY AFFECTED.

REVISION 1 OF THE DESIGN CRITERIA AND SAFETY ANALYSIS INCORPORATE AND RESOLVE PRE-PORC COMMENTS TO THE REVISION 0 DOCUMENTS ORIGINALLY ISSUED.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND THE EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70. THE EVENTS RELATED TO THIS MODIFICATION ARE FIRES AND SEISMIC EVENTS.

THE PROPOSED MODIFICATION NEITHER PENETRATES ANY EXISTING FIRE BARRIERS NOR DOES IT AFFECT ANY EXISTING FIRE SUPPRESSION SYSTEMS. THUS, THIS MODIFICATION DOES NOT INCREASE ANY PREVIOUSLY DETERMINED FIRE LOADINGS, WHICH WILL BE VERIFIED BY APPENDIX R CONFORMANCE REVIEW.

ADDITIONALLY, THE MATERIALS USED IN CONSTRUCTION OF THE PIPE SUPPORTS WILL NOT ADD TO THE FIRE LOAD OF THE AREA BECAUSE THEY ARE NOT FLAMMABLE IN NATURE.

THE EFFECTS OF A SEISMIC EVENT HAVE BEEN ADDRESSED IN SECTION 4.0, 9.0 AND 13.0 OF THE DESIGN CRITERIA. AS DESCRIBED IN THE DESIGN CRITERIA, THE PIPING WILL BE TREATED AS SAFETY SIGNIFICANT AND WILL BE ANALYZED PER SEISMIC II/I CRITERIA. THIS WILL IMPROVE THE SUPPORT SCHEME OF THE CONDENSATE STORAGE PIPING SECTION BEING MODIFIED AND PROTECT THE SERVICE WATER LINE DURING A SAFE SHUTDOWN EARTHQUAKE EVENT, THUS ELIMINATING ANY POTENTIAL IMPACT ON THE PERFORMANCE OF THE SERVICE WATER LINE.

THUS, THIS MODIFICATION NEITHER INCREASES THE CONSEQUENCES, NOR DOES IT REDUCE THE MARGINS OF SAFETY FOR:

1) FIRE PROTECTION FEATURES

2) EQUIPMENT REQUIRED TO EFFECT A SAFETY SHUTDOWN OF THE PLANT DURING AND FOLLOWING SEISMIC EVENTS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT WILL NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

EWR-5097
CONTAINMENT RECIRCULATION UNIT 1D EQUIPMENT REMOVAL

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE INSTALLATION OF STRUCTURAL STEEL RIGGING AND REWORK OF THE RECIRCULATION UNIT 1D PLENUM TO PROVIDE A MEANS OF REMOVING THE FAN MOTOR AND HEAT EXCHANGER FOR MAINTENANCE, INSPECTION, OR REPLACEMENT.

A REVIEW HAS BEEN PERFORMED OF ALL EVENTS ANALYZED IN THE GINNA STATION UPDATED FINAL SAFETY ANALYSIS REPORT. THE EVENTS RELATED TO THIS MODIFICATION ARE:

1) FIRES
2) SEISMIC EVENTS
3) LOSS OF COOLANT ACCIDENT
4) MAIN STEAM LINE BREAK

THE MODIFICATION WILL NOT ALTER THE AREA FIRE LOADING, THE SOURCES OF FIRE INITIATION, NOR THE ACCEPTABILITY OF THE CONSEQUENCES OF A FIRE SINCE THE MATERIAL USED WILL MEET THE FLAME TEST REQUIREMENTS OF IEEE 383-1974, AND WHENEVER PRACTICAL NON-COMBUSTIBLE AND HEAT-RESISTANT MATERIALS WILL BE USED.

THE MODIFICATION SHALL BE DESIGNED TO WITHSTAND AN OPERATIONAL BASIS AND SAFE SHUTDOWN EARTHQUAKE.

THE MODIFICATION WILL NOT AFFECT THE FUNCTIONAL PERFORMANCE OF THE CONTAINMENT RECIRCULATION FANS REQUIRED FOR THE MITIGATION OF A LOCA BECAUSE 1) THE MAXIMUM AIR FLOW THROUGH THE FAN CANNOT EXCEED THE FLOW RATE USED IN THE LOCA ACCIDENT ANALYSIS, SINCE THE PROPOSED MODIFICATION COULD ONLY REDUCE AIR FLOW DUE TO THE ADDED HARDWARE IN THE AIR FLOW PATH, AND, 2) THE MAXIMUM SERVICE WATER FLOW THROUGH THE FAN COOLER CAN NOT EXCEED THE FLOW RATE USED IN THE LOCA ACCIDENT ANALYSIS, SINCE THE PROPOSED MODIFICATION WILL EITHER MAINTAIN THE CURRENT FLOW CONFIGURATION OR ADD FLOW RESTRICTIONS.

THE MODIFICATION WILL NOT AFFECT THE FUNCTIONAL PERFORMANCE OF THE CONTAINMENT RECIRCULATION FANS REQUIRED FOR THE MITIGATION OF A MSLB BECAUSE 1) PT-37.5 "CONTAINMENT RECIRCULATION FANS MASS AIR FLOW CHECK" SHALL BE PERFORMED BOTH BEFORE AND AFTER THE MODIFICATION WORK TO ENSURE THE FLOW THROUGH THE FAN IS MAINTAINED ABOVE THE FLOW RATE USED IN THE MSLB ACCIDENT ANALYSIS, AND, 2) A FLOW CHECK WILL BE PERFORMED TO VERIFY THE FLOW IS ABOVE THE MINIMUM VALUE USED IN THE MSLB ACCIDENT ANALYSIS, IF A FLOW RESISTANCE IS ADDED TO THE SERVICE WATER LINE.

BASED ON THE PREVIOUS ANALYSIS:

1) THE MARGINS OF SAFETY DURING NORMAL OPERATION AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

2) THE STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION FOR THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

3) THE PROPOSED MODIFICATION DOES NOT INVOLVE A CHANGE IN THE GINNA TECHNICAL SPECIFICATIONS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-5097B</u>
<u>CONTAINMENT RECIRCULATION UNIT 1A EQUIPMENT REMOVAL</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE INSTALLATION OF STRUCTURAL STEEL RIGGING AND REWORK OF THE RECIRCULATION UNIT 1A PLENUM TO PROVIDE A MEANS OF REMOVING THE FAN MOTOR AND HEAT EXCHANGER FOR MAINTENANCE, INSPECTION, OR REPLACEMENT.

A REVIEW HAS BEEN PERFORMED OF ALL EVENTS ANALYZED IN THE GINNA STATION UPDATED FINAL SAFETY ANALYSIS REPORT. THE EVENTS RELATED TO THIS MODIFICATION ARE:

1) FIRES
2) SEISMIC EVENTS
3) LOSS OF COOLANT ACCIDENT
4) MAIN STEAM LINE BREAK

THE MODIFICATION WILL NOT ALTER THE AREA FIRE LOADING, THE SOURCES OF FIRE INITIATION, NOR THE ACCEPTABILITY OF THE CONSEQUENCES OF A FIRE SINCE, NON-COMBUSTIBLE AND HEAT-RESISTANT MATERIALS WILL BE USED, WHENEVER PRACTICAL. THE MODIFICATION WILL NOT DEGRADE EXISTING FIRE BARRIERS OR AFFECT THE PERFORMANCE OF ANY EXISTING FIRE PROTECTION EQUIPMENT, AND AN APPENDIX R CONFORMANCE VERIFICATION SHALL BE PREPARED TO DETERMINE IF THE PROPOSED MODIFICATION IMPACTS APPENDIX R COMPLIANCE.

THE MODIFICATION SHALL BE DESIGNED TO WITHSTAND AN OPERATIONAL BASIS AND SAFE SHUTDOWN EARTHQUAKE.

THE MODIFICATION WILL NOT AFFECT THE FUNCTIONAL PERFORMANCE OF THE CONTAINMENT RECIRCULATION FANS REQUIRED FOR THE MITIGATION OF A LOCA BECAUSE THE MAXIMUM AIR FLOW THROUGH THE FAN SHALL NOT EXCEED THE FLOW RATE USED IN THE LOCA ACCIDENT ANALYSIS, SINCE THE PROPOSED MODIFICATION COULD ONLY REDUCE AIR FLOW DUE TO THE ADDED HARDWARE IN THE AIR FLOW PATH.

THE MODIFICATION WILL NOT AFFECT THE FUNCTIONAL PERFORMANCE OF THE CONTAINMENT RECIRCULATION FANS REQUIRED FOR THE MITIGATION OF A MSLB. PT-37.5 "CONTAINMENT RECIRCULATION FANS MASS AIR FLOW CHECK" SHALL BE PERFORMED BOTH BEFORE AND AFTER THE MODIFICATION WORK TO ENSURE THE FLOW THROUGH THE FAN IS MAINTAINED ABOVE THE FLOW RATE USED IN THE MSLB ACCIDENT ANALYSIS.

BASED ON THE PREVIOUS ANALYSIS:

1) THE MARGINS OF SAFETY DURING NORMAL OPERATION AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE STATION ARE NOT REDUCED.

2) THE STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION FOR THE CONSEQUENCES OF ACCIDENTS ARE ADEQUATE.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

EWR-5114
SEISMIC AND HIGH ENERGY SUPPORTS

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE NEED TO DOCUMENT SUPPORTS WHICH ARE: 1) INCLUDED IN THE SEISMIC UPGRADE PROGRAM (SUP) BUT ARE ALSO BEYOND THE ASME SAFETY CLASS BOUNDARY; 2) IDENTIFICATION OF "HIGH ENERGY" AND II/I SUPPORTS; 3) INCORPORATE MARKING OF HIGH ENERGY BOUNDARIES ON GINNA P&ID DRAWINGS, AND 4) MODIFICATION OF HIGH ENERGY SUPPORTS REQUIRED AS A RESULT OF THE HIGH ENERGY ISI PROGRAM.

THE PIPING SYSTEMS OUTSIDE CONTAINMENT CONSIDERED UNDER THIS EWR ENCOMPASS BOTH MAIN STEAM AND FEEDWATER.

A REVIEW HAS BEEN MADE OF ALL EVENTS ANALYZED IN THE GINNA STATION UFSAR AND THE EVENTS REQUIRING ANALYSIS BY USNRC REG. GUIDE 1.70. THE EVENTS ASSOCIATED WITH THIS MODIFICATION ARE:

A) FIRE
B) SEISMIC EVENTS
C) INCREASE IN HEAT REMOVAL BY SECONDARY SYSTEM
D) DECREASE IN HEAT REMOVAL BY SECONDARY SYSTEM
E) ANTICIPATED TRANSIENTS WITHOUT SCRAM
F) PIPE BREAKS OUTSIDE CONTAINMENT
G) FLOODING

ALL MATERIALS USED IN THE MODIFICATION SHALL NOT INCREASE THE PROBABILITY OR CONSEQUENCE OF A FIRE, AND WILL NOT AFFECT THE PERFORMANCE OF ANY EXISTING FIRE PROTECTION EQUIPMENT. ADDITIONALLY, THE MODIFICATIONS UNDER THIS EWR WILL BE REVIEWED AGAINST THE ASSUMPTIONS OF 10CFR50 APPENDIX R TO SHOW CONTINUED COMPLIANCE.

THE DESIGN OF ALL HIGH ENERGY SUPPORTS SHALL BE COMPATIBLE WITH THE ORIGINAL DESIGN BASES, SUCH THAT THE HIGH ENERGY PIPING AND SUPPORTS DO NOT IMPACT SAFETY-RELATED EQUIPMENT.

AN INCREASE/DECREASE IN THE HEAT REMOVAL BY THE SECONDARY SYSTEM WILL NOT BE CREATED, SINCE ALL MODIFICATIONS TO HIGH ENERGY PIPING SYSTEMS AND SUPPORTS SHALL BE PERFORMED IN ACCORDANCE WITH APPLICABLE CODE DESIGN, INSPECTION, AND TESTING REQUIREMENTS.

OCCURRENCE OF ANTICIPATED TRANSIENTS WITHOUT SCRAM'S ARE UNAFFECTED SINCE MODIFICATIONS WILL NEITHER AFFECT OR IMPACT ACTIVE COMPONENTS OR SYSTEM OPERATIONS, CONTROL SYSTEMS, OR RODS NOR AFFECT SAFETY RELATED COMPONENTS REQUIRED FOR SAFE SHUTDOWN.

THE PROBABILITY FOR PIPE BREAKS OUTSIDE CONTAINMENT AND FLOODING WILL NOT BE INCREASED AS MODIFICATIONS TO HIGH ENERGY PIPE SUPPORTS SHALL BE PERFORMED IN ACCORDANCE WITH APPLICABLE CODE DESIGN, INSTALLATION, INSPECTION, AND TESTING REQUIREMENTS.

THUS, THIS MODIFICATION NEITHER INCREASES THE CONSEQUENCES, NOR DOES IT REDUCE THE MARGINS OF SAFETY FOR:

1) EQUIPMENT REQUIRED TO FUNCTION DURING AND FOLLOWING SEISMIC AND FLOODING EVENTS

2) FIRE PROTECTION FEATURES

3) EQUIPMENT REQUIRED TO EFFECT A SAFE SHUTDOWN OF THE PLANT INCLUDING VARIOUS SCENARIOS FOR HEAT REMOVAL BY THE SECONDARY PLANT AND REACTOR SHUTDOWN.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

<u>EWR-10073</u>
<u>TURBINE RUNBACK DELETION</u>

THIS ENGINEERING WORK REQUEST (EWR) ADDRESSES THE DELETION OF THE DROPPED ROD TURBINE RUNBACK FEATURE AT GINNA STATION.

GINNA STATION HAS EXPERIENCED A NUMBER OF TURBINE RUNBACK ACTUATIONS DUE TO SPURIOUS DROPPED ROD SIGNALS FROM THE NUCLEAR INSTRUMENTATION SYSTEM (NIS) AND ROD POSITION INDICATION (RPI) SYSTEM. THIS RESULTS IN AN UNNECESSARY 20% POWER CUTBACK AS WELL AS CHALLENGES TO PLANT MECHANICAL AND ELECTRICAL SYSTEMS. RECENT ANALYSES BY WESTINGHOUSE HAVE SHOWN THAT THE PLANT'S DNB (DEPARTURE FROM NUCLEATE BOILING) DESIGN BASIS CAN BE MET DURING THE COURSE OF A DROPPED ROD EVENT WITH NO TURBINE RUNBACK. THE METHODOLOGIES FOR PERFORMING THE PLANT SPECIFIC ANALYSES AND IMPLEMENTING THE REQUIRED CHANGES ARE WCAP-11394-P-A AND WCAP-12282 RESPECTIVELY. THE GINNA SPECIFIC ANALYSIS WHICH JUSTIFIES REMOVAL OF THE DROPPED ROD TURBINE RUNBACK LOGIC IS CONTAINED IN THE CYCLE 23 RELOAD SAFETY EVALUATION FOR THE R.E. GINNA NUCLEAR POWER PLANT DATED JANUARY 1993.

TURBINE RUNBACK AS A RESULT OF A DROPPED ROD SENSED BY ONE OUT OF FOUR NIS CHANNELS OR ONE OUT OF 29 RPI CHANNELS WILL BE DELETED AT GINNA STATION. THIS MODIFICATION WILL <u>NOT</u> AFFECT OTHER CONTROL FUNCTIONS WHICH MAY SHARE PORTIONS OF THE EXISTING DROPPED ROD TURBINE RUNBACK LOGIC. CONTROL FUNCTIONS <u>NOT</u> AFFECTED ARE:

- TURBINE RUNBACK DUE TO ONE OUT OF FOUR OVERTEMPERATURE HIGH DELTA T CHANNELS OR ONE OUT OF FOUR OVERPOWER HIGH DELTA T CHANNELS.

- BLOCK AUTO ROD WITHDRAWAL DUE TO DROPPED ROD SENSED BY ONE OUT OF FOUR NIS CHANNELS OR ONE OUT 29 RPI CHANNELS.

A REVIEW HAS BEEN MADE OF THE DESIGN BASIS EVENTS TO DETERMINE THOSE RELATED TO THE MODIFICATION. THE EVENTS ASSOCIATED WITH THIS WORK ARE: SEISMIC EVENT, FIRES, AND DROPPED ROD.

THE MODIFICATION INVOLVES WIRING AND LABEL CHANGES ONLY; NO COMPONENTS WILL BE ADDED OR REMOVED. WIRING AND LABEL CHANGES HAVE NO IMPACT ON THE SEISMIC QUALIFICATION BECAUSE ALL MODIFIED WIRES ARE REQUIRED TO BE INSULATED AND SECURED PER EE-29; NO WIRE ENDS WILL BE LEFT UNSECURED.

THE MODIFICATION WILL NOT ADD ANY NEW COMPONENTS OR ELECTRICAL WIRING. ONLY WIRE LABELS WILL CHANGE. CHANGING LABELS DOES NOT IMPACT FIRES.

THE METHODOLOGY FOR DETERMINING THE EFFECT OF A DROPPED ROD WITHOUT TURBINE RUNBACK ON CORE PEAKING FACTORS HAS BEEN EVALUATED IN WCAP-11394-P-A, METHODOLOGY FOR ANALYSIS OF DROPPED ROD EVENT. THIS EVALUATION HAS BEEN REVIEWED AND APPROVED BY THE NRC. THE WCAP-11394-P-A CONCLUDES:

> "THE WESTINGHOUSE ANALYSIS, RESULTS AND COMPARISONS ARE REACTOR AND CYCLE SPECIFIC. NO CREDIT IS TAKEN FOR ANY DIRECT REACTOR TRIP DUE TO DROPPED RCCA(S). ALSO, THE ANALYSIS ASSUMES NO AUTOMATIC POWER REDUCTION FEATURES ARE ACTUATED BY THE DROPPED RCCA(S). A FURTHER REVIEW BY THE STAFF (FOR EACH CYCLE) IS NOT NECESSARY, GIVEN THE UTILITY ASSERTION THAT THE ANALYSIS DESCRIBED BY WESTINGHOUSE HAS BEEN PERFORMED AND THE REQUIRED COMPARISONS HAVE BEEN MADE WITH FAVORABLE RESULTS".

