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PUBLIC S[illegible]ICE COMPANY OF OKLAHOMA
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PSO

Public Service Company of Oklahoma
Black Fox Station
Response to Lessons Learned Report
USNRC Docket Nos. STN 50-556, 50-557

July 27, 1979
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Mr. Steven A. Varga, Assistant Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20002

Dear Mr. Varga:

The meeting between our respective organizations on July 19 in Bethesda was very productive. We especially appreciated your positive comments concerning our analysis of the lessons to be learned from TMI-2 as they apply to the Construction Permit application for the Black Fox Station. This analysis, which was sent to Mr. Harold R. Denton, Director, Nuclear Reaction Regulation, on June 15, 1979, represented the initial effort by Public Service Company of Oklahoma (PSO) to respond to the events at TMI and documented our long-term corporate commitment to fully analyze every facet of the TMI-2 accident and to incorporate the lessons learned into the design, construction, staffing, training and operation of the Black Fox Station.

With the issuance on July 19 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," we compared the 23 lessons learned in that report with the PSO analysis. Our understanding of NUREG-0578 and the comparison of the two documents were greatly facilitated by the helpful explanations and advice offered by you and your staff during our meeting on July 19. These discussions, along with information provided by Mr. Denton during his meeting with our President, Mr. R. O. Newman on July 20, enable us to respond promptly to your request for commitments to the requirements and recommendations of NUREG-0578.

Although our June 15 lessons learned analysis addressed most of the issues discussed in NUREG-0578, the organization of the material is different. Consequently, to facilitate your review, we are reiterating our commitments in a format consistent with the organization of NUREG-0578. In addition to specifically addressing every recommendation and requirement of NUREG-0578, this submittal also addresses matters applicable to Black Fox which were developed by the Bulletins and Orders Task Force, and the Emergency Preparedness group headed by Mr. Brian Grimes.

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We concur with the view presented during the meetings of July 19 and 20, that all of the commitments and actions required of us by the NRC Staff can be satisfied during the post-construction permit phase of the Black Fox design and construction effort, and that the documentation of these activities should be set forth in the Final Safety Analysis Report for the Black Fox Station. Our commitments reflect this understanding and philosophy.

The TMI-2 accident has stalled progress on the Black Fox application, and as you know, we are quite anxious to overcome this licensing delay. Consequently, we have responded directly and completely to all of the issues applicable to the Black Fox Station as presented by the two Task Forces and Mr. Grimes's group; this submittal should satisfy all of those concerns. In these circumstances, we do believe it reasonable to expect the NRC Staff to complete its report quickly and to respond to the Licensing Board Order of June 13, 1979 in the very near future.

Please call Mr. Vaughn Conrad, Manager, Licensing and Compliance at (918) 583-3611 if you have any questions regarding this submittal.

Sincerely yours,

T. N. Ewing

T. N. Ewing, Manager
Black Fox Station Nuclear Project

TNE:VLC:dm

Attachment

xc: (w/ attachment) BFS Service List

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BLACK FOX STATION SERVICE LIST

CERTIFICATE OF SERVICE

I hereby certify that a copy of the foregoing PSO Response to the TMI Event has been served on each of the following persons by deposit in the United States mail, first-class postage prepaid, this 27th day of July, 1979.

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RESPONSE OF

PUBLIC SERVICE COMPANY OF OKLAHOMA

BLACK FOX STATION, UNITS 1 & 2

USNRC DOCKET NOS. STN 50-556, 50-557

TO

NUREG-0578, Appendix A

TMI-2 Lessons Learned Task Force

Short-Term Recommendations

Inspection & Enforcement Bulletin 79-08

Selected Issues on Emergency Preparedness

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INTRODUCTION

AND DESCRIPTION OF METHODOLOGY

On June 15, 1979, Public Service Company of Oklahoma (PSO) submitted an analysis of the lessons to be learned from the events at Three Mile Island-Unit 2 as they apply to the construction permit application for the Black Fox Station (BFS). The submittal was documentation of the Company's long-term corporate commitment to incorporate those lessons into the design, staffing, training and operation of BFS. In addition, the document represented the initial effort by the PSO Technical Advisory Committee (TAC) constituted by the President and Chief Executive officer as an ongoing body expressly to study the events at TMI and to implement the lessons learned into our project.