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

SECTION B - COMPLETED STATION MODIFICATIONS (SMs)

This section contains a description of station modification procedures performed in the facility as described in the safety analysis report. Station modification procedures are written to complete a portion of an Engineering Work Request (EWR) or Technical Staff Request (TSR) identified by the same parent number. Station Modifications are reviewed by the Plant Operations Review Committee to ensure that no unreviewed safety questions or Technical Specifications changes are involved with the procedure.

The basis for inclusion of an SM in this section is closure of the SM where portions of the parent EWR or TSR, in the form of other SMs or other documentation, remain to be completed.

SM-92-002.1
CVCS SYSTEM - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSR 91-140.

SM-92-002.2
SAFW SYSTEM - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSR 90-197.

SM-92-002.3
PIPING MODIFICATION TO CCW SUPPLY LINE TO SI PUMP C SEAL HEAT EXCHANGER B

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE MODIFICATION TO THE CCW SUPPLY LINE TO SI PUMP C SEAL HEAT EXCHANGER B.

SM-92-002.4
SAFETY INJECTION SYSTEM - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENTS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSRS 91-183 AND 89-047.

SM-92-002.5
SERVICE WATER SYSTEM (GROUP A) - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSRS 90-004 AND 92-002.

SM-92-002.6
SERVICE WATER SYSTEM (GROUP B) - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSR 92-002.

SM-92-002.7
SERVICE WATER SYSTEM (GROUP C) - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSRS 91-183 AND 92-002.

SM-92-002.8
SERVICE WATER SYSTEM (GROUP D) - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSR 92-002.

SM-92-002.9
AUXILIARY FEEDWATER SYSTEM - 1993 REFUELING OUTAGE VALVE INSPECTIONS, UPGRADES, AND REFURBISHMENT

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE INSPECTIONS, UPGRADES, AND REFURBISHMENTS TO THE VALVES AND/OR VALVE OPERATORS IN ACCORDANCE WITH TSR 91-009.

SM-1594.18
SPENT FUEL POOL COOLING SYSTEM - FIT 8667 AND LAL-661 REPLACEMENT WITH ANNUNCIATOR WINDOW ALARM RELOCATION

THE PURPOSE OF THIS REVISED PROCEDURE IS TO CONTROL THE INSTALLATION TESTING AND TURNOVER FOR THE NEW DIFFERENTIAL PRESSURE SWITCH (FIT-8667) AND LEVEL SWITCH LAL-661. THIS PROCEDURE WAS REVISED TO INCORPORATE THE NEW NPS WORK PROCEDURES.

SM-4230.4

AMSAC POWER LEVEL TIME DELAY LOCK-IN FEATURE MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE AMSAC POWER LEVEL TIME DELAY LOCK-IN FEATURE MODIFICATION. THIS POWER LEVEL LOCK-IN FEATURE WILL "LATCH" THE TIMING VALUE OF THE VARIABLE TIMER, FOR THAT POWER, AT THE MOMENT THAT THE ATWS EVENT ACTUATED.

SM-4324.18

"B" STEAM GENERATOR BLOWDOWN ISOLATION VALVES REPLACEMENT

THE PURPOSE OF THIS NEW PROCEDURE IS TO ALLOW REPLACEMENT OF "B" S/G BLOWDOWN ISOLATION VALVES AND SAMPLE VALVES.

SM-4324.19

S/G BLOWDOWN SYSTEM PIPE SUPPORT MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE S/G BLOWDOWN SYSTEM PIPE SUPPORT MODIFICATION.

SM-4534B.1

"A" RCP OIL LEVEL MONITORING INSTRUMENT MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "A" RCP OIL LEVEL MONITORING INSTRUMENT MODIFICATION.

SM-4607.2

480 V. ELECTRICAL DISTRIBUTION SYSTEM MOTOR CONTROL CENTER A CONDUIT L130 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE 480 V. ELECTRICAL DISTRIBUTION SYSTEM MOTOR CONTROL CENTER 1A CONDUIT L130 RELOCATION.

SM-4607.4

CONDENSER UPGRADE MECHANICAL INTERFERENCE MODIFICATIONS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES PERTAINING TO MECHANICAL INTERFERENCES ASSOCIATED WITH THE CONDENSER UPGRADE MODIFICATION.

SM-4607.6
CONDENSER UPGRADE STRUCTURAL INTERFERENCE MODIFICATIONS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL AND TURNOVER ACTIVITIES PERTAINING TO THE STRUCTURAL INTERFERENCES ASSOCIATED WITH THE CONDENSER UPGRADE MODIFICATION.

SM-4618.5
MAIN FEEDWATER PUMP ROOM HVAC TEMPERATURE CONTROL SYSTEM MODIFICATION

THE PURPOSE OF THIS NEW PROCEDURE IS TO CONTROL THE INSTALLATION TESTING AND TURNOVER OF THE MAIN FEEDWATER PUMP ROOM HVAC TEMPERATURE CONTROL SYSTEM MODIFICATION.

SM-4658C.1
"A" AFW PUMP DRAIN VALVE 4343 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "A" AFW PUMP DRAIN VALVE 4343 MODIFICATION.

SM-4658C.2
"B" AFW PUMP DRAIN VALVE 4342 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "B" AFW PUMP DRAIN VALVE 4342 MODIFICATION.

SM-4658C.3
TDAFW PUMP DRAIN VALVE 4358A MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE TDAFW PUMP DRAIN VALVE 4358A MODIFICATION.

SM-4658C.4
"C" SAFW PUMP DRAIN VALVE 9716A MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "C" SAFW PUMP DRAIN VALVE 9716A MODIFICATION.

SM-4658C.5
"D" SAFW PUMP DRAIN VALVE 9716B MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "D" SAFW PUMP DRAIN VALVE 9716B MODIFICATION.

SM-4658C.6
"A" AND "B" COMPONENT COOLING WATER HEAT EXCHANGER TEMPERATURE INDICATORS TE-6871, TE-6872, TE-6873, AND TE-6874 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "A" AND "B" COMPONENT COOLING WATER HEAT EXCHANGER TEMPERATURE INDICATORS TE-6871, TE-6872, TE-6873, TE-6874 MODIFICATION.

SM-4658C.7
"A" AND "B" CONTAINMENT RECIRCULATION FAN MOTOR COOLER TEMPERATURE INDICATORS TE-6875 AND TE-6876 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "A" AND "B" CONTAINMENT RECIRCULATION FAN MOTOR COOLER TEMPERATURE INDICATORS TE-6875, AND TE-6876 MODIFICATION.

SM-4658C.8
"C" AND "D" CONTAINMENT RECIRCULATION FAN MOTOR COOLER TEMPERATURE INDICATORS TE-6877 AND TE-6878 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "C" AND "D" CONTAINMENT RECIRCULATION FAN MOTOR COOLER TEMPERATURE INDICATORS TE-6877, AND TE-6878 MODIFICATION.

SM-4755B.1
RWST VACUUM BREAKER V-2850 AND V-2851 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL AND TURNOVER ACTIVITIES ASSOCIATED WITH THE RWST VACUUM BREAKER V-2850 AND V-2851 MODIFICATION.

SM-4755B.2
NAOH SPRAY ADDITIVE TANK VACUUM BREAKER V-845C AND V-845D MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE NAOH SPRAY ADDITIVE TANK VACUUM BREAKER V-845C AND V-845D MODIFICATION.

SM-4755B.3
COMPONENT COOLING WATER SURGE TANK VACUUM BREAKER V-651 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE CCW SURGE TANK VACUUM BREAKER V-651 MODIFICATION.

SM-4882.2
TURBINE BUILDING FIRE DAMPERS REPLACEMENT MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE TURBINE BUILDING FIRE DAMPERS, UPGRADE OF FIRE BARRIER PENETRATION SEAL.

SM-4932.1
INTERMEDIATE BUILDING HVAC - DOOR S36 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING AND TURNOVER OF THE NEW ROLLING FIRE BARRIER DOOR AND SECURITY BARRIER ACROSS THE EXISTING DOOR S36 OPENING.

SM-4941.1
FIRE BARRIER PENETRATION SEALS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION AND TURNOVER OF FIREPROOFING MATERIALS FOR THE SEALING OF VARIOUS FIRE BARRIER PENETRATIONS.

SM-4970.2
RHR REDUNDANT FLOW TRANSMITTER FT-689 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE MECHANICAL INSTALLATION ACTIVITIES ASSOCIATED WITH THE RHR FLOW TRANSMITTER FT-689 MODIFICATION.

SM-4970.3
RHR FLOW TRANSMITTER FT-689 AND FLOW INDICATOR FI-626/689 MODIFICATION ACCEPTANCE TESTING

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE ELECTRICAL AND MECHANICAL TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE RHR REDUNDANT FLOW TRANSMITTER FT-689 AND FLOW INDICATOR FI-626/689 MODIFICATIONS.

SM-5098.2
SERVICE WATER VALVE REPLACEMENT - MOV 4609 AND MOV 4780

THE PURPOSE OF THIS PROCEDURE IS REPLACE MOV-4609 AND MOV-4780.

SM-5098.3
TURBINE BUILDING SERVICE WATER ISOLATION MOV 4614 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE REPLACEMENT OF MOV 4614.

SM-5098.4
AIR CONDITIONING SERVICE WATER ISOLATION MOV 4733 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE REPLACEMENT OF MOV 4733.

SM-5167C.1
CONTAINMENT REACTOR CAVITY PLATFORM SAFETY MODIFICATIONS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE CONTAINMENT REACTOR CAVITY PLATFORM SAFETY MODIFICATIONS.

SM-5168.26
CHLORINE INJECTION PIPING MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE MODIFICATION AND TURNOVER OF THE CHLORINE INJECTION PIPING DUE TO FLOW BLOCKAGE CONCERNS.

SM-5275.1
UNITS "A" & "B" CONTAINMENT RECIRCULATION FAN COOLER REPLACEMENTS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION AND TURNOVER OF THE "A" AND "B" CONTAINMENT RECIRCULATION FAN COOLERS.

SM-5275.2
UNITS "C" & "D" CONTAINMENT RECIRCULATION FAN COOLER REPLACEMENTS

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE REMOVAL, INSTALLATION AND TURNOVER OF THE "C" & "D" CONTAINMENT RECIRCULATION FAN COOLERS.

SM-5275.5
SERVICE WATER SYSTEM FLOW BALANCE TEST

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE TESTING OF THE SERVICE WATER SYSTEM FLOW BALANCE TEST.

SM-5275.6
CONTAINMENT RECIRCULATION FAN AIR FLOW TEST

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE TESTING OF THE CONTAINMENT RECIRCULATION FAN AIR FLOW TEST.

SM-5275.7
CONTAINMENT RECIRCULATION FAN COOLERS THERMAL PERFORMANCE TEST

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE TESTING OF THE CONTAINMENT RECIRCULATION FAN COOLERS THERMAL PERFORMANCE TEST.

SM-5282C.1
REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2 & PS-4 FOR BUS 14 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2 AND PS-4 FOR BUS 14 MODIFICATION.

SM-5282C.2
REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2, & PS-4 FOR BUS 18 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2 AND PS-4 FOR BUS 18 MODIFICATION.

SM-5282C.3
REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2, AND PS-4 FOR BUS 16 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2 AND PS-4 FUR BUS 16 MODIFICATION.

SM-5282C.4

REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2, AND PS-4 FOR BUS 17 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE REPLACEMENT OF UVPS POWER SUPPLIES PS-1, PS-2 AND PS-4 FOR BUS 17 MODIFICATION.

SM-5284.1

"A" AND "B" COMPONENT COOLING WATER PUMP CHECK VALVE 723A AND 723B MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "A" AND "B" COMPONENT COOLING WATER PUMP CHECK VALVE 723A AND 723B MODIFICATION.

SM-5284B.1

"A" AND "B" SERVICE WATER PUMP CHECK VALVE 4601 AND 4602 MODIFICATION

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER ACTIVITIES ASSOCIATED WITH THE "A" AND "B" SERVICE WATER PUMP CHECK VALVE 4601 AND 4602 MODIFICATION.

SM-5393.3

CONTAINMENT RECIRCULATION FAN MOTOR COOLERS PERFORMANCE TEST

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE TESTING OF THE FOR NEW CONTAINMENT RECIRCULATION FAN MOTOR COOLERS.

SM-5411.1

FIRE SERVICE WATER PIPING TIE-IN - AUXILIARY BUILDING

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE FIRE SERVICE WATER PIPING TIE-IN IN THE AUXILIARY BUILDING, UPSTREAM OF FIRE SYSTEM S-36.

SM-5424.1

RECORDER REPLACEMENT (PHASE I)

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING AND TURNOVER OF THE REPLACEMENT OR REMOVAL OF RECORDERS IN THE CONTROL ROOM.

SM-10069.1
P.A.S.S. WASTE TRANSFER PUMP

THE PURPOSE OF THIS PROCEDURE IS TO CONTROL THE INSTALLATION, TESTING, AND TURNOVER OF THE NEW P.A.S.S. WASTE TRANSFER PUMP.

SECTION C - COMPLETED TECHNICAL EVALUATIONS (TEs)

This section contains a description of changes to the facility as described in the safety analysis report performed as technical evaluations. These are typically changes of low safety significance that do not require the full controls of a modification. Technical Evaluations are reviewed by the Plant Operations Review Committee to ensure that no unreviewed safety questions or Technical Specification changes are involved.

TSR 93-137
ROD CONTROL SETTINGS

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF ADJUSTING THE LEAD TIME CONSTANTS OF THE ROD CONTROL SYSTEM.

THE INSTALLATION OF EQUIPMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS EVALUATION.

TSR 93-031
INSTRUMENT AIR SYSTEM SPARE VALVE ADDITION

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF A SPARE CONNECTION TO THE EXISTING INSTRUMENT AIR SYSTEM. THIS CONNECTION WILL GENERALLY CONSIST OF A VALVE AND MISCELLANEOUS PIPE FITTINGS OF A PRESSURE AND TEMPERATURE SUITABLE TO THE INSTRUMENT AIR SYSTEM. THIS ANALYSIS WILL ALSO CONSIDER THE SAFETY CONSEQUENCES OF THE CONNECTION OF AN EXTERNAL AIR SUPPLY TO THE SYSTEM.

THE INSTALLATION OF EQUIPMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS EVALUATION.

TSR 93-096
GLAND SEALING WATER PIPING FLEXIBLE CONNECTION

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF THE INSTALLATION OF A FLEXIBLE CONNECTION BETWEEN THE GLAND SEALING WATER SYSTEM PIPING AND ANY NON-SAFETY CONDENSATE SYSTEM VALVE. THIS CONNECTION WILL GENERALLY CONSIST OF MISCELLANEOUS PIPE FITTINGS OF A PRESSURE AND TEMPERATURE SUITABLE TO THE GLAND SEALING WATER SYSTEM.

THE INSTALLATION OF EQUIPMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS EVALUATION.

TSR 93-179
GENERIC BLANK FLANGE ADDITION FOR TEMPORARY MODIFICATIONS

THIS SAFETY EVALUATION COVERS THE ADDITION OF BLANK FLANGES TO ISOLATE SELECTED PLANT EQUIPMENT. IT PROVIDES GUIDELINES THAT MUST BE MET VIA A DOCUMENTED ENGINEERING EVALUATION TO BE WITHIN THE SCOPE OF THIS SAFETY ANALYSIS. IT IS TO BE USED ADMINISTRATIVELY IN CONJUNCTION WITH THE TEMPORARY MODIFICATION PROCESS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 91-209
CREVICE CLEANING TUBING REMOVAL

THIS EVALUATION COVERS REMOVAL OF TUBING IN THE TURBINE BUILDING THAT IS USED DURING CREVICE CLEANING OPERATIONS. THE TUBING IS NORMALLY ISOLATED AND TAPS OFF THE MAIN STEAM HEADER SAMPLE LINES (NON-NUCLEAR SAFETY PORTION). TWO RUNS OF 3/8" TUBING WILL BE REMOVED AS WELL AS TWO PRESSURE GAUGES. THE CONNECTIONS TO THE SAMPLE SYSTEM WILL THEN BE CAPPED.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 93-142
SERVICE WATER VALVE INSTALLATION

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF INSTALLING AN ADDITIONAL ISOLATION VALVE TO THE SERVICE WATER CHLORINE INJECTION SYSTEM.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 93-123
INSTALLATION OF DOMESTIC WATER HOSE STATIONS IN THE INTERMEDIATE BUILDING CLEAN SIDE

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF INSTALLING HOSE STATIONS IN THE INTERMEDIATE BUILDING CLEAN SIDE.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 93-169
LAUNDRY ROOM EQUIPMENT REMOVAL

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF THE REMOVAL OF THE NON-SAFETY CLASS LAUNDRY ROOM WASHER AND DRYER AND MISC. ASSOCIATED COMPONENTS.

THE INSTALLATION OF EQUIPMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 92-066
CONDENSATE DEMINERALIZER SAMPLE SINK HOLDING TANK C PIPING REROUTE

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF ALTERING PIPING FOR NON-SAFETY CLASS CONDUCTIVITY ELEMENT AE-4261 AND THE CNDST DI SAMPLE SINK DRAIN TANK C, REMOVING THE DRAIN TANK AND/OR PUMP AND/OR MISC ATTACHED PIPING AND DE-TERMING ASSOCIATED PUMP ELECTRICAL CONNECTIONS TO PREVENT DRAIN TANK SPILLAGE THAT FREQUENTLY OCCURS.

THE INSTALLATION/REMOVAL OF EQUIPMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 92-224

AIR COMPRESSOR AFTERCOOLER PIPING CHANGE

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF CHANGING THE SIZE OF THE SERVICE WATER PIPING AND CONFIGURATION OF THE AIR PIPING ON THE INLET AND OUTLET CONNECTIONS ON THE NON-SAFETY CLASS INSTRUMENT AIR AND SERVICE AIR AFTERCOOLERS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS HAVE NOT BEEN AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR 94-012

AIR COMPRESSOR TAH INSTRUMENT REPLACEMENT

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF 1) RELOCATING THE IA AND SA COMPRESSORS DISCHARGE AIR TEMPERATURE INDICATORS TO A LOCATION DOWNSTREAM OF THEIR PRESENT LOCATION AND 2) TO TEMPORARILY REMOVE TAH-2005A FROM SERVICE TO ACCOUNT FOR INTERFERENCES BETWEEN THE ORIGINAL MOUNTING LOCATIONS AND CURRENT INSTRUMENTATION AND PARTS AVAILABILITY PER THE TECHNICAL EVALUATION.

THE INSTALLATION/REMOVAL OF EQUIPMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS WILL NOT BE AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR-93-050
AOV-6494 REPLACEMENT

THE PURPOSE OF THIS ANALYSIS IS TO EVALUATE THE SAFETY CONSEQUENCES OF REPLACING THE EXISTING 1" FISHER, TYPE 667ES SIZE 30 AOV WITH A 1 1/2" FISHER, TYPE 667ET SIZE 34 AOV IN THE HOUSE HEATING STEAM AND CONDENSATE SYSTEM.

THE AOV-6494 REPLACEMENT DESCRIBED ABOVE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION BASED UPON THE DISCUSSION CONTAINED WITHIN THE SAFETY ANALYSIS.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS WILL NOT BE AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

TSR-94-025
RELIEF VALVE ADDITION

THE SYSTEMS ENGINEER PRESENTED THIS SAFETY EVALUATION FOR THE ADDITION OF RELIEF VALVES WHICH WILL SERVE TO PROTECT THE 826 VALVES FROM OVERPRESSURIZATION IN THE EVENT THAT HEAT TRACING IS TURNED ON IN THE SECTION OF PIPING BETWEEN VALVES 826A/B AND 826C/D (CLOSED) AND THEREFORE THE RELIEF VALVES AND THE ASSOCIATED RELIEF PATH(S) ARE CONSIDERED SAFETY SIGNIFICANT.

THIS CHANGE HAS NO EFFECT ON THE ABILITY OF THE 826 VALVES TO MAINTAIN THE SAFETY INJECTION PRESSURE BOUNDARY, SINCE THE 826 VALVES ARE NOT BEING MODIFIED AND OVERPRESSURE PROTECTION IS BEING PROVIDED ON THE SAFETY SIGNIFICANT SIDE OF THESE VALVES.

THE ADDITION OF THE RELIEF VALVES DOES NOT INCREASE THE PROBABILITY OF AN OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY AS PREVIOUSLY EVALUATED IN THAT THE RELIEF VALVES SERVE A PIPING SECTION THAT PERFORMS NO SAFEGUARDS FUNCTIONS.

THE ADDITION OF THE RELIEF VALVES DOES NOT CREATE THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT IN THAT THE EFFECTS OF THEIR PLACEMENT ARE LIMITED TO NON SAFEGUARDS EQUIPMENT AND SYSTEMS.

THE ADDITION OF THE RELIEF VALVES DOES NOT REDUCE THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION IN THAT THE CHANGE IS TO NON-SAFEGUARDS EQUIPMENT AS DESCRIBED BY TECHNICAL SPECIFICATION AMENDMENT #57.

BASED UPON A REVIEW OF THE UFSAR AND THE REQUIREMENTS OF GINNA STATION TECHNICAL SPECIFICATIONS, IT HAS BEEN CONCLUDED THAT THE MARGINS OF SAFETY DURING NORMAL OPERATIONS AND TRANSIENT CONDITIONS ANTICIPATED DURING THE LIFE OF THE PLANT HAVE NOT BEEN REDUCED. IT HAS ALSO BEEN CONCLUDED THAT THE ADEQUACY OF STRUCTURES, SYSTEMS, AND COMPONENTS PROVIDED FOR THE PREVENTION OF ACCIDENTS AND THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS WILL NOT BE AFFECTED BY THE IMPLEMENTATION OF THIS MODIFICATION.

SECTION D - TEMPORARY MODIFICATIONS

This section contains safety evaluations of temporary changes to the facility pursuant to the requirements of 10 CFR 50.59(b).

SAFETY EVALUATION

Temporary Modification 94-011
EQ EQUIPMENT TEMPERATURE MONITORING

Revision 0

GINNA STATION
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

March 16, 1994

Prepared By:	Staff Engineer	3/21/94 Date
Reviewed By:	Reactor Engineer	3/23/94 Date
Reviewed By:	Manager, Technical Engineering	3/23/94 Date
Approved By:	Chairman, Plant Operations Review Committee	3/23/94 Date

1.0 <u>Scope of Analysis</u>

1.1 It has been recommended that the R.E. Ginna Station establish a Temperature Monitoring Program for EQ equipment. This program will include EQ equipment whose qualified life determination is based on actual temperature data that is reduced or elevated from the design basis containment atmosphere temperature of 120°F.

Currently, temperature at six locations throughout Containment is monitored and recorded on a strip chart recorder (ILRT Panel) and sent to the PPCS. In addition to these six EQ RTD's, eighteen spare terminal locations throughout Containment are available for installation of temporary RTD's to assist in data collection. Of these eighteen locations only eight will be utilized for this temporary modification, and will be required to collect data for one year.

1.2 The following EQ equipment will be monitored by the RTD's.

FT-3-1B	FT-474	V-590
FT-3-1A	FT-475	V-591
FT-464	FT-924	V-592
FT-465	FT-925	V-593

LT-426	LT-471	SOV 14115S
LT-427	LT-472	MOV 515
LT-428	LT-473	MOV 516
LT-461	LT-490A	
LT-462	LT-490B	
LT-463		

The temporary RTD's will be monitored using TELOG 2400 Series Data Recorders. These data points will be monitored in conjunction with the data from the six permanently installed RTD's. After one year, an analysis will be performed to correlate all the data from these RTD's.

1.3 This analysis addresses the consequences on installing temporary RTD's to existing structures in the plant near the EQ equipment to be monitored.

2.0 <u>Reference Documents</u>

2.1 Ginna Station Procedure , A-303, "Preparation, Review, and Approval of Safety Analysis"

2.2 Ginna Station Procedure, A-1406, "Control of Temporary Modifications"

2.3 R.E. Ginna Nuclear Power Plant Updated Final Safety Analysis Report

2.4 Letter, George Wrobel from Terry Dresbach, "ASCO Solenoid Valve 10024", January 7, 1993, 13N1-RR-L3976

2.5 Letter, Mike Farnan from Terry Dresbach, "Containment Temperature Monitoring for EQ Equipment", March 1, 1993, 13N-RR-L4029

2.6 Ginna Station Safety Analysis for Temporary Modification 93-028 "EQ Equipment Temperature Monitoring", PORC approved 4/20/93

2.7 Letter, Bruce Yager from Terry Dresbach, "Containment Temporary Monitoring for EQ Equipment", March 8, 1994, 13N1-RR-L4429

2.8 Update of R.E. Ginna Nuclear Station Fire Protection Evaluation, Report No. 02-0950-01340, Rev. 1, Oct. 1986.

3.0 Safety Analysis

3.1 A review has been made of all events analyzed in the Ginna Station UFSAR. The events related to this temporary modification are:

1) Seismic Event
2) Major and Minor Fires

3.1.1 All temporary instrument cable installed shall be routed to follow existing seismically mounted conduit and piping. No seismic impact is anticipated since the instrument cable weight is negligible compared to the pipe/insulation weight and conduit/insulation weight. All cables shall be installed in accordance with A-1406 using nylon wire wraps.

3.1.2 All temporary RTD assemblies shall be mounted to existing seismic supports in the area of the selected instruments to be monitored. The RTD assemblies shall be located within 3 feet of the instrument and mounted as close as possible to the base of the existing updated seismic support. The support shall be of considerable size with respect to the size of the RTD assembly, and no original plant installation instrument supports shall be used. No seismic impact is anticipated since the RTD assemblies weigh less then three (3) pounds.

3.1.3 This temporary modification will not propagate a major fire. Cables used for the RTD's are 3/c #16 stranded copper with XLPE insulation, aluminum mylar shield and drain wire, and hypalon jacket rated and qualified to IEEE-383 flame requirements as a minimum. The additional fire loading for each level in containment is as follows:

	BSMT	INTER	OPER
EXISTING FIRE LOAD BTU'S/SQ FT	22934	5720	3176
ADDITIONAL BY TEM. MOD BTU'S/SQ FT	22	3	22
TOTAL BTU'S/SQ FT	22956	5723	3198
TOTAL ALLOWED BTU'S/SQ FT	240,000	240,000	240,000

All Fire Loading on the three (3) levels in containment are well below the maximum permissible amount of 240,000 BTU's/sq ft per level.

3.1.5 This temporary modification does not effect the safe shutdown analysis in Appendix R submittal since there is no effect on separation of existing circuits, associated circuits, or fire area boundaries. Temporary cables shall not be routed in existing cable trays, raceways, or conduits.

3.1.5 This temporary modification will not effect the capabilities of the alternative shutdown system. Furthermore, none of the existing procedures for obtaining an alternative safe shutdown will be effected, This temporary modification therefore, complies with 10CFR50, Appendix R.

3.2 This temporary modification does not degrade the capability of any safety system to perform its function. The assumptions and conclusions of existing analyses are unchanged. No new types of events are postulated.

3.2.1 Therefore, it has been determined that the margins of safety during normal operations and transient conditions anticipated during the life of the station have not been effected. It has been determined that the adequacy of structures, systems, and components provided for the consequences of accidents have not been effected.

4.1 The proposed temporary modification would not increase the probability of occurrence of an accident evaluated previously in the UFSAR because the temporary equipment would not introduce any likely ignition sources that could start a fire and does not perform a safety function.

4.2 The proposed temporary modification would not increase the consequences of an accident previously evaluated in the UFSAR because:

1) All cables and RTD's will be anchored and supported so that it does not effect safety related equipment during a seismic event.

2) All cables are rated and qualified to IEEE-383 flame requirements as a minimum and located not to interfere with equipment required to maintain safe shutdown conditions, 10CFR50, Appendix R compliance.

4.3 The proposed temporary modification would not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR due to the anchoring and location of cables/RTD's away from this equipment.

4.4 The proposed temporary modification would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR due to anchoring and location of cables/RTD's away from this equipment.

4.5 The proposed temporary modification would not create the possibility of an accident of a different type then previously evaluated in the UFSAR because the installed cables/RTD's do not perform a safety function and do not interfere with any equipment that performs a safety function.

4.6 The proposed temporary modification would not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the UFSAR because the installed cables/RTD's do not perform a safety function and do not interfere with any equipment that performs a safety function.

4.7 The proposed temporary modification would not reduce any margin of safety as defined in the basis of any technical specification, because the installed cables/RTD's do not interfere with the function of any structure, system, or component in the technical specification.

4.8 The proposed temporary modification does not involve an unreviewed safety question or technical specification change.

8104.1149

SAFETY ANALYSIS

TM 94-012

Diesel Generator Cooler Alternate
Discharge Flow Path

Rev. 0

Ginna Station

Rochester Gas and Electric Corporation

89 East Ave

Rochester, New York 14649

March 11, 1994

Prepared by: ______________________ 3-12-94
Staff Engineer — Date

Reviewed by: ______________________ 3/12/94
Reactor Engineer — Date

Reviewed by: ______________________ 3/12/94
Manager, Technical Engineering — Date

Approved by: ______________________ 3/12/94
Chairman, Plant Operations Review Committee — Date

1.0 Scope of Analysis

1.1 The purpose of this analysis is to evaluate the safety consequences of installing a temporary alternate discharge flow path for the DG Coolers. This system alignment is only permitted with the reactor completely defueled, at which time, the one DG remaining in service may be considered "Available" (See Reference 2.6). A hose shall be connected to the Cooler outlet flush connections(valves 11503A and 11503B). In the event the "Available" DG must be started: normal Service Water supply to the cooler shall be in service, Operator action shall be required to open the respective cooler flush connection valves, and manually start the DG. An Operator shall be posted in the DG room while in this configuration.

2.0 References

2.1 Temporary Modification 94-012, Diesel Generator Cooler Alternate Discharge Flow Path, Rev 0.
2.2 Ginna UFSAR section 9.1.3.4.3.
2.3 Ginna P&ID 33013-1250 sht 1, Service Water Cooling Water
2.4 Ginna Technical Specifications sections 3.7.1.1 and 3.11.1.
2.5 Short Form EWR 10229 (Design Analysis DA-ME-94-028)
2.6 Letter dated 3/01/94, G. Wrobel to J. Wayland
2.7 Letter dated 3/03/94, J. Wayland to PORC

3.0 Structures, Systems, and Components Affected (SSC)

3.1 The Station Service Water, including Diesel Generator Lube Oil and Jacket Water cooling, is affected by this change.

4.0 Safety Functions of Affected SSCs

4.1 The function of the Station Service Water System is to provide cooling to various turbine plant loads as well as auxiliary reactor plant loads. The system supplies seal water to the circulating water pumps and the vacuum pumps, flushing water to the traveling screens and makeup water to the fire water storage tank via the fire booster pump. Service water is the normal supply to the standby auxiliary feedwater system and an alternate supply to the auxiliary feedwater system. All portions of the service water system serving safeguards equipment are designed Seismic Category I. All other portions of the service water system serving non-safety loads are designated as non-seismic and are capable of being isolated from the Seismic Category I portion.

4.2 The main function of the Emergency Diesel Generators is to supply the emergency source of power to the 480 Volt buses 14, 16, 17 and 18. Diesel Generator 1A can provide power to buses 14 and 18 while Diesel Generator 1B can provide power to buses 16 and 17.

5.0 Effects on Safety

5.1 Reference 2.5 evaluates the design capability and requirements of this temporary modification. Per the discussion of the analysis, there will not be a reduction to the safety margins for the Service Water or Emergency Diesel Generator systems since supplied cooling water will allow for Diesel operation within manufacturer specified requirements.

5.2 With the reactor in Cold Shutdown and completely defueled, the Technical Specification 3.11.4 requires "The spent fuel pool temperature shall be limited to 150 degrees F". Similarly, the Technical Specification 3.7.1 does not require a Emergency Diesel Generator to be operable; however, as an electrical backup to the normal power supply to the Spent Fuel Pool Cooling system an Emergency Diesel Generator shall be "Available" as defined in Reference 2.6.

5.3 The following Hazards were considered in the evaluation of this temporary modification: Internal flood and Pipe whip.In both classes, operator actions shall minimize the hazards; an operator shall be stationed in the Diesel Generator room while this temporary modification is in service and will be capable of shutting down the diesel and isolating Service Water supply to the diesel.

5.4 The reliability of the Emergency Diesel Generator system shall not be reduced by this temporary modification because the cooling requirements normally provided by Service Water system will be sufficiently supplied, as analyzed in Reference 2.5. The Service Water system can readily be isolated from the diesel generator; therefore, its reliability is also not reduced.

6.0 Unreviewed Safety Question Conclusions

6.1 The proposed modification will not increase the probability of occurrence of an accident previously evaluated in the UFSAR because no safety functions will be affected with the reactor in Cold Shutdown and defueled. An Emergency Diesel Generator is being maintained as "Available" as a prudent outage management requirement.

6.2 The proposed modification will not increase the consequences of an accident previously evaluated in the UFSAR because the Emergency Diesel Generator system is not required in the defueled configuration.

6.3 The proposed modification will not increase the probability of a malfunction of equipment important to safety previously evaluated in the UFSAR because adequate cooling requirements are provided to the diesel generators by this temporary modification as analyzed in Reference 2.5.

6.4 The proposed modification will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because adequate cooling requirements are provided to the diesel generators by this temporary modification as analyzed in Reference 2.5. Additionally, an operator shall be stationed in the diesel generator room while operating in the configuration described by this temporary modification and will be able to secure the diesel generator or isolate Service Water supply .

6.5 The proposed modification will not create the possibility for a accident of a different type than any evaluated in the UFSAR because the Emergency Diesel Generator system is not required in the defueled configuration.

6.6 The proposed modification will not create the possibility for a malfunction of equipment of a different type than any evaluated in the UFSAR because the Emergency Diesel Generator system is not required in the defueled configuration.

6.7 The proposed modification will not reduce any margin of safety as defined in the basis of any Technical Specification because the Emergency Diesel Generator system is not required by Technical Specifications while in Cold Shutdown with the reactor defueled. The required Service Water system functions will remain operable with this temporary modification.

7.0 Conclusion

Based on the above discussion the proposed modification does not present an unreviewed safety question.

ROCHESTER GAS AND ELECTRIC CORPORATION

89 EAST AVENUE

ROCHESTER, NEW YORK 14649

GINNA STATION

SAFETY ANALYSIS

FOR

TEMPORARY MODIFICATION

94-019

REVISION 0

May 19, 1994

MAIN STEAM SYSTEM TEMPORARY INSTRUMENT INSTALLATION

PREPARED BY: ______________________ 5/25/94
Responsible Staff Engineer — Date

REVIEWED BY: Peter Bamford 5/25/94
Reactor Engineer — Date

REVIEWED BY: ______________________ 5/25/94
Technical Manager — Date

APPROVED BY: ______________________ 5/26/94
Chairman, Plant Operations Review Committee — Date

5231.2381

ROCHESTER GAS ANS ELECTRIC CORPORATION

SAFETY ANALYSIS

MAIN STEAM SYSTEM TEMPORARY INSTRUMENT INSTALLATION

1.0 SCOPE OF ANALYSIS:

1.1 The purpose of this analysis is to evaluate the safety consequences of installing instrumentation and misc piping components (hereafter all items identified as only instrumentation), pressure transmitters, pipe fittings, etc for example, of a pressure and temperature rating meeting or exceeding system design requirements to an existing system connection in the safety class portion of the Main Steam System. Connections will be made downstream of valves 3455 and 3456 at valves 3455A and 3456A.

2.0 REFERENCES:

2.1 A-1406, Control of Temporary Modifications.

2.2 A-303 Preparation, Review and Approval of Safety Analysis.

2.3 RG&E P&ID 33013-1231.

2.4 Temporary Modification 94-019.

3.0 STRUCTURES, SYSTEMS AND COMPONENTS AFFECTED (SSC)

3.1 Structures, systems and components affected directly or indirectly by the temporary modification are the Main Steam Piping and components located between the Main Steam Reverse Check Valves and the Containment Penetrations.

4.0 SAFETY FUNCTION OF AFFECTED SSCs

4.1 Test valves 3455 and 3456 serve as a Containment isolation pressure boundary.

4.2 The temporary instrumentation downstream of valves 3455 and 3456 does not perform a safety function.

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5.0 EFFECTS

5.1 This installation will not adversely affect any other components or equipment with any safety functions. Per the discussion of section 6.0, the control of the instrument and installation process will not create any adverse affects on Safety Related components on the Main Steam Header. Instruments will be supported and restrained per the requirements of the design input and thus will not become dislodged and impact any equipment important to safety.