With the issuance on July 19 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," the TAC compared the 23 lessons learned with our submittal. Although our June 15 analysis addressed most of the issues discussed in NUREG-0578, we found the organization of the material to differ in form. Hence, we chose to reiterate our commitments herein in accordance with the format of Appendix A to NUREG-0578.

Prior to development of this document, consultants to and members of the Technical Advisory Committee met on June 19 with appropriate members of the regulatory staff, including Mr. Varga, Mr. Thomas, Mr. Silver, Mr. Williams, to review the intent of the NUREG-0578 technical positions.

In study of the twenty-three issues, we found that three (2.1.1, 2.1.7a, 2.1.7b) did not apply to BFS because the issue was specific to pressurized water reactors. Three others (2.1.5 a, b, c) were not applicable because of the design features of the Black Fox Station which utilizes the BWR/6 Mark III System. Finally, one issue (2.2.3) did not apply since it is to be the subject of rulemaking.

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For the balance, the intent of each commitment by PSO is to meet the express position of the regulatory staff as stated in NUREG-0578, Appendix A.

During our meetings with the regulatory staff and the Director of Nuclear Reactor Regulation, Mr. Denton, on July 19 and 20, it became apparent that the BFS was expected to address itself to the activity of the Bulletins and Orders Task Force. In the meeting of June 20, Messrs. Novak and Kane of the B&O TF stated that the only issues that need to be addressed by the BFS were those contained in Inspection and Enforcement Bulletin (IEB) 79-08.

The June 15 submittal by PSO was intended to incorporate all of the requirements statedin IEB 79-08. In order to be completely responsive, each of the IEB 79-08 Tasks are repeated in this submittal followed by the appropriate PSO commitment for BFS.

The IEB 79-08 was specifically addressed to licensees with operating boiling water reactors and response was required very quickly. For projects such as BFS having yet to receive a full construction permit and where operation is projected well into the future, the requirements of IEB 79-08 were provided for information purposes. No written response was required, but actions will be completed prior to start of operation. The PSO commitments to action require completion of the efforts described during final design as detailed in the FSAR and in subsequently developed operating procedures.

PSO recognizes that the "Lessons Learned" requirements and the IEB 79-08 requirements represent separate activities within the regulatory staff. Thus, there exists some duplication of subject matter with the possibility of different interpretations of the PSO response between the two task forces. If such differences are identified, PSO commits to work with the NRC Staff to reconcile them.

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There are several issues related to the events at TMI which relate to radiological emergency planning. These are being evaluated by a NRC group headed by Mr. Brian Grimes who met with PSO on July 20, 1979. Mr. Grimes identified six matters which PSO should address in this submittal. Most were covered in our June 15 assessment.

Included in the emergency preparedness section is a letter from the Governor of the State of Oklahoma, George Nigh to Joseph Hendrie, Chairman USNRC. Therein, the status of the State Emergency Response Plan, PSO's role in development, and a commitment to have a NRC approved plan in effect well before BFS commercial operation is discussed.

PSO has also confirmed the feasibility of implementing a protective action plan over the area covered by a ten-mile radius from the BFS generation complex, a possible future licensing criteria mentioned by Mr. Grimes.

The PSO Technical Advisory Committee concurs with the view presented during the meetings of July 19 and 20, that all of the commitments and actions required by the NRC Staff can be satisfied during the post-construction permit phase of the Black Fox design and construction effort, and that th documentation of these activities should be set forth in the Final Safety Analysis Report and Station Operating Procedures for the Black Fox Station. Our commitments reflect this understanding and philosophy.

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RESPONSE TO

NUREG-0578, Appendix A

TMI-2 Lessons Learned Task Force

Short-Term Recommendations

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NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWR's (Section 2.1.1).

This issue is not applicable to the BWR/6 Nuclear Steam Supply System of the Black Fox Station, Units 1 and 2.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Performance Testing for BWR and PWR Relief and Safety Valves (Section 2.1.2).

NRC STAFF POSITION

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor cooling system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The signal failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analyses procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and support as well as the valves themselves.

PSO COMMITMENT

PSO believes that it is important to assure that the safety and relief valves installed in the BFS reactor coolant boundary will function as intended and maintain their integrity under expected operating conditions for design basis transients and accidents. Analysis of accidents and transients will be conducted during the final design stage to determine the most severe operating conditions and dynamic forces experienced by the safety and relief valves during the selected events. PSO, in cooperation with other applicants and licensees, will conduct necessary testing to qualify the reactor coolant system relief and safety valves for the most severe conditions identified.