6.0 UNREVIEWED SAFETY QUESTION CONCLUSIONS

6.1 The proposed installation will not increase the probability of occurrence of an accident previously evaluated in the UFSAR. The addition of an instrument will be in accordance with or exceed the applicable design, material and construction standards of the affected system and does not affect overall system performance since an instrument is generally installed on a deadleg and does not affect system hydraulic design. The typical piping configuration that an instrument would be added to is conservatively designed such that the insignificant weight addition would produce no appreciable increase in induced stress. In the event that the configuration was such that an unacceptable stress level occurred in excess of Code allowables, the governing design document responsible for installation would ensure the proper supporting of the component. This addition will be to a safety significant, seismic components and connect to a safety related system. The safety class of the system is not compromised since the installed materials meet or exceed system design requirements. No analysis for seismic acceptability is required since the installation is intended to be utilized for short periods of time to perform system troubleshooting. During this period the instrument and system isolation valves will be manned to allow for immediate action in the event of a seismic event and would therefore not increase the probability of an accident that was previously evaluated.

6.2 The proposed installation will not increase the consequences of an accident previously evaluated in the UFSAR. This installation will be on safety related system piping will be evaluated to be within the allowable stress requirements of the system piping per the governing design document responsible for installation and is therefore acceptable. An event of

6231.2383

instrument tubing failure would be bounded by a Main Steam Line Break outside Containment since instrument tubing will be a maximum of 3/8" OD and Main Steam piping is a nominal 30" OD.

6.3 The proposed installation will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. This addition is on safety related piping but is intended to be installed for short periods of time, is within system design requirements, will be continuously manned and thus does not adversely affect any equipment important to safety. The instrument will be added to existing system piping and will be designed to meet or exceed the applicable design requirements of the corresponding system.

6.4 The proposed installation will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because this addition does not adversely affect any radiological or fission product barriers. This addition will not cause any safety related equipment to fail in a position not already evaluated since the Operator stationed at the system isolation valve will be capable of manualy closing the valve based upon plant conditions.

6.5 The proposed installation will not create a possibility for an accident of a different type than any evaluated previously in the UFSAR. This installation does not affect the operation or overall design of any existing plant system, thus not creating any condition different than previously considered since weight is negligible for the seismic analysis and materials meet or exceed system design requirements.

6.6 The proposed installation will not create a possibility for a malfunction of equipment of a different type than any evaluated previously in the UFSAR. The instrument will be installed to provide allowable stress levels within the system per the Code (of construction) and will thus ensure an acceptable installed condition.

6.7 The proposed installation will not reduce any margin of safety as defined in the basis of any Technical Specification. Section 3.6.1 allows for the intermittent opening of closed valves that ensure Containment integrity with proper administrative controls. The Temporary Modification, Operations Hold Procedures and Work Order Package instructions provide these controls to an adequate level of detail in order to maintain an

adequate margin of safety. The basis of Technical Section 3.6.1 indicates the opening of closed containment isolation valves on an intermittent basis under administrative controls includes the following considerations: (1) stationing an individual qualified in accordance with station procedures, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring the environmental conditions will not preclude access to isolate the boundary and that this action will prevent the release of radioactivity outside the containment.

7.0 CONCLUSION

7.1 The installation of additional instrumentation to the safety class portion of the Main Steam System does not involve an unreviewed safety question based upon the discussion contained within this document. This addition is inconsequential in that the additional items will have no adverse affect on system operation or design and the additional mass is negligible in comparison to the existing system piping.

5231.2385

SAFETY ANALYSIS

TM 94-022

Removal and Replacement of FIA-2035 with Local Flow Indicator

Rev. 0

Ginna Station

Rochester Gas and Electric Corporation

89 East Ave

Rochester, New York 14649

June 1, 1994

Prepared by:	Staff Engineer	6-2-94 Date
Reviewed by:	Reactor Engineer	6/2/94 Date
Reviewed by:	Manager, Technical Engineering	6-3-94 Date
Approved by:	Chairman, Plant Operations Review Committee	6/5/94 Date

1.0 Scope of Analysis

1.1 The purpose of this analysis is to evaluate the safety consequences of removing FIA-2035(Alarming Flow Indicator for "C" Containment Recirculation Fan Cooler(CRFC)), disabling its MCB alarm(C-10) function, and replacing it with a flow indicator that will provide only local flow indication without MCB alarm function.

2.0 References

2.1 Temporary Modification Evaluation, 94-022, Removal and Replacement of FIA-2035 with Local Flow Indicator, Rev 0

2.2 Ginna UFSAR section 6.2.2.1.1.2 and 6.2.2.1.5

2.3 Ginna Elementary Wiring Diagrams: 10905-320 & 369

2.4 Ginna P&ID 33013-1250 Sht 3, Station Service Cooling Water Safety Related

3.0 Structures, Systems, and Components Affected (SSC)

3.1 The Station Service Water is affected by this change. Specically, the flow indication and the MCB alarming function of FIA-2035 is being altered.

4.0 Safety Functions of Affected SSCs

4.1 The safety function of the portion of the Station Service Water System affected by the temporary modification addressed by this safety analysis is to provide cooling water to the Containment Recirculation Fan Coolers. The safety function of the specific component, FIA-2035, being altered by this temporary modification is to provide local flow indication and MCB alarm(C-10) annunciation for "C" Containment Recirculation Fan Cooler cooling water discharge.

5.0 Effects on Safety

5.1 Per reference 1, the temporary replacement instrument installation satisfies the applicable original piping and instrumentation design requirements. Therefore, there will not be a reduction to the safety margins for the Service Water system.

5.2 Per A-52.4, section 3.1.8, FIA-2035 is not classified as an attendant instrument, being that if inoperable, it does not prevent the "system, subsystem, train, component, or device", from performing the specified function. The absence of the alarm function on FIA-2035 will not reduce capability of the "C" CRFC to perform its safety function, in that, the heat removal capability of the cooler is not affected. The temporary replacement instrument maintains the capability of local flow indication.

5.3 In place of the MCB alarm from "C" CRFC, local indication shall be utilized by Operations to assure required flow to the "C" CRFC unit.

6.0 Unreviewed Safety Question Conclusions

6.1 The proposed modification will not increase the probability of occurrence of an accident previously evaluated in the UFSAR because the Temporary Modification Evaluation, 94-022, determined the temporary replacement instrument installation meets the design requirements of the safety-related Service Water system.

6.2 The proposed modification will not increase the consequences of an accident previously evaluated in the UFSAR because the heat removal capability of the "C" CRFC will not be affected.

6.3 The proposed modification will not increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because the capability to verify flow to the "C" CRFC shall be maintained and the temporary replacement instrument installation meets the design requirements of the safety-related Service Water system.

6.4 The proposed modification will not create the possibility of an accident or malfunction of equipment of a different type than any evaluated in the UFSAR because the temporary replacement instrument installation meets the design requirements of the safety-related Service Water system.

6.5 The proposed modification will not reduce any margin of safety as defined in the basis of any Technical Specification because FIA-2035 is not considered attendant instrumentation for the "C" CRFC.

7.0 Conclusion

Based on the above discussion the proposed modification does not present an unreviewed safety question.

SECTION E - PROCEDURE CHANGES

This section contains a description of the changes to procedures as described in the UFSAR and a summary of the safety evaluation pursuant to the requirements of 10 CFR 50.59(b).

FIGURE 2

SAFETY EVALUATION SUMMARY FORM

Procedure #: RF 8.4 Date: 5/24/94 PCN #: 94-T-0546

Exclusion from Screening Criteria Items 1, 2, or 6

If "YES" is answered for Items 1 or 2, provide the type of "inconsequential change" or the referenced 10CFR50.59 Safety Evaluation below:

Change Type: ______

If "NO" was answered for Item 6, provide the basis for exclusion below:

Basis for Exclusion: ______

10CFR50.59 Safety Evaluation - Item 7

If "NO" has been answered for each question in Items 7a through 7g, this change is not an Unreviewed Safety Question. Document the justification for these conclusions below. List any material referenced in the space provided.

Written Justification: THIS CHANGE DOES NOT INCREASE THE POSSIBILITY, PROBABILITY OR CONSEQUENCES OF AN ACCIDENT DESCRIBED IN THE UFSAR. THE TRACK BUMPERS PROTECT THE LINER PLATE FROM BEING CONTACTED BY THE SFP BRIDGE CRANE, THEY DO NOT PREVENT CRANE MALFUNCTION. PROCEDURE STEPS AND PRECAUTIONS PROTECT THE SFP LINER FROM DAMAGE

Referenced Material: UFSAR CHAPTER 1, 3 AND 9

If "YES" was answered for Item 3, check this box ☑ and submit a UFSAR change per A-65.

If "YES" was answered for Item 7, PORC shall review and approve this submittal. This proposed change is an Unreviewed Safety Question (USQ) and requires submittal to the NRC for their review.

Submitted By: [signature]

PORC'd
3-23-94

FIGURE 2
SAFETY EVALUATION SUMMARY FORM

Procedure # A-201.5 Date 11-15-93 PCN # 93-1395

Exclusion from Screening Criteria Items 1, 2, or 6

If "YES" is answered for Items 1 or 2, provide the type of "inconsequential change" or the referenced 10CFR50.59 Safety Evaluation below:

Change Type: clarification of STA duties

If "NO" was answered for Item 6, provide the basis for exclusion below:

Basis for Exclusion:

10CFR50.59 Safety Evaluation - Item 7

If "NO" has been answered for each question in Items 7a through 7g, this change is not an Unreviewed Safety Question. Document the justification for these conclusions below. List any material referenced in the space provided.

Written Justification: The Ginna organization and responsibilities are described in the UFSAR. The recent reorganization of Ginna has changed who is responsible for certain activities. No responsibilities were eliminated. Therefore this is strictly an administrative change and does not create an unreviewed safety question.

Referenced Material: UFSAR section 13.1

If "YES" was answered for Item 3, check this box ☑ and submit a UFSAR change per A-65.

If "YES" was answered for Item 7, PORC shall review and approve this submittal. This proposed change is an Unreviewed Safety Question (USQ) and requires submittal to the NRC for their review.

Submitted By: [signature]

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LF BUSINESS FORMS, INC.

ROCHESTER GAS AND ELECTRIC CORPORATION

GINNA STATION PROCEDURE CHANGE NOTICE

PROCEDURE NO. A-201.5 REV. NO. 6 INITIATION DATE 11/15/93 PCN NO. 93-4395

- [ ] NEW PROCEDURE — INITIATOR: K Wayland
- [x] PERMANENT CHANGE — RESPONSIBLE MANAGER: [signature]
- [ ] TEMPORARY - REVIEW ONLY — PLANT STAFF: [signature]
- [ ] TEMPORARY - PERMANENT — DUTY ENGINEER *
- [x] PERIODIC REVIEW — SHIFT SUPERVISOR *

* EXPIRATION DATE ______ * PORC REVIEW BY DATE ______

* (FOR TEMPORARY PCN'S ONLY)

* [ ] ONE TIME USE ONLY EQUIPMENT ID ______ *

see attached

REASON: Technical section re-organization & enhancement of ~~STA~~ description of STA responsibilities + periodic review

PORC RECOMMENDATION:

APPROVAL [x] DISAPPROVAL [ ]

PORC COMMENTS OR MODIFICATIONS: ______

PORC REVIEW DATE ______ EFFECTIVE DATE ______

FIGURE 2

SAFETY EVALUATION SUMMARY FORM

PROCEDURE # A-1 DATE 4/1/93 PCN # 93-6039

Exclusion from Screening Criteria - Items 1, 2, or 6

If "yes" is answered for Items 1 or 2, provide the type of "inconsequential change" or the referenced 10CFR50.59 safety evaluation below:

Change Type: ____________

If "no" was answered for Item 6, provide the basis for exclusion below:

Basis for Exclusion: ____________

10CFR50.59 Safety Evaluation - Item 7

If "no" has been answered for each question in items 7a through 7g this change is not an Unreviewed Safety Question. Document the justification for these conclusions below. List any material referenced in the space provided.

Written Justification: This change does not represent a USQ. The changes are based on revision to 10CFR20 issued May 21, 1991 and effective June 20, 1991. Implementation date is January 1, 1994. See attached for details

Referenced Material: 10CFR20 §§ 20.1001 - 20.2401

| If "yes" was answered for Item 3, check this box |X| and submit
| a UFSAR change per A-65.

If "yes", was answered for Item 7, PORC shall review and approve this submittal. This proposed change is an Unreviewed Safety Question (USQ) and requires submittal to the NRC for their review.

Submitted By: William H Thomson

ATTACHMENT 7.1

PROCEDURE CHANGE SUMMARY

REV 1

Procedure No.: A-1	PCN: 93-6039	TCR/N ☐ N/A
Procedure Title: Radiation Control Manual		
Other Procedures Affected by Change Identified on 10CFR20 changes Procedure Matrix	Training required ☑ Initial --- ☐ OJT ☑ Class RP ; GET ☑ Continuing - ☐ OJT ☑ Class ☐ Meeting ☐ RR	
Reason for Change ☐ Commitment ☐ New/change equipment ☑ Regulation, standard or code ☐ Other______________		
Point of Contact: Jeff Johnston		Ext: 6583

SUMMARY OF CHANGE

1. Reformatted procedure to conform w/ A-502 format requirements.
2. Revised dose limits and definitions to comply w/ the revised 10CFR20.
3. Reorganized procedure sections to more closely follow the text of ND-RPP "Rad. Protection Program"
4. Removed specific information concerning Radwaste shipping - info will be included in Radwaste Section procedures.

10CFR50.59 Support Documentation
Procedure # A-1
PCN # 93-069

1. A-1 Section 3.1 Definitions

The following definitions are traceable to the revised USNRC 10CFR20.1003 Definitions

Airborne radioactivity area
ALARA
ALI
Committed dose equivalent (CDE)
Committed effective dose equivalent (CEDE)
Controlled area
Declared pregnant woman
Deep dose equivalent (DDE)
DAC, DAC-hr
Eye dose equivalent (LDE)
High Radiation area
Occupational dose
Planned Special Exposure (PSE)
Restricted area
Radiation area
Shallow dose equivalent (SDE)
Site boundary
Total effective dose equivalent (TEDE)
Very High Radiation area
Whole body

Protected Area is traceable to the USNRC 10CFR73.2 Definitions

Total organ dose equivalent (TODE) is traceable to USNRC Regulatory Guide 8.7 Section 2.2

Unrestricted area is traceable to the current USNRC 10CFR20.3 Definitions

2. Titles, terms, and units used in A-1

Title changes from "Health Physics" to "Radiation Protection", as used in titles such as the Radiation Protection Program, Manager-Radiation Protection and Chemistry, and Radiation Protection technician, are traceable to USNRC 10CFR20.1101, USNRC Regulatory Guide 1.8, and USNRC Regulatory Guide 8.8.

Revised USNRC 10CFR20.2101 requires licensees to use the units of rem (including multiples and subdivisions) on all records and surveys required by the regulation. In addition , the exposure unit roentgen is no longer defined in the revised part 20. Most of the radiological monitoring equipment used for surveys at the Ginna station provide results in the units of roentgen or rad. For interpretation of these survey results, the assumption shall be made that:

$$1 \text{ roentgen/hr} = 1 \text{ rad/hr} = 1 \text{ rem/hr}$$

given

$$\dot{D} = \frac{87.7}{100} \times \frac{(\mu_{en}/\rho)_m}{(\mu_{en}/\rho)_a} \times \dot{X}$$

where

$\dot{D}$ = absorbed dose rate in medium of interest (rad)
$\dot{X}$ = exposure rate in air (R)
$(\mu_{en}/\rho)_m$ = mass energy-absorption coefficient of medium
$(\mu_{en}/\rho)_a$ = mass energy-absorption coefficient of air

and

USNRC revised 10CFR20.1004, Table 1004(b).1 lists the quality factor for X-, gamma, or beta radiation as 1.0.

3. A-1 Section 3.6 Station Internal Dose Monitoring and Control

Based on historical data reviewed at the Ginna station for intake of radioactive material by individuals working in radiologically controlled areas, the maximally exposed individual received a committed dose equivalent of 227 mrem, and in 1993 no individual has received a committed dose equivalent greater than 151 mrem to date. Given that 2.5 mrem is approximately equal to 1 DAC-hr, no individual working at the Ginna station has received greater than 200 DAC-hrs, or a committed dose equivalent of 500 mrem, in any given year. Therefore, internal dose monitoring of individuals at the Ginna station is not required, as ~~specified~~ allowed in the revised USNRC 10CFR20.1502.

Procedures have been developed and shall be available in the event of an unusual circumstance resulting in an internal dose in excess of the 10% limit so that a summation of internal and external doses may be performed.

FIGURE 1

10CFR50.59
CRITERIA/CHECKLIST FOR REVIEWING
PROCEDURE CHANGES

The following set of questions will be used to screen procedure changes and to determine if the change requires a 10 CFR 50.59 safety evaluation. The attached Safety Evaluation Summary Form shall be utilized to document the basis and justification for the conclusions. The completed Safety Evaluation Summary Form shall be attached to the PCN and submitted for PORC review.

NOTE: When developing this checklist, if there is any doubt as to any particular answer to a question, answer so as to continue processing this form.

NOTE: A 10CFR50.59 safety evaluation does not have to be completed for changes to the Ginna Security procedures (GS) and Emergency Plan Implementing procedures (EPIP'S). Changes to these procedures should be made in accordance with 10 CFR 50.54(p) and 10 CFR 50.54(q) respectively.

1. Is this change an inconsequential change as listed in the following examples?

    (a) a typographical change
    (b) a spelling change or grammar correction
    (c) a clarification such as:

    - identification of additional plant equipment such as valve lists, etc.
    - addition of illustrative diagrams.
    - lists of personnel names.
    - addition of notes or clarification of steps.
    - addition of steps that add conservatisms, such as verification signoffs.

    (d) Expansion of steps of a procedure into additional steps or substeps that clarify, but do not change the procedure

    (e) Deletion of a procedure no longer required to comply with any existing commitments. (i.e. Deletion of an SM procedure that fulfills the requirements of the design package.)

NOTE: When considering these exceptions it is noted that reordering of steps of a procedure may be considered an inconsequential change only if this reordering does not change the intent of that series of steps.
Inconsequential changes do not affect the purpose,

(# 1 Cont'd)

methodology (procedural instructions), or the technical content of the procedure.
Examples of changes to "technical content" are:

- initial conditions
- acceptance criteria
- safety limits
- tolerances
- reference values
- setpoints

Examples of "intent" changes:

- method of accomplishment
- sequence of accomplishment
- intent of individual steps of the procedure.
- overall purpose of the procedure

_____ yes √ no

If "yes", denote this type of change on the Safety Evaluation Summary Form as described in items 1(a) through 1(e) above. No further processing need be done for this change. Submit for PORC review along with the rest of the PCN package.

If "no" continue processing this form.

2. Is this change the result of another procedure change that has already had a 10CFR50.59 safety evaluation developed on it?

_____ yes √ no

If "yes" the original safety evaluation covers this change. No further processing of this form is required. Document this basis on the Safety Evaluation Summary Form and submit for PORC review along with the rest of the PCN package.

If "no" continue processing this form.

3. Does this change result in a non-compliance with a procedure or procedural commitment that is described (not just listed by individual procedure number, or as a category) in the UFSAR? Does this change cause a description in the UFSAR to be incorrect?

√ yes _____ no

(# 3 Cont'd)

Examples:

(a) If in the description of a radioactive waste system in the UFSAR, it is stated that the Shift Supervisor will authorize all radioactive liquid releases, a safety evaluation to meet the requirements of 10CFR50.59 would be required before assigning this function to another individual. On the other hand, if the UFSAR merely states that radioactive liquid releases will be authorized as detailed by plant procedures, the redesignation of the authorization function would not require a safety evaluation under the requirements of 10CFR50.59.

(b) If the reactor startup procedure, as described in the UFSAR, contains eight fundamental sequences, the decision to eliminate one of the sequences would require a safety evaluation in accordance with 10 CFR50.59. On the other hand, if the change consolidated the eight fundamental sequences and did not alter the basic functions performed, it would not be necessary to conduct a safety evaluation under the requirements of 10CFR50.59

√ yes ____ no

If "yes" proceed directly to question #7 of this form, and conduct a 10CFR50.59 Safety Evaluation, and submit a UFSAR change in accordance with A-65.

If "no" continue processing this form.

4. Does this change result in a non-compliance with the Technical Specifications?

____ yes √ no

If "yes" this change will require a Technical Specification amendment submittal in accordance with 10 CFR50.90, and this change request will be put on an administrative hold pending receipt of an approved amendment.

If "no" continue processing this form.

5. Does this change degrade the operational capability to preclude or mitigate plant transients or postulated accidents?

____ yes √ no

If "yes" proceed directly to question #7 of this form and conduct a 10CFR50.59 Safety Evaluation.

If "no" continue processing this form.

6. Will this procedure change create the potential for operation of plant equipment outside the equipments design parameters or place the equipment/system in a configuration not originally intended?

_____ yes __√__ no

If "yes" proceed directly to question #7 of this form and conduct a 10CFR50.59 Safety Evaluation.

If "no", document basis on the Safety Evaluation Summary Form and submit for PORC review, along with the rest of the PCN package. No further processing of this form is required.

7. 10CFR50.59 Safety Evaluation

Please answer the following seven questions concerning the procedure change. Documentation in the form of a written justification shall be provided on the Safety Evaluation Summary Form. This shall include referenced material such as Technical Specifications, Vendor Manuals, UFSAR etc., and the basis for the conclusion drawn. Refer to the "10 CFR 50.59 Guidance Document" for guidance concerning the written justification to support the answers to the seven questions identified below. As a result of the proposed procedure change:

a) Will the probability of occurrence of an accident previously evaluated in the UFSAR be increased?

Yes _______ No __√__

b) Will the consequences of an accident previously evaluated in the UFSAR be increased?

Yes _______ No __√__

c) Will the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

Yes _______ No __√__

d) Will the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR be increased?

Yes _______ No __√__

e) Will the possibility of an accident of a different type than any previously evaluated in the UFSAR be created?

Yes _______ No __√__

(#7 Cont'd)

f) Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR be created?

Yes ________ No ✓________

g) Will the margin of safety as defined in the basis for any technical specification be reduced?

Yes ________ No ✓________

See attached documentation

Uptakes in 4/75 - largest in history of Trunia

$$CEDE = \frac{\text{Intake } (\mu Ci)}{(5)\ ALI \text{ from APP B, 10 CFR20}} \times 5 \text{ rem}$$

	µCi	Class	µCi (arrival)	µCi (net)
Co-60 (Lung)	1.39	(Y)	0.19	1.20
Co-58 (Lung)	1.84	(W)	0.08	1.76
Zr-95 (Lung)	0.785	(W)	—	—
Nb-95 (Lung)	0.878	(W)	—	—
Mn-54 (Lung)	0.082	(Y)	—	—
I-131 (Thyroid)	0.105	(D)	—	—

Co-60 = $\frac{1.20}{30} \times 5$ = 0.2 rem = 200 mrem

Co-58 = $\frac{1.76}{1000} \times 5$ = 0.00263 rem = 3 mrem

Zr-95 $\frac{0.785}{400} \times 5$ = 0.0088 rem = 9 mrem

Nb-95 $\frac{0.815}{1000} \times 5$ = 0.0098 rem = 10 mrem

Mn-54 $\frac{0.082}{800} \times 5$ = 0.0041 rem = 4 mrem

I-131 $\frac{0.105}{200} \times 5$ = 0.0005 rem = 1 mrem

CEDE = 227 mrem 4.52%

we were fined 4,000

BRQuinn

	Fed Rep # 11 (CEDE)	10 CFR 20 (ALI)	ICRP 30 (ALI)
Co-60	219 mRem/μCi	167 mRem/μCi	185 mRem/μCi
(1.2 μCi)	{263}	{200}	{222}
Co-58	10.9 mRem/μCi	7.14 mRem/μCi	6.17 mRem/μCi
(1.76)	{19.2}	{12.6}	{10.9}
Zr-95	23.3 mRem/μCi	16.7 mRem/μCi	14.0 mRem/μCi
(0.785)	{18.3}	{13.1}	{11}
Nb-95	5.81 mRem/μCi	5.00 mRem/μCi	4.63 mRem/μCi
(0.878)	{5.1}	{4.39}	{4.07}
Mn-54	6.7 mRem/μCi	6.25 mRem/μCi	6.17 mRem/μCi
(0.082)	{0.55}	{0.513}	{0.506}
I-131	32.9 mRem/μCi	25 mRem/μCi	30.9 mRem/μCi
(0.105)	{3.45}	{2.63}	{3.24}
	~310 mRem	~233 mRem	~~[illegible]~~ ~~[illegible] mRem~~ ~252 mRem

I suggest using 10CFR20. Its safe to use the NRC method, which is based on ICRP-20. Fed Report 11 does not drop organ with < 10% contribution to CEDE / Isotope.

500 mRem using EPA Fed Report # 11

Data used to calculate CEDE for typical power plant radionuclides.

Radionuclides	Fed. Rep. No. 11 CEDE Con Fact. (Sv/Bq)	Fed. Rep. No. 11 CEDE Con Fact. (mrem/uCi)	10 CFR Part 20 ALI (uCi)	10 CFR Part 20 DAC (uCi/mL)	A(uCi)
H-3	1.73E-11	6.40E-02	8.00E+04	2.00E-05	7.81E+03
C-14	6.36E-12	2.35E-02	2.00E+05	9.00E-05	2.12E+04
Cr-51	9.03E-11	3.34E-01	2.00E+04	8.00E-06	1.50E+03
Mn-54	1.81E-09	6.70E+00	8.00E+02	3.00E-07	7.47E+01
Mn-56	8.91E-11	3.30E-01	2.00E+04	9.00E-06	1.52E+03
Co-58	2.94E-09	1.09E+01	7.00E+02	3.00E-07	4.60E+01
Co-60	5.91E-08	2.19E+02	3.00E+01	1.00E-08	2.29E+00
Fe-55	3.61E-10	1.34E+00	4.00E+03	2.00E-06	3.74E+02
Fe-59	3.30E-09	1.22E+01	5.00E+02	2.00E-07	4.10E+01
Zn-65	5.51E-09	2.04E+01	3.00E+02	1.00E-07	2.45E+01
Br-84	2.27E-11	8.40E-02	6.00E+04	3.00E-05	5.95E+03
Kr-85				1.00E-04	
Kr-85m				2.00E-05	
Kr-87				5.00E-06	
Kr-88				2.00E-06	
Rb-86	1.79E-09	6.62E+00	8.00E+02	3.00E-07	7.55E+01
Rb-88	2.26E-11	8.36E-02	6.00E+04	3.00E-05	5.98E+03
Rb-89	1.16E-11	4.29E-02	1.00E+05	6.00E-05	1.16E+04
Sr-89	1.12E-08	4.14E+01	1.00E+02	6.00E-08	1.21E+01
Sr-90	3.51E-07	1.30E+03	4.00E+00	2.00E-09	3.85E-01
Sr-91	4.49E-10	1.66E+00	4.00E+03	1.00E-06	3.01E+02
Sr-92	2.18E-10	8.07E-01	7.00E+03	3.00E-06	6.20E+02
Y-90	2.28E-09	8.44E+00	6.00E+02	3.00E-07	5.93E+01
Y-91	1.32E-08	4.88E+01	1.00E+02	5.00E-08	1.02E+01
Y-92	2.11E-10	7.81E-01	8.00E+03	3.00E-06	6.40E+02
Zr-95	6.31E-09	2.33E+01	3.00E+02	1.00E-07	2.14E+01
Zr-97	1.17E-09	4.33E+00	1.00E+03	5.00E-07	1.16E+02
Nb-95	1.57E-09	5.81E+00	1.00E+03	5.00E-07	8.61E+01
Mo-99	1.07E-09	3.96E+00	1.00E+03	6.00E-07	1.26E+02
Tc-99m	7.21E-12	2.67E-02	2.00E+05	1.00E-04	1.87E+04
Ru-103	2.42E-09	8.95E+00	6.00E+02	3.00E-07	5.58E+01
Ru-105	1.23E-10	4.55E-01	1.00E+04	5.00E-06	1.10E+03
Ru-106	1.29E-07	4.77E+02	1.00E+01	5.00E-09	1.05E+00
Rh-105	2.58E-10	9.55E-01	6.00E+03	2.00E-06	5.24E+02
Ag-110m	2.17E-08	8.03E+01	9.00E+01	4.00E-08	6.23E+00
Sb-122	1.39E-09	5.14E+00	1.00E+03	4.00E-07	9.72E+01
Sb-124	6.80E-09	2.52E+01	2.00E+02	1.00E-07	1.99E+01
Sb-125	3.30E-09	1.22E+01	5.00E+02	2.00E-07	4.10E+01
Sb-127	1.63E-09	6.03E+00	9.00E+02	4.00E-07	8.29E+01
Sb-129	1.74E-10	6.44E-01	9.00E+03	4.00E-06	7.77E+02
Te-127	8.60E-11	3.18E-01	2.00E+04	7.00E-06	1.57E+03
Te-127m	5.81E-09	2.15E+01	3.00E+02	1.00E-07	2.33E+01
Te-129	2.09E-11	7.73E-02	7.00E+04	3.00E-05	6.47E+03
Te-129m	6.47E-09	2.39E+01	2.00E+02	1.00E-07	2.09E+01
Te-131m	1.73E-09	6.40E+00	9.00E+02	3.75E-07	7.81E+01
Te-132	2.55E-09	9.44E+00	6.00E+02	2.50E-07	5.30E+01
Te-134	3.23E-11	1.20E-01	5.00E+04	2.10E-05	4.18E+03

Data used to calculate CEDE for typical power plant radionuclides.

	Fed. Rep. No. 11	Fed. Rep. No. 11	10 CFR Part 20	10 CFR Part 20	
Radionuclides	CEDE Con Fact. (Sv/Bq)	CEDE Con Fact. (mrem/uCi)	ALI (uCi)	DAC (uCi/mL)	A(uCi)
I-131	8.89E-09	3.29E+01	2.00E+02	8.30E-08	1.52E+01
I-132	1.03E-10	3.81E-01	1.00E+04	4.20E-06	1.31E+03
I-133	1.58E-09	5.85E+00	9.00E+02	3.80E-07	8.55E+01
I-134	3.55E-11	1.31E-01	5.00E+04	2.10E-05	3.81E+03
I-135	3.32E-10	1.23E+00	4.00E+03	1.70E-06	4.07E+02
Xe-133				1.00E-04	
Xe-133m				1.00E-04	
Xe-135				1.00E-05	
Xe-135m				9.00E-04	
Xe-138				4.00E-06	
Cs-134	1.25E-08	4.63E+01	1.00E+02	4.00E-08	1.08E+01
Cs-136	1.98E-09	7.33E+00	7.00E+02	3.00E-07	6.83E+01
Cs-137	8.63E-09	3.19E+01	2.00E+02	6.00E-08	1.57E+01
Cs-138	2.74E-11	1.01E-01	6.00E+04	2.00E-05	4.93E+03
Ba-140	1.01E-09	3.74E+00	1.00E+03	6.00E-07	1.34E+02
La-140	1.31E-09	4.85E+00	1.00E+03	5.00E-07	1.03E+02
Ce-141	2.42E-09	8.95E+00	6.00E+02	2.00E-07	5.58E+01
Ce-143	9.16E-10	3.39E+00	2.00E+03	7.00E-07	1.48E+02
Ce-144	1.01E-07	3.74E+02	1.00E+01	6.00E-09	1.34E+00
Pr-143	2.19E-09	8.10E+00	7.00E+02	3.00E-07	6.17E+01
Pr-144	1.17E-11	4.33E-02	1.00E+05	5.00E-05	1.16E+04
Nd-147	1.85E-09	6.85E+00	8.00E+02	4.00E-07	7.30E+01
Np-239	6.78E-10	2.51E+00	2.00E+03	9.00E-07	1.99E+02
Pu-238	7.79E-05	2.88E+05	2.00E-02	8.00E-12	1.73E-03
Pu-239	8.33E-05	3.08E+05	2.00E-02	8.30E-12	1.62E-03
Pu-240	8.33E-05	3.08E+05	2.00E-02	8.30E-12	1.62E-03
Pu-241	1.34E-06	4.96E+03	1.00E+00	4.20E-10	1.01E-01
Am-241	1.20E-04	4.44E+05	1.00E-02	4.20E-12	1.13E-03
Cm-242	4.67E-06	1.73E+04	3.00E-01	1.25E-10	2.89E-02
Cm-244	6.70E-05	2.48E+05	2.00E-02	8.30E-12	2.02E-03

Calculations of the CEDE and percent differences between methods.

Radionuclides	CEDE (mrem) Fed. Rep. No. 11	CEDE (mrem) ALI	CEDE (mrem) DAC	% Difference No. 11 & ALI	% Difference No. 11 & DAC	% Difference ALI & DAC
H-3	5.00E+02	4.88E+02	8.14E+02	2	48	50
C-14	5.00E+02	5.31E+02	4.92E+02	6	2	8
Cr-51	5.00E+02	3.74E+02	3.90E+02	29	25	4
Mn-54	5.00E+02	4.67E+02	5.18E+02	7	4	11
Mn-56	5.00E+02	3.79E+02	3.51E+02	27	35	8
Co-58	5.00E+02	3.28E+02	3.19E+02	41	44	3
Co-60	5.00E+02	3.81E+02	4.76E+02	27	5	22
Fe-55	5.00E+02	4.68E+02	3.90E+02	7	25	20
Fe-59	5.00E+02	4.10E+02	4.27E+02	20	16	4
Zn-65	5.00E+02	4.09E+02	5.11E+02	20	2	22
Br-84	5.00E+02	4.96E+02	4.13E+02	1	21	20
Kr-85						
Kr-85m						
Kr-87						
Kr-88						
Rb-86	5.00E+02	4.72E+02	5.24E+02	6	5	11
Rb-88	5.00E+02	4.98E+02	4.15E+02	0	21	20
Rb-89	5.00E+02	5.82E+02	4.04E+02	15	21	36
Sr-89	5.00E+02	6.03E+02	4.19E+02	21	20	36
Sr-90	5.00E+02	4.81E+02	4.01E+02	4	22	20
Sr-91	5.00E+02	3.76E+02	6.27E+02	28	23	50
Sr-92	5.00E+02	4.43E+02	4.30E+02	12	15	3
Y-90	5.00E+02	4.94E+02	4.12E+02	1	21	20
Y-91	5.00E+02	5.12E+02	4.27E+02	2	16	20
Y-92	5.00E+02	4.00E+02	4.45E+02	22	12	11
Zr-95	5.00E+02	3.57E+02	4.46E+02	33	11	22
Zr-97	5.00E+02	5.78E+02	4.81E+02	14	4	20
Nb-95	5.00E+02	4.30E+02	3.59E+02	15	33	20
Mo-99	5.00E+02	6.31E+02	4.39E+02	23	13	36
Tc-99m	5.00E+02	4.69E+02	3.90E+02	6	25	20
Ru-103	5.00E+02	4.65E+02	3.88E+02	7	25	20
Ru-105	5.00E+02	5.49E+02	4.58E+02	9	9	20
Ru-106	5.00E+02	5.24E+02	4.36E+02	5	14	20
Rh-105	5.00E+02	4.36E+02	5.46E+02	14	9	22
Ag-110m	5.00E+02	3.46E+02	3.24E+02	36	43	6
Sb-122	5.00E+02	4.86E+02	5.06E+02	3	1	4
Sb-124	5.00E+02	4.97E+02	4.14E+02	1	21	20
Sb-125	5.00E+02	4.10E+02	4.27E+02	20	16	4
Sb-127	5.00E+02	4.61E+02	4.32E+02	8	15	6
Sb-129	5.00E+02	4.31E+02	4.04E+02	15	21	6
Te-127	5.00E+02	3.93E+02	4.68E+02	24	7	17
Te-127m	5.00E+02	3.88E+02	4.85E+02	25	3	22
Te-129	5.00E+02	4.62E+02	4.49E+02	8	11	3
Te-129m	5.00E+02	5.22E+02	4.35E+02	-4	14	20
Te-131m	5.00E+02	4.34E+02	4.34E+02	14	14	0
Te-132	5.00E+02	4.42E+02	4.42E+02	12	12	0
Te-134	5.00E+02	4.18E+02	4.15E+02	20	21	1

Calculations of the CEDE and percent differences between methods.

	CEDE (mrem)	CEDE (mrem)	CEDE (mrem)	% Difference	% Difference	% Difference
Radionuclides	Fed. Rep. No. 11	ALI	DAC	No. 11 & ALI	No. 11 & DAC	ALI & DAC
I-131	5.00E+02	3.80E+02	3.82E+02	27	27	0
I-132	5.00E+02	6.56E+02	6.51E+02	27	26	1
I-133	5.00E+02	4.75E+02	4.69E+02	5	6	1
I-134	5.00E+02	3.81E+02	3.78E+02	27	28	1
I-135	5.00E+02	5.09E+02	4.99E+02	2	0	2
Xe-133						
Xe-133m						
Xe-135						
Xe-135m						
Xe-138						
Cs-134	5.00E+02	5.41E+02	5.63E+02	8	12	4
Cs-136	5.00E+02	4.88E+02	4.74E+02	3	5	3
Cs-137	5.00E+02	3.91E+02	5.44E+02	24	8	33
Cs-138	5.00E+02	4.11E+02	5.14E+02	20	3	22
Ba-140	5.00E+02	6.69E+02	4.65E+02	29	7	36
La-140	5.00E+02	5.16E+02	4.30E+02	3	15	20
Ce-141	5.00E+02	4.65E+02	5.82E+02	7	15	22
Ce-143	5.00E+02	3.69E+02	4.39E+02	30	13	17
Ce-144	5.00E+02	6.69E+02	4.65E+02	29	7	36
Pr-143	5.00E+02	4.41E+02	4.29E+02	13	15	3
Pr-144	5.00E+02	5.78E+02	4.81E+02	14	4	20
Nd-147	5.00E+02	4.57E+02	3.80E+02	9	27	20
Np-239	5.00E+02	4.98E+02	4.61E+02	0	8	8
Pu-238	5.00E+02	4.34E+02	4.52E+02	14	10	4
Pu-239	5.00E+02	4.06E+02	4.07E+02	21	20	0
Pu-240	5.00E+02	4.06E+02	4.07E+02	21	20	0
Pu-241	5.00E+02	5.04E+02	5.00E+02	1	0	1
Am-241	5.00E+02	5.63E+02	5.59E+02	12	11	1
Cm-242	5.00E+02	4.82E+02	4.82E+02	4	4	0
Cm-244	5.00E+02	5.04E+02	5.06E+02	1	1	0

List of ALI's and DAC's from 10 CFR Part 20 and ICRP 30 for radionuclides

	10 CFR Part 20	10 CFR Part 20	ICRP 30	ICRP 30	ICRP 30	ICRP 30
NUCLIDES	ALI (uCi)	DAC (uCi/mL)	ALI (Bq)	ALI (uCi)	DAC (Bq/m^3)	DAC (uCi/mL)
H-3	8.00E+04	2.00E-05	3.00E+09	8.11E+04	8.00E+05	2.16E-05
C-14	2.00E+05	9.00E-05				
Cr-51	2.00E+04	8.00E-06	7.00E+08	1.89E+04	3.00E+05	8.11E-06
Mn-54	8.00E+02	3.00E-07	3.00E+07	8.11E+02	1.00E+04	2.70E-07
Mn-56	2.00E+04	9.00E-06	8.00E+08	2.16E+04	3.00E+05	8.11E-06
Co-58	7.00E+02	3.00E-07	3.00E+07	8.11E+02	1.00E+04	2.70E-07
Co-60	3.00E+01	1.00E-08	1.00E+06	2.70E+01	5.00E+02	1.35E-08
Fe-55	4.00E+03	2.00E-06	2.00E+08	5.41E+03	6.00E+04	1.62E-06
Fe-59	5.00E+02	2.00E-07	2.00E+07	5.41E+02	8.00E+03	2.16E-07
Zn-65	3.00E+02	1.00E-07	1.00E+07	2.70E+02	4.00E+03	1.08E-07
Br-84	6.00E+04	3.00E-05	2.00E+09	5.41E+04	1.00E+06	2.70E-05
Kr-85		1.00E-04				
Kr-85m		2.00E-05				
Kr-87		5.00E-06				
Kr-88		2.00E-06				
Rb-86	8.00E+02	3.00E-07	3.00E+07	8.11E+02	1.00E+04	2.70E-07
Rb-88	6.00E+04	3.00E-05	2.00E+09	5.41E+04	1.00E+06	2.70E-05
Rb-89	1.00E+05	6.00E-05	5.00E+09	1.35E+05	2.00E+06	5.41E-05
Sr-89	1.00E+02	6.00E-08	5.00E+06	1.35E+02	2.00E+03	5.41E-08
Sr-90	4.00E+00	2.00E-09	1.00E+05	2.70E+00	6.00E+01	1.62E-09
Sr-91	4.00E+03	1.00E-06	1.00E+08	2.70E+03	5.00E+04	1.35E-06
Sr-92	7.00E+03	3.00E-06	2.00E+08	5.41E+03	1.00E+05	2.70E-06
Y-90	6.00E+02	3.00E-07	2.00E+07	5.41E+02	9.00E+03	2.43E-07
Y-91	1.00E+02	5.00E-08	4.00E+06	1.08E+02	2.00E+03	5.41E-08
Y-92	8.00E+03	3.00E-06	3.00E+08	8.11E+03	1.00E+05	2.70E-06
Zr-95	3.00E+02	1.00E-07	1.00E+07	2.70E+02	4.00E+03	1.08E-07
Zr-97	1.00E+03	5.00E-07	5.00E+07	1.35E+03	2.00E+04	5.41E-07
Nb-95	1.00E+03	5.00E-07	4.00E+07	1.08E+03	2.00E+04	5.41E-07
Mo-99	1.00E+03	6.00E-07	5.00E+07	1.35E+03	2.00E+04	5.41E-07
Tc-99m	2.00E+05	1.00E-04	9.00E+09	2.43E+05	4.00E+06	1.08E-04
Ru-103	6.00E+02	3.00E-07	2.00E+07	5.41E+02	1.00E+04	2.70E-07
Ru-105	1.00E+04	5.00E-06	4.00E+08	1.08E+04	2.00E+05	5.41E-06
Ru-106	1.00E+01	5.00E-09	4.00E+05	1.08E+01	2.00E+02	5.41E-09
Rh-105	6.00E+03	2.00E-06	2.00E+08	5.41E+03	9.00E+04	2.43E-06
Ag-110m	9.00E+01	4.00E-08	3.00E+06	8.11E+01	1.00E+03	2.70E-08
Sb-122	1.00E+03	4.00E-07	4.00E+07	1.08E+03	2.00E+04	5.41E-07
Sb-124	2.00E+02	1.00E-07	9.00E+06	2.43E+02	4.00E+03	1.08E-07
Sb-125	5.00E+02	2.00E-07	2.00E+07	5.41E+02	8.00E+03	2.16E-07
Sb-127	9.00E+02	4.00E-07	3.00E+07	8.11E+02	1.00E+04	2.70E-07
Sb-129	9.00E+03	4.00E-06	3.00E+08	8.11E+03	1.00E+05	2.70E-06
Te-127	2.00E+04	7.00E-06	6.00E+08	1.62E+04	3.00E+05	8.11E-06
Te-127m	3.00E+02	1.00E-07	9.00E+06	2.43E+02	4.00E+03	1.08E-07
Te-129	7.00E+04	3.00E-05	3.00E+09	8.11E+04	1.00E+06	2.70E-05
Te-129m	2.00E+02	1.00E-07	9.00E+06	2.43E+02	4.00E+03	1.08E-07
Te-131m	9.00E+02	3.75E-07	4.00E+07	1.08E+03	1.00E+04	2.70E-07
Te-132	6.00E+02	2.50E-07	1.00E+07	2.70E+02	3.00E+03	8.11E-08
Te-134	5.00E+04	2.10E-05	7.00E+08	1.89E+04	1.00E+05	2.70E-06

List of ALI's and DAC's from 10 CFR Part 20 and ICRP 30 for radionuclides

	10 CFR Part 20	10 CFR Part 20	ICRP 30	ICRP 30	ICRP 30	ICRP 30
NUCLIDES	ALI (uCi)	DAC (uCi/mL)	ALI (Bq)	ALI (uCi)	DAC (Bq/m^3)	DAC (uCi/mL)
I-131	2.00E+02	8.30E-08	6.00E+06	1.62E+02	7.00E+02	1.89E-08
I-132	1.00E+04	4.20E-06	6.00E+08	1.62E+04	1.00E+05	2.70E-06
I-133	9.00E+02	3.80E-07	3.00E+07	8.11E+02	4.00E+03	1.08E-07
I-134	5.00E+04	2.10E-05	2.00E+09	5.41E+04	7.00E+05	1.89E-05
I-135	4.00E+03	1.70E-06	2.00E+08	5.41E+03	2.00E+04	5.41E-07
Xe-133		1.00E-04				
Xe-133m		1.00E-04				
Xe-135		1.00E-05				
Xe-135m		9.00E-04				
Xe-138		4.00E-06				
Cs-134	1.00E+02	4.00E-08	4.00E+06	1.08E+02	2.00E+03	5.41E-08
Cs-136	7.00E+02	3.00E-07	2.00E+07	5.41E+02	1.00E+04	2.70E-07
Cs-137	2.00E+02	6.00E-08	6.00E+06	1.62E+02	2.00E+03	5.41E-08
Cs-138	6.00E+04	2.00E-05	2.00E+09	5.41E+04	9.00E+05	2.43E-05
Ba-140	1.00E+03	6.00E-07	5.00E+07	1.35E+03	2.00E+04	5.41E-07
La-140	1.00E+03	5.00E-07	4.00E+07	1.08E+03	2.00E+04	5.41E-07
Ce-141	6.00E+02	2.00E-07	2.00E+07	5.41E+02	9.00E+03	2.43E-07
Ce-143	2.00E+03	7.00E-07	6.00E+07	1.62E+03	2.00E+04	5.41E-07
Ce-144	1.00E+01	6.00E-09	5.00E+05	1.35E+01	2.00E+02	5.41E-09
Pr-143	7.00E+02	3.00E-07	3.00E+07	8.11E+02	1.00E+04	2.70E-07
Pr-144	1.00E+05	5.00E-05	4.00E+09	1.08E+05	2.00E+06	5.41E-05
Nd-147	8.00E+02	4.00E-07	3.00E+07	8.11E+02	1.00E+04	2.70E-07
Np-239	2.00E+03	9.00E-07	9.00E+07	2.43E+03	4.00E+04	1.08E-06
Pu-238	2.00E-02	8.00E-12	6.00E+02	1.62E-02	3.00E-01	8.11E-12
Pu-239	2.00E-02	8.30E-12	6.00E+02	1.62E-02	2.00E-01	5.41E-12
Pu-240	2.00E-02	8.30E-12	6.00E+02	1.62E-02	2.00E-01	5.41E-12
Pu-241	1.00E+00	4.20E-10	3.00E+04	8.11E-01	1.00E+01	2.70E-10
Am-241	1.00E-02	4.20E-12	4.00E+02	1.08E-02	8.00E-02	2.16E-12
Cm-242	3.00E-01	1.25E-10	1.00E+04	2.70E-01	4.00E+00	1.08E-10
Cm-244	2.00E-02	8.30E-12	7.00E+02	1.89E-02	2.00E-01	5.41E-12

Comparison of calculated ALI using DAC from 10 CFR Part 20 to the tabulated ALI in 10 CFR Part 20

NUCLIDES	Calculated ALI (uCi) from 10 CFR 20 DAC (uCi/mL)	10 CFR Part 20 ALI (uCi)
H-3	4.80E+04	8.00E+04
C-14	2.16E+05	2.00E+05
Cr-51	1.92E+04	2.00E+04
Mn-54	7.20E+02	8.00E+02
Mn-56	2.16E+04	2.00E+04
Co-58	7.20E+02	7.00E+02
Co-60	2.40E+01	3.00E+01
Fe-55	4.80E+03	4.00E+03
Fe-59	4.80E+02	5.00E+02
Zn-65	2.40E+02	3.00E+02
Br-84	7.20E+04	6.00E+04
Kr-85	2.40E+05	
Kr-85m	4.80E+04	
Kr-87	1.20E+04	
Kr-88	4.80E+03	
Rb-86	7.20E+02	8.00E+02
Rb-88	7.20E+04	6.00E+04
Rb-89	1.44E+05	1.00E+05
Sr-89	1.44E+02	1.00E+02
Sr-90	4.80E+00	4.00E+00
Sr-91	2.40E+03	4.00E+03
Sr-92	7.20E+03	7.00E+03
Y-90	7.20E+02	6.00E+02
Y-91	1.20E+02	1.00E+02
Y-92	7.20E+03	8.00E+03
Zr-95	2.40E+02	3.00E+02
Zr-97	1.20E+03	1.00E+03
Nb-95	1.20E+03	1.00E+03
Mo-99	1.44E+03	1.00E+03
Tc-99m	2.40E+05	2.00E+05
Ru-103	7.20E+02	6.00E+02
Ru-105	1.20E+04	1.00E+04
Ru-106	1.20E+01	1.00E+01
Rh-105	4.80E+03	6.00E+03
Ag-110m	9.60E+01	9.00E+01
Sb-122	9.60E+02	1.00E+03
Sb-124	2.40E+02	2.00E+02
Sb-125	4.80E+02	5.00E+02
Sb-127	9.60E+02	9.00E+02
Sb-129	9.60E+03	9.00E+03
Te-127	1.68E+04	2.00E+04
Te-127m	2.40E+02	3.00E+02
Te-129	7.20E+04	7.00E+04
Te-129m	2.40E+02	2.00E+02
Te-131m	9.00E+02	9.00E+02
Te-132	6.00E+02	6.00E+02
Te-134	5.04E+04	5.00E+04

Comparison of calculated ALI using DAC from 10 CFR Part 20 to the tabulated ALI in 10 CFR Part 20

NUCLIDES	Calculated ALI (uCi) from 10 CFR 20 DAC (uCi/mL)	10 CFR Part 20 ALI (uCi)
I-131	1.99E+02	2.00E+02
I-132	1.01E+04	1.00E+04
I-133	9.12E+02	9.00E+02
I-134	5.04E+04	5.00E+04
I-135	4.08E+03	4.00E+03
Xe-133	2.40E+05	
Xe-133m	2.40E+05	
Xe-135	2.40E+04	
Xe-135m	2.16E+06	
Xe-138	9.60E+03	
Cs-134	9.60E+01	1.00E+02
Cs-136	7.20E+02	7.00E+02
Cs-137	1.44E+02	2.00E+02
Cs-138	4.80E+04	6.00E+04
Ba-140	1.44E+03	1.00E+03
La-140	1.20E+03	1.00E+03
Ce-141	4.80E+02	6.00E+02
Ce-143	1.68E+03	2.00E+03
Ce-144	1.44E+01	1.00E+01
Pr-143	7.20E+02	7.00E+02
Pr-144	1.20E+05	1.00E+05
Nd-147	9.60E+02	8.00E+02
Np-239	2.16E+03	2.00E+03
Pu-238	1.92E-02	2.00E-02
Pu-239	1.99E-02	2.00E-02
Pu-240	1.99E-02	2.00E-02
Pu-241	1.01E+00	1.00E+00
Am-241	1.01E-02	1.00E-02
Cm-242	3.00E-01	3.00E-01
Cm-244	1.99E-02	2.00E-02

Data used to calculate CEDE for typical power plant radionuclides in SI units.

	Fed. Rep. No. 11	10 CFR Part 20	10 CFR Part 20	
Radionuclides	CEDE Con Fact. (Sv/Bq)	ALI (Bq)	DAC (Bq/m^3)	A (Bq)
H-3	1.73E-11	2.96E+09	7.40E+05	2.89E+08
C-14	6.36E-12	7.40E+09	3.33E+06	7.86E+08
Cr-51	9.03E-11	7.40E+08	2.96E+05	5.54E+07
Mn-54	1.81E-09	2.96E+07	1.11E+04	2.76E+06
Mn-56	8.91E-11	7.40E+08	3.33E+05	5.61E+07
Co-58	2.94E-09	2.59E+07	1.11E+04	1.70E+06
Co-60	5.91E-08	1.11E+06	3.70E+02	8.46E+04
Fe-55	3.61E-10	1.48E+08	7.40E+04	1.39E+07
Fe-59	3.30E-09	1.85E+07	7.40E+03	1.52E+06
Zn-65	5.51E-09	1.11E+07	3.70E+03	9.07E+05
Br-84	2.27E-11	2.22E+09	1.11E+06	2.20E+08
Kr-85			3.70E+06	
Kr-85m			7.40E+05	
Kr-87			1.85E+05	
Kr-88			7.40E+04	
Rb-86	1.79E-09	2.96E+07	1.11E+04	2.79E+06
Rb-88	2.26E-11	2.22E+09	1.11E+06	2.21E+08
Rb-89	1.16E-11	3.70E+09	2.22E+06	4.31E+08
Sr-89	1.12E-08	3.70E+06	2.22E+03	4.46E+05
Sr-90	3.51E-07	1.48E+05	7.40E+01	1.42E+04
Sr-91	4.49E-10	1.48E+08	3.70E+04	1.11E+07
Sr-92	2.18E-10	2.59E+08	1.11E+05	2.29E+07
Y-90	2.28E-09	2.22E+07	1.11E+04	2.19E+06
Y-91	1.32E-08	3.70E+06	1.85E+03	3.79E+05
Y-92	2.11E-10	2.96E+08	1.11E+05	2.37E+07
Zr-95	6.31E-09	1.11E+07	3.70E+03	7.92E+05
Zr-97	1.17E-09	3.70E+07	1.85E+04	4.27E+06
Nb-95	1.57E-09	3.70E+07	1.85E+04	3.18E+06
Mo-99	1.07E-09	3.70E+07	2.22E+04	4.67E+06
Tc-99m	7.21E-12	7.40E+09	3.70E+06	6.93E+08
Ru-103	2.42E-09	2.22E+07	1.11E+04	2.07E+06
Ru-105	1.23E-10	3.70E+08	1.85E+05	4.07E+07
Ru-106	1.29E-07	3.70E+05	1.85E+02	3.88E+04
Rh-105	2.58E-10	2.22E+08	7.40E+04	1.94E+07
Ag-110m	2.17E-08	3.33E+06	1.48E+03	2.30E+05
Sb-122	1.39E-09	3.70E+07	1.48E+04	3.60E+06
Sb-124	6.80E-09	7.40E+06	3.70E+03	7.35E+05
Sb-125	3.30E-09	1.85E+07	7.40E+03	1.52E+06
Sb-127	1.63E-09	3.33E+07	1.48E+04	3.07E+06
Sb-129	1.74E-10	3.33E+08	1.48E+05	2.87E+07
Te-127	8.60E-11	7.40E+08	2.59E+05	5.81E+07
Te-127m	5.81E-09	1.11E+07	3.70E+03	8.61E+05
Te-129	2.09E-11	2.59E+09	1.11E+06	2.39E+08
Te-129m	6.47E-09	7.40E+06	3.70E+03	7.73E+05
Te-131m	1.73E-09	3.33E+07	1.39E+04	2.89E+06
Te-132	2.55E-09	2.22E+07	9.25E+03	1.96E+06
Te-134	3.23E-11	1.85E+09	7.77E+05	1.55E+08

Data used to calculate CEDE for typical power plant radionuclides in SI units.

	Fed. Rep. No. 11	10 CFR Part 20	10 CFR Part 20		
Radionuclides	CEDE Con Fact. (Sv/Bq)	ALI (Bq)	DAC (Bq/m^3)	A (Bq)	
I-131	8.89E-09	7.40E+06	3.07E+03	5.62E+05	
I-132	1.03E-10	3.70E+08	1.55E+05	4.85E+07	
I-133	1.58E-09	3.33E+07	1.41E+04	3.16E+06	
I-134	3.55E-11	1.85E+09	7.77E+05	1.41E+08	
I-135	3.32E-10	1.48E+08	6.29E+04	1.51E+07	
Xe-133			3.70E+06		
Xe-133m			3.70E+06		
Xe-135			3.70E+05		
Xe-135m			3.33E+07		
Xe-138			1.48E+05		
Cs-134	1.25E-08	3.70E+06	1.48E+03	4.00E+05	
Cs-136	1.98E-09	2.59E+07	1.11E+04	2.53E+06	
Cs-137	8.63E-09	7.40E+06	2.22E+03	5.79E+05	
Cs-138	2.74E-11	2.22E+09	7.40E+05	1.82E+08	
Ba-140	1.01E-09	3.70E+07	2.22E+04	4.95E+06	
La-140	1.31E-09	3.70E+07	1.85E+04	3.82E+06	
Ce-141	2.42E-09	2.22E+07	7.40E+03	2.07E+06	
Ce-143	9.16E-10	7.40E+07	2.59E+04	5.46E+06	
Ce-144	1.01E-07	3.70E+05	2.22E+02	4.95E+04	
Pr-143	2.19E-09	2.59E+07	1.11E+04	2.28E+06	
Pr-144	1.17E-11	3.70E+09	1.85E+06	4.27E+08	
Nd-147	1.85E-09	2.96E+07	1.48E+04	2.70E+06	
Np-239	6.78E-10	7.40E+07	3.33E+04	7.37E+06	
Pu-238	7.79E-05	7.40E+02	2.96E-01	6.42E+01	
Pu-239	8.33E-05	7.40E+02	3.07E-01	6.00E+01	
Pu-240	8.33E-05	7.40E+02	3.07E-01	6.00E+01	
Pu-241	1.34E-06	3.70E+04	1.55E+01	3.73E+03	
Am-241	1.20E-04	3.70E+02	1.55E-01	4.17E+01	
Cm-242	4.67E-06	1.11E+04	4.63E+00	1.07E+03	
Cm-244	6.70E-05	7.40E+02	3.07E-01	7.46E+01	

Comparison of CEDE per unit activity from EPA Federal Report No. 11and ICRP 30 along with comparison of ALI calculated using ICRP 30 with 10 CFR 20 ALI

	Fed. Rep. No. 11	ICRP Publication 30	ALI (uCi) calc. using ICRP 30	10 CFR Part 20
NUCLIDES	CEDE Con Fact. (Sv/Bq)	CEDE Con Fact (Sv/Bq)	CEDE Conv Factor	ALI (uCi)
H-3	1.73E-11	1.70E-11	7.95E+04	8.00E+04
C-14	6.36E-12			2.00E+05
Cr-51	9.03E-11	7.11E-11	1.90E+04	2.00E+04
Mn-54	1.81E-09	1.72E-09	7.86E+02	8.00E+02
Mn-56	8.91E-11	6.40E-11	2.11E+04	2.00E+04
Co-58	2.94E-09	1.90E-09	7.11E+02	7.00E+02
Co-60	5.91E-08	4.10E-08	3.30E+01	3.00E+01
Fe-55	3.61E-10	3.31E-10	4.08E+03	4.00E+03
Fe-59	3.30E-09	2.69E-09	5.02E+02	5.00E+02
Zn-65	5.51E-09	5.00E-09	2.70E+02	3.00E+02
Br-84	2.27E-11	2.10E-11	6.44E+04	6.00E+04
Kr-85				
Kr-85m				
Kr-87				
Kr-88				
Rb-86	1.79E-09	1.81E-09	7.47E+02	8.00E+02
Rb-88	2.26E-11	2.17E-11	6.23E+04	6.00E+04
Rb-89	1.16E-11	1.00E-11	1.35E+05	1.00E+05
Sr-89	1.12E-08	1.15E-08	1.18E+02	1.00E+02
Sr-90	3.51E-07	3.40E-07	3.97E+00	4.00E+00
Sr-91	4.49E-10	3.95E-10	3.42E+03	4.00E+03
Sr-92	2.18E-10	2.07E-10	6.53E+03	7.00E+03
Y-90	2.28E-09	2.18E-09	6.20E+02	6.00E+02
Y-91	1.32E-08	1.20E-08	1.13E+02	1.00E+02
Y-92	2.11E-10	1.74E-10	7.77E+03	8.00E+03
Zr-95	6.31E-09	4.90E-09	2.76E+02	3.00E+02
Zr-97	1.17E-09	1.07E-09	1.26E+03	1.00E+03
Nb-95	1.57E-09	1.23E-10	1.10E+04	1.00E+03
Mo-99	1.07E-09	9.90E-10	1.37E+03	1.00E+03
Tc-99m	7.21E-12	5.67E-12	2.38E+05	2.00E+05
Ru-103	2.42E-09	2.09E-09	6.47E+02	6.00E+02
Ru-105	1.23E-10	1.12E-10	1.21E+04	1.00E+04
Ru-106	1.29E-07	1.20E-07	1.13E+01	1.00E+01
Rh-105	2.58E-10	2.31E-10	5.85E+03	6.00E+03
Ag-110m	2.17E-08	1.40E-08	9.65E+01	9.00E+01
Sb-122	1.39E-09	1.27E-09	1.06E+03	1.00E+03
Sb-124	6.80E-09	5.65E-09	2.39E+02	2.00E+02
Sb-125	3.30E-09	2.60E-09	5.20E+02	5.00E+02
Sb-127	1.63E-09	1.46E-09	9.26E+02	9.00E+02
Sb-129	1.74E-10	1.57E-10	8.61E+03	9.00E+03
Te-127	8.60E-11	7.80E-11	1.73E+04	2.00E+04
Te-127m	5.81E-09	5.25E-09	2.57E+02	3.00E+02
Te-129	2.09E-11	1.80E-11	7.51E+04	7.00E+04
Te-129m	6.47E-09	5.46E-09	2.48E+02	2.00E+02
Te-131m	1.73E-09	1.51E-09	8.95E+02	9.00E+02
Te-132	2.55E-09	2.10E-09	6.44E+02	6.00E+02
Te-134	3.23E-11	2.68E-11	5.04E+04	5.00E+04

Comparison of CEDE per unit activity from EPA Federal Report No. 11and ICRP 30 along with comparison of ALI calculated using ICRP 30 with 10 CFR 20 ALI

	Fed. Rep. No. 11	ICRP Publication 30	ALI (uCi) calc. using ICRP 30	10 CFR Part 20
NUCLIDES	CEDE Con Fact. (Sv/Bq)	CEDE Con Fact (Sv/Bq)	CEDE Conv Factor	ALI (uCi)
I-131	8.89E-09	8.80E-09	1.54E+02	2.00E+02
I-132	1.03E-10	9.10E-11	1.49E+04	1.00E+04
I-133	1.58E-09	1.50E-09	9.01E+02	9.00E+02
I-134	3.55E-11	2.99E-11	4.52E+04	5.00E+04
I-135	3.32E-10	3.03E-10	4.46E+03	4.00E+03
Xe-133				
Xe-133m				
Xe-135				
Xe-135m				
Xe-138				
Cs-134	1.25E-08	1.25E-08	1.08E+02	1.00E+02
Cs-136	1.98E-09	2.01E-09	6.72E+02	7.00E+02
Cs-137	8.63E-09	8.70E-09	1.55E+02	2.00E+02
Cs-138	2.74E-11	2.36E-11	5.73E+04	6.00E+04
Ba-140	1.01E-09	9.65E-10	1.40E+03	1.00E+03
La-140	1.31E-09	1.17E-09	1.16E+03	1.00E+03
Ce-141	2.42E-09	2.25E-09	6.01E+02	6.00E+02
Ce-143	9.16E-10	8.50E-10	1.59E+03	2.00E+03
Ce-144	1.01E-07	9.50E-08	1.42E+01	1.00E+01
Pr-143	2.19E-09	2.01E-09	6.72E+02	7.00E+02
Pr-144	1.17E-11	1.10E-11	1.23E+05	1.00E+05
Nd-147	1.85E-09	1.65E-09	8.19E+02	8.00E+02
Np-239	6.78E-10	5.68E-10	2.38E+03	2.00E+03
Pu-238	7.79E-05	8.20E-05	1.65E-02	2.00E-02
Pu-239	8.33E-05	8.91E-05	1.52E-02	2.00E-02
Pu-240	8.33E-05	8.91E-05	1.52E-02	2.00E-02
Pu-241	1.34E-06	1.56E-06	8.66E-01	1.00E+00
Am-241	1.20E-04	1.41E-04	9.58E-03	1.00E-02
Cm-242	4.67E-06	4.66E-06	2.90E-01	3.00E-01
Cm-244	6.70E-05	7.40E-05	1.83E-02	2.00E-02

Prospective Evaluation of the Need for Internal Monitoring

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Abstract

Under the revision of 10 CFR 20, workers must be monitored for internal dose only if a prospective evaluation shows that they are likely to exceed 10 percent of an ALI in a year. Past positive whole body counts were reviewed at the Perry Nuclear Power Plant, and the largest uptake was found to be 1.3 percent of an ALI. Past RWPs which had the potential for significant airborne exposure were identified and reviewed. The highest possible uptake was calculated to be 2.5 percent of an ALI, not taking credit for respiratory protection. Committed dose from alpha and pure beta emitters which would not be identified by gamma-sensitive bioassay was found to be negligible. Based on this prospective evaluation, monitoring personnel for internal dose is not required at this facility.

Key Words

dosimetry, internal
evaluations
respiratory protection
whole body counting

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INTRODUCTION

Under the revision of 10 CFR 20, personnel must be monitored for internal dose if they are likely to receive internal doses in excess of 10 percent of an Annual Limit on Intake (ALI). The decision whether or not to monitor is to be based on a prospective evaluation.

This paper presents the regulatory framework and technical justifications which were used as the basis for the prospective evaluation relative to internal monitoring at the Perry Nuclear Power Plant (PNPP).

REGULATORY BASES

The revised version of 10 CFR 20 states in paragraph 20.1502, in part, that "Each licensee shall monitor exposures to...

> (b)(1) Adults likely to receive, in 1 year, an intake in excess of 10 percent of the applicable ALI(s)..."

In the Supplementary Information section of the Federal Register Notice for the revised Part 20, the NRC states:

> "The monitoring threshold is a predetermined level of anticipated dose for carrying out bioassay procedures and does not represent a required level of detection sensitivity. If, by a reasonable analysis of the working environment, it appears that a worker is likely to inhale radioactive materials at concentrations that could produce an annual committed effective dose equivalent of 0.5 rem (10 percent of the

5-rem limit) or more, then that worker's intake should be monitored using measurements of exposure (e.g., estimates of DAC-hours based upon measured air concentrations) or intake (such as by whole-body counting or other bioassay technique) or by measurements of both exposure and intake. Whether the actual doses received were in excess of 10 percent of the limits could only be determined from these subsequent measurements. The TEDE has to be evaluated if both the internal and external dose components have to be monitored."

Further clarification of the need to monitor for internal dose was provided by the NRC (NRC91), as follows:

> "The results of bioassays alone cannot be used to determine if the licensee must monitor internal exposures or sum internal and external dose under 10 CFR Part 20. Monitoring for internal is required for adults 'likely to receive' in a year an intake greater than 10 percent of the limit. Determination of what an individual is likely to receive is a prospective assessment of intake. Bioassay is a retrospective assessment of intake. Future intakes are not necessarily the same as past intakes. However, bioassay data may be used together with other information as a basis for the prospective intake assessment. For example, if the uses of radioactive materials in a facility are not going to change significantly and bioassays of individuals employed in the facility have shown that no one has ever received an intake greater than 10 percent, then one might reasonably conclude that no one is 'likely to receive' an intake in excess of 10 percent of the limit."

DISCUSSION

The decision whether or not "monitoring" is required impacts several portions of radiation protection programs in nuclear power plants. If workers are not required to be monitored because a prospective evaluation showed workers were not likely to receive in excess of 10 percent of an ALI in a year, then:

a) Many of the routine whole body counts (WBCs) need not be performed.

b) Extensive investigations of internal exposure events, which identify inconsequentially low-levels of internal contamination, may be discontinued. This would free up valuable Health Physics manpower for more important duties.

c) Insignificant internal doses are not required to be summed with external dose. Extensive and expensive modification to dosimetry software may be avoided.

In order to determine whether or not personnel are likely to receive internal exposures of 10 percent of an ALI at PNPP, two separate evaluations were performed. First, all positive WBCs in the history of plant operation were examined and the maximum internal exposure cases identified. This exercise resulted in identifying the maximum fractional ALI which has been incurred by a worker to date at PNPP.

Second, radiological work which exhibited the potential for significant internal exposure (even though none may have occurred) was identified. The highest potential internal exposures were then calculated for these work activities with no credit taken for respiratory protection (even though respirators were in fact used in most cases).

The endpoint of both of these evaluations was the fraction of an ALI to which workers were, or could have been exposed, at PNPP.

EVALUATION OF PAST INTERNAL EXPOSURES

Assessment of internal exposure at Perry has historically been accomplished by WBC. WBCs have been performed under four conditions:

Table 1: Maximum Internal Contamination Events at PNPP

Nuclide	Annual Limit on Intake[1] (μCi)		Maximum Activity (nCi)	Fractional ALI
	Ingestion	Inhalation		
Cr-51	4E+4	2E+4	144	7E-6
Co-58	1E+3	7E+2	34.6	5E-5
Co-60	2E+2	3E+1	172	6E-3
Cs-134	7E+1	1E+2	2.11	3E-5
Cs-137	1E+2	2E+2	16.9	2E-4
Fe-59	8E+2	3E+2	15.5	5E-5
I-131[2]	3E+1	5E+1	35.2	1E-3
Mn-54	2E+3	8E+2	64.0	8E-5
Sb-124[3]	5E+2	2E+2	0.067	3E-7
Zn-65	4E+2	3E+2	257	9E-4
Zr-95	1E+3	1E+2	12.9	1E-4

[1] ALIs listed are the most restrictive values for the thyroid, and the fractional ALI is calculated based upon the ingestion pathway.

[2] I-131 was identified during an entrance whole body count; this worker had an uptake from another facility.

[3] Sb-124 uptake was most probably an analytical artifact. This whole body count was performed prior to significant power operation and this nuclide has not been identified since this one occurrence.

a) Worker Entrance - prior to allowing access of a new worker to the Radiologically Restricted Area.

b) Worker Exit - upon termination of worker employment.

c) Routine Annual - annually for all workers

d) Special - when internal exposure is known or suspected to have occurred. Special WBCs have been performed following the detection of facial/nasal contamination, entrance of a worker into an airborne contamination area, medical uptakes, and otherwise at the discretion of Health Physics Supervision.

There have been over 100 instances of individuals who have been confirmed by WBC to have internal contamination (other than medical) since PNPP commenced operation in 1986. In order to evaluate these internal exposures relative to the revised Part 20, these WBC results were examined in detail. A total of 11 radionuclides were identified as internal contaminants from this data. Each of these radionuclides, along with its respective limiting

Allowable Limit on Intake (ALI) for both ingestion and inhalation is listed in Table 1. Except for the I-131 (entrance WBC) and the Sb-124 (analytical artifact) cases, all other positive WBCs were performed very soon after the contaminating event (i.e., less than 2 hours). Also presented in Table 1 is the activity detected in the individual maximally exposed to each radionuclide and the fractional ALI calculated using the most conservative ALI.

The case of the individual internally contaminated with Co-60 was found to represent the maximally exposed individual in the history of plant operation. In addition to 172 nCi of Co-60, this individual's internal contamination included 15.1, 40.5, 160 and 2.1 nCi of Co-58, Mn-54, Zn-65 and Cs-134, respectively. This internal contamination (like essentially all others at PNPP) was cleared from the individual with an observed retention function which indicated GI tract clearance, hence the event was treated as an ingestion. A summation of the fractional ingestion ALIs yielded 1.3E-2 (1.3 percent) ALI. Although internal contamination measured by WBC doesn't reflect intake in the true technical sense, it is close enough for purposes of this evaluation since the WBC was conducted very soon after the event.

Based on this historical data, PNPP workers have not been exposed to radioactive materials producing internal contamination in excess of 10 percent of an ALI.

EVALUATION OF MAXIMUM POSSIBLE INTERNAL EXPOSURE

Work activities in the period of 1989 through 1991 were evaluated to identify the maximum internal exposures which could have occurred if no respiratory protection had been used. Several methods were used to identify these "high airborne" jobs:

a) The PNPP Radiation Protection Dosimetry Information System database was sorted to identify Radiation Work Permits (RWPs) from 1989 through 1991 under which airborne exposures had occurred. MPC-hr postings of greater than 0.2 MPC-hr to individual worker dosimetry files were used as a search criterion.

b) ALARA Reviews performed during job planning from 1989 through 1991 were examined to identify RWPs in which significant airborne activity may have been present.

c) Health Physics Shift Supervisors were polled to identify any jobs which they felt met the search criteria, i.e., potential for significant internal dose assuming respirators had failed or not been used.

These three independent methods were used to identify a set of RWPs under which significant airborne exposure did or could have occurred. Each RWP was examined to identify plant personnel who entered the airborne area, with or without respiratory protection. MPC-hr calculations were obtained from the dosimetry files of the 115 personnel identified by this process.

The MPC-hr calculations were further examined, and 66 personnel were identified as having entered an area, sometime during the three-year period, in which the airborne concentration was greater than 1 MPC. For each of these 66 personnel, all entries into airborne areas in the three-year period were summarized such that MPC-hrs of exposure could be calculated as if each person had made all entries without any respiratory protection. Table 2 presents the frequency distribution of airborne exposures from this evaluation.

The highest calculated potential airborne concentration in the three-year period was 62.6 MPC-hrs; 59.5 MPC-hrs of which was from a single job. To convert this potential exposure to DAC-hrs, the air sample analysis

Table 2: Distribution of Potential Internal Exposure

Total MPC-hr	# of Cases
< 10	48
10-20	7
20-30	2
30-40	2
40-50	2
50-60	3
60-70	2
> 70	0

Note that a screening criteria was used, rejecting individuals who had not at any time during the 3 year period entered an area with airborne radioactive materials at greater than 1 MPC concentration. Note also that Total MPC-hr is the sum of the airborne exposure to each individual for the entire 3 year period, taking no credit for respiratory protection.

data were obtained from the RWP. The most-limiting-case DACs were used to calculate the DAC-hrs of exposure for the maximally exposed individual. The results of these calculations are presented in Table 3.

ALPHA AND PURE BETA EMITTERS

Not all radionuclides present in plant contamination to which a worker may be exposed are strong gamma emitters, hence they may not be identified during air sample or WBC data analysis. Although it is unrealistic to postulate exposure to mixtures of nuclides which are pure beta and/or alpha emitters only, it is important to understand what fraction of the total committed dose following an uptake is due to nuclides which are (theoretically) present but which will not in fact be identified during air sample (gamma spectrum) analysis or whole body counting.

In order to evaluate the significance relative to internal dose of alpha and pure beta emitters expected to be present at PNPP, the results of a sample analyzed for 10 CFR 61 dry active waste (DAW) compliance was used. Note that 10 CFR 61 waste stream sample analysis

Table 3: Internal Dose to Maximally Exposed Individual

Nuclide	Observed µCi/cc	DAC µCi/cc	Fractional DAC
Co-58	2.12E-8	3E-7	0.07
Co-60	1.41E-7	1E-8	14.1
Cr-51	2.59E-7	1E-5	0.03
Fe-59	1.76E-8	1E-7	0.18
Mn-54	7.58E-8	3E-7	0.25
Zn-65	8.67E-9	1E-7	0.09
Total DAC			14.7
Total MPC			18.6

Note that the maximally exposed individual was exposed for 3.2 hours to a total of 47.0 DAC-hrs. This individual was also exposed during other work to 1.0 MPC for 2.8 hours and 0.78 MPC for 0.4 hours. Using the ratio of 14.7 DAC to 18.6 MPC to scale these exposures, this individual's total potential dose is calculated to be 49.5 DAC-hrs.

results may represent unrealistic worst-case mixtures of radionuclides to which workers may be exposed, since:

a) 10 CFR 61 samples generally represent longer-lived materials which are concentrated by systems such as demineralizers, over weeks or months of operation.

b) The time from sample collection to sample analysis is commonly fairly long (weeks to months).

These factors lead to mixtures of nuclides which are enhanced in the longer-lived alpha and pure beta emitters relative to typical plant contamination, which may have a much higher fraction of shorter-lived, gamma-emitting fission and activation products.

Table 4 lists the observed concentrations of all identified radionuclides in a dry active waste (DAW) stream sample from PNPP. Also listed are the ingestion ALIs for each radionuclide, followed by the calculated fractional ALI represented by each nuclide in the sample. The total fractional ALI of the sample is shown to be 1.8E-3, and the total fractional ALI due to nuclides which are easily detectable gamma emitters is calculated to be 1.73E-3, about 96 percent of the total committed dose. Analysis of this concentration data using inhalation ALIs leads to a result of about 93 percent of the total committed dose attributed to easily detectable gamma emitters.

This demonstration of "detectability" is necessarily plant specific. The use of 10 CFR 61 sample analysis results permits a simple worst-case (unrealistically conservative) evaluation of potential alpha and pure beta contributions to internal dose. Some plants, due to efficiency and operation cycles of cleanup systems, fission and activation product source terms and other factors, will find that 10 CFR 61 sample results are so conservative as to be limiting, and that another type of demonstration of "detectability" is warranted.

CONCLUSIONS AND RECOMMENDATIONS

The maximally exposed individual in the history of PNPP operation received an ingestion uptake equivalent to 1.3 percent of an ALI.

Based on evaluation of RWP data from 1989 through 1991, the maximum internal dose which could have occurred at PNPP if respirators had failed completely (or equivalently are not worn in the future) was 49.5 DAC-hrs, equivalent to about 2.5 percent of an ALI. The frequency distribution of potential internal exposures further demonstrates that if respiratory protection is not used or fails, significant internal doses will not result for the types and concentrations of materials encountered at PNPP.

The committed dose due to alpha and pure beta emitters which may not be easily identifiable is small (less than about 7 percent) relative to the committed dose due to gamma emitters.

Based on this prospective evaluation, it is not likely that an individual will receive an internal dose at PNPP in excess of 10 percent of an ALI, even with no credit taken for respiratory protection. As such, monitoring for internal dose under 10 CFR 20.1502 is not required.

PNPP is not required to "monitor" for internal dose, and as such routine annual and many of the non-routine WBCs currently required by plant procedures may be discontinued. If the bulk of the currently required WBCs are discontinued, PNPP will most probably still conduct employment entrance and exit WBCs on all personnel, as a matter of simple legal protection. Additionally, non-routine whole body counts may be performed under the following circumstances:

a) When individuals cause portal monitor alarms at the egress point which cannot be attributed to external contamination (see article "Passive Internal Monitoring Program" in this issue of *RPM*).

Table 4: 10 CFR 61 Sample - Ingestion Dose Analysis

Nuclide	Concentration µCi/gm	ALI (µCi)	Fractional ALI
H-3	5.1E-4	8E+4	6.4E-9
C-14	2.2E-3	2E+3	1.1E-6
Mn-54[1]	4.9E-2	2E+3	2.5E-5
Fe-55	4.6E-1	9E+3	5.1E-5
Co-58[1]	1.6E-2	1E+3	1.6E-5
Co-60[1]	1.8E-1	2E+2	9.0E-4
Ni-63	4.1E-3	9E+3	4.6E-7
Zn-65[1]	2.7E-1	4E+2	6.8E-4
Sr-90	9.6E-5	3E+1	3.2E-6
Ag-110m	5.4E-3	5E+2	1.1E-5
Cs-134[1]	3.9E-3	7E+1	5.6E-5
Cs-137[1]	5.4E-3	1E+2	5.4E-5
Ce-141	2.4E-4	2E+3	1.2E-7
Ce-144	1.2E-4	2E+2	6.0E-7
Pu-238	4.2E-7	9E-1	4.7E-7
Pu-239,240	3.4E-7	8E-1	4.3E-7
Pu-241	2.7E-5	4E+1	6.8E-7
Am-241	7.9E-8	8E-1	9.9E-8
Cm-242	1.5E-6	3E+1	5.0E-8
Cm-243,244	6.9E-8	1E+0	6.9E-8
Total			1.8E-3

[1] The sum of the fractional ALIs for the gamma emitting (easily detectable) nuclides is 1.73E-3, i.e., 96 percent of the committed dose is caused by easily detectable nuclides. Analysis of the above data using inhalation ALIs leads to a corresponding result of 93 percent of the committed dose due to easily detectable nuclides.

b) When an individual begins employment at PNPP with significant internal contamination from another facility.

c) At the specific direction of the Plant Health Physicist. This may happen when there is reason to believe a significant internal exposure may have occurred, e.g., following significant facial contamination or following multiple alarms on whole body contamination monitors at exit of the radiological restricted area.

d) Following a formal declaration of pregnancy by a female worker, in accordance with 10 CFR 20.1502(b)(2).

If, based on this evaluation, management adopts the position that formal "monitoring" for internal dose is not required and will not be performed, then this type of prospective evaluation should be performed on an annual or fuel cycle basis. Each subsequent evaluation should account for any significant changes in work activities or radionuclide distributions.

References

FR91 "Standards for Protection Against Radiation; Final Rule," Federal Register Vol. 56, No. 98, Tuesday, May 21, 1991.

NRC91 "First Set of Questions and Answers on the New Part 20," prepared by the NRC and distributed through NUMARC in January, 1991.

The Authors

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ATTACHMENT 1

ROCHESTER GAS AND ELECTRIC CORPORATION
R.E. GINNA NUCLEAR POWER PLANT
UFSAR CHANGE NOTICE
No. ___/_____

1. Applicable Section(s)
   12.5.2.1, 12.5.2.2, 12.5.2.4, 12.5.3.2, 12.5.3.3, 12.5.4, Tab./Fig 12.5-1, 13.5

2. Reference(s)
   Revised 10CFR20 §§ 20.1001 - 20.2401

3. Justification (attach additional sheets if necessary)
   These changes are to align the UFSAR applicable sections with the revised 10CFR20 §§ 20.1001 - 20.2401 effective June 20, 1991 and implementation January 1, 1994

4. 10CFR50.59 Screening Form Attached: YES [✓] , EXEMPT []
   Safety Evaluation Reference, if applicable (Exempt Only):

   Number __________ Title __________________________

Originator William H Thomson date 4/12/93

Section Manager ______________________ date __________

(Forward to UFSAR Coordinator if Exempt)

Operations Manager[1] ______________________ date __________

Technical Manager[2] ______________________ date __________

Manager, Nuclear Safety & Licensing[2] ______________________ date __________

UFSAR Coordinator (NS&L) ______________________ date __________
[] Include in UFSAR [] Exclude (See Reason for Exclusion below)

Reason for Exclusion:
(UFSAR Coordinator)

Note 1 - required if change relates to or affects plant operation
Note 2 - required if change results in new safety evaluation

ATTACHMENT 2

No.____/_________

UFSAR CHANGE 10CFR50.59 SCREENING CRITERIA
ROCHESTER GAS AND ELECTRIC CORPORATION

NOTE: A completed copy shall be included with the UFSAR Change Notice for non-exempt changes.

The following set of questions shall be used to screen UFSAR changes which are not the result of an EWR, TSR, PCN, TSEE or NCR and to determine if the change requires a 10CFR50.59 safety evaluation. Document the basis for the answer to each question in the space provided.

NOTE: If there is a doubt as to the answer to a question, answer it conservatively so as to continue processing the form.

1. Is the change an inconsequential change as listed in the following examples?

YES	NO		
[]	[✓]	a)	Clarifications or additions such as: identification of information, lists, references, figures or procedures explicitly described or referenced in another UFSAR section
[]	[✓]	b)	Addition of notes that add specificity or clarity
[]	[✓]	c)	Addition of information that adds conservatism and does not change intent
[]	[✓]	d)	Including new revisions of P&IDs or Electrical, Instrumentation and Control Drawings
[]	[✓]	e)	Movement of text from one section to another

If YES, sign and date the form as PREPARED BY, and submit with the UFSAR Change Notice.

If NO, continue with QUESTION 2

ATTACHMENT 2 (cont)

2. Does this change result in non-compliance with any Technical Specification?

[ ]YES If YES, this change will require a Tech Spec Amendment in accordance with 10CFR50.90 and NRC approval prior to implementation of the change. List the affected TS Section, sign and date the form AS PREPARED BY, and submit with the UFSAR Change Notice.

TS Section affected ____________________

[✓]NO If NO, provide the basis for this conclusion and continue to QUESTION 3.

Basis:

3. Does the change alter the design, function or method of performing a function of a structure, system or component described in the UFSAR; degrade its operational capability; or create the potential for operation of plant equipment outside its design parameters or a configuration not originally intended?

[ ]YES If YES, proceed directly to QUESTION 5.

[✓]NO If NO, provide the basis for this conclusion and continue with QUESTION 4.

Basis:

4. Would the change alter a statement involving a commitment in the UFSAR?

[ ]YES If YES, proceed to QUESTION 5.

[✓]NO If NO, sign and date the form as PREPARED BY and submit with the UFSAR Change Notice.

5. A 10CFR50.59 Safety Evaluation is required. Prepare a written safety evaluation using A-303. Submit the safety evaluation with the UFSAR Change Notice.

Prepared by: William H Thomson Date: 4/12/93

SECTION F - COMPLETED SPECIAL TESTS (ST) AND EXPERIMENTS

This section contains a description of special tests and experiments performed in the facility, pursuant to the requirements of 10 CFR 50.59(b).

ROCHESTER GAS AND ELECTRIC CORPORATION

89 EAST AVENUE

ROCHESTER, NEW YORK 14649

GINNA STATION

SAFETY ANALYSIS

FOR

SPECIAL TEST

ST-93-002

REVISION 0

December 6, 1993

SYSTEMS OPERATING PARAMETERS

FOR A AND B BORIC ACID TRANSFER PUMPS

PREPARED BY: Charles M Rioch — 12-6-93
Responsible Staff Engineer — Date

REVIEWED BY: Peter Baumford — 12/8/93
Reactor Engineer — Date

REVIEWED BY: [illegible] — 12/13/93
Technical Manager — Date

APPROVED BY: [illegible] — 12/15/93
Chairman, Plant Operations Review Committee — Date

1.0 SCOPE OF ANALYSIS:

1.1 During the last year there have been three failures of the A Boric Acid Transfer Pump. The purpose of this test procedure is to measure and to document actual system operating data for both A and B Boric Acid Transfer Pumps. Data collected during the conduct of this procedure will be compared to historical data and existing pump curves. Additionally, the information will be used as an aid in Root Cause Investigation M-93-016, "Repetitive Pump A Failures". This analysis will evaluate the safety consequences of performing the test at power.

2.0 REFERENCES:

2.1 Special Test Procedure ST-93.002

2.2 RG&E Drawing 33013-1266

2.3 R.E. Ginna Technical Specifications, sections 3.2.2.c, 3.3.1.1.j and 3.3.1.6

3.0 STRUCTURES, SYSTEMS AND COMPONENTS AFFECTS (SSC)

3.1 The components affected by this procedure are the A and B Boric Acid Transfer Pumps and their associated recirculation piping and valves.

4.0 SAFETY FUNCTION OF AFFECTED SSC'S

4.1 The Boric Acid Transfer Pumps provide the motive force required to move borated water from the Storage Tanks to the charging pump suction via the blender and emergency borate flowpaths.

5.0 EFFECTS ON SAFETY:

5.1 This procedure will attach pressure gauges/transmitters downstream of the system sample valves located on both suction and discharge of A and B Boric Acid transfer pumps. At no time will the flowpath to the blender or emergency boration be isolated.

5.2 A qualified auxiliary operator will be employed to make all valve lineup changes directed in the procedure, and will be stationed at the pressure gauges for the duration of the test.

5.3 Plastic tubing will be used to connect the pressure instrumentation downstream of valves 330A/B and 346A/B. The instrumentation is considered decoupled from the system due to material, size of tubing and the fact that all instrumentation will not be supported by the tubing connection, and therefore its weight need not be analyzed.

5.4 The likelihood of an increase in leakage of liquid effluent is not increased because:

a) The Auxiliary Operator can immediately isolate the tubing from the system by closing the sample valve.

b) The tubing is rated at a burst pressure of 600 psig at a conservative upper temperature of 175°F, both above the system operating parameters of 160°F and 90 psig with heat trace on and the transfer pump(s) in operation.

5.5 During a majority of the testing, the system will be operated in its normal line up for sampling with the pumps running in recirculation back to the Storage Tanks. Where pump suction and discharge paths deviate from the normal lineup, an Auxiliary Operator will be stationed to return the system to normal, should the need arise.

6.0 UNREVIEWED SAFETY QUESTION CONCLUSIONS

6.1 The proposed procedure will not increase the probability of occurrence of an accident previously evaluated in the UFSAR because the system design parameters will not be exceeded.

6.2 The proposed procedure will not increase the consequences of an accident previously evaluated in the UFSAR. The system will remain in service in its normal at power alignment and therefore be able to respond to accident conditions.

6.3 The proposed procedure will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR because the system will be operated in its normal at-power alignment.

6.4 The proposed procedure will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR because the design and operational requirements will be met during the test.

6.5 The proposed procedure will not create a possibility for an accident of a different type than any evaluated previously in the UFSAR since the temporary equipment meets the system design conditions and the system function and performance will not be affected.

6.6 The temporary changes proposed in the evaluation will not create a possibility for a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR since the temporary equipment meet the system design conditions and the system function and performance will not be affected.

6.7 The proposed installation will not reduce any margin of safety as defined in the basis of any Technical Specification because all actions will be in accordance with Technical Specification requirements.

7.0 <u>CONCLUSION</u>

7.1 This procedure involves an operational evolution which will not compromise system integrity and will be accomplished in a time frame which will not jeopardize plant safety. Thus this installation does not constitute an unreviewed safety question. It does not result in a change to the Technical Specifications.

CONCLUSIONS:

All of the above were reviewed by the PORC committee with respect to the Technical Specifications and the committee has determined that no Technical Specification changes or violations were involved.

Additionally, these changes were reviewed in committee to determine if they presented an Unreviewed Safety Question and the general summations of these reviews are as follows:

1. These changes do not increase the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR, because:

   These changes were performed to ensure continued operability/availability of plant equipment and will not result in any equipment being operated outside of its normal operating range. This results in continued operability/availability of equipment important to safety. These changes additionally will not result in a change of operating characteristics of equipment used in the transient/accident mitigation which precludes an increase in the probability of occurrence of an accident. Because these changes ensure continued availability of plant equipment, the limits shown in the Technical Specifications, and the assumptions of the safety analyses of the Updated Final Safety Analysis Report continue to be met. As a result there is no increase in the consequences of any presently postulated accident.

2. These changes do not create the possibility for a new or different kind of accident, or a malfunction of a different type from any accident previously evaluated in the UFSAR because:

   The changes do not present new failure mechanisms outside of those presently anticipated, and are bounded by the events contained in the Updated Final Safety Analysis Report.

3. The changes do not reduce the margin of safety because:

   Present margins as contained in the Technical Specifications are valid, and these changes are performed within those limits. These changes will not result in violating the baseline assumptions made for equipment availability in the Technical Specifications and the Updated Final Safety Analysis Report.