Qualification of the associated control circuitry and piping and supports will be verified at the test conditions selected for the safety and relief valves. Documentation will be contained in the FSAR at the time of submittal in support of the operating license application.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's (Section 2.1.3.a).

NRC STAFF POSITION

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

PSO COMMITMENT

PSO will provide a reliable safety and relief valve position indication in the control room for the nineteen reactor main steam safety/relief valves in each nuclear steam supply system. Design detail will be provided in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Instrumentation for Detection of Inadequate Core Cooling in PWR's and BWR's (Section 2.1.3.b).

NRC STAFF POSITION

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instructions as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

PSO COMMITMENT

The ability of station operators to easily and unambiguously determine the status of core cooling and to provide adequate cooling is essential to the operation of the Black Fox Station. PSO will review the instrumentation presently provided within the BFS design to assure that adequate information is available for the clear definition of core cooling status. Should modifications or additional instrumentation be required to provide operators with clear, easily interpreted information, appropriate modifications or additions to instrumentation will be provided during final design. Operating procedures will be developed to guide the operator in recognizing inadequate core cooling, and operators will be throroughly trained in the procedure and utilization of instrumentation to assure correct interpretation of the core cooling status. A description of system functional requirements and of the instrumentation provided to enable operators to evaluate core cooling will be presented in the FSAR.

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NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Containment Isolation Provisions for PWR's and BWR's (Section 2.1.4).

NRC STAFF POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containmentisolation valves shall require deliberate operator action.

PSO COMMITMENT

PSO recognizes the importance for timely and effective isolation of the containment under accident conditions. PSO will review the design of BFS to assure that the final design provides for:

1. Diversity in the parameters sensed for the initiation of containment isolation, in accordance with SRP 6.2.4;
2. Automatic isolation of non-essential systems upon containment isolation signal;
3. Reopening of containment isolation valves only by deliberate operator action. The control system design will not cause the automatic reopening of containment isolation valves upon resettling of the isolation signal.

The definition of essential and non-essential systems will be re-evaluated to carefully identify essential systems and non-essential systems to assure that the bases for selection of essential systems are described, and that the containment isolation design is consistent with the definition. The results of the re-evaluation will be reflected in the final containment design as presented in the FSAR, including information on the definition of essential and non-essential systems.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems (Section 2.1.5.a).

NRC STAFF POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

Black Fox Station is designed for the installation of 100% redundant hydrogen recombiners within the containment of each unit. This position is therefore not applicable.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Inerting BWR Containments (Section 2.1.5.b).

NRC STAFF POSITION

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWR's.

Black Fox Station is designed with a Mark III Containment. This position is not applicable.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant (Section 2.1.5.c).

NRC STAFF POSITION (Minority View).

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

Black Fox Station is designed for the installation of 100% redundant hydrogen recombiners within the containment of each unit. This position is therefore not applicable to BFS.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWR's and BWR's (Section 2.1.6.a).

NRC STAFF POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as practical levels. This program shall include the following:

1. Immediate Leak Reduction.

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction.

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

PSO COMMITMENT

PSO will perform a review during the course of final design and make changes accordingly to provide a means of practical leak detection in systems outside containment which could be expected to have highly radioactive fluids as a result of a serious transient or accident. The review will also examine methods of leak repairs to achieve ALARA. Prior to initial operations, a preventive maintenance program shall be implemented to control the leakage, including periodic integrated leak rate tests, at a frequency not to exceed the refueling cycle interval. The FSAR will contain the results of the above design and operations review.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Design Review of Plant Shielding of Spaces for Post-Accident Operations (Section 2.1.6.b).

NRC STAFF POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

PSO COMMITMENT

PSO recognizes, as a result of the TMI-2 event, the need to assure necessary access to vital areas and protection of vital equipment under the impact of post-accident releases of radioactivity. PSO will identify vital areas and equipment, and based on the post-accident radioactivity releases described in Regulatory Guide 1.3, will evaluate the BFS design for unacceptable limitations on personnel access and occupancy or undue degradation of safety-related equipment during post-accident operations. The evaluation will consider alternatives, including layout changes, increased use of permanent shielding, temporary shielding, or procedural controls. The evaluation will determine changes needed throughout Black Fox Station. The results of the evaluation and a description of the changes will be reflected in the final design presented in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Automatic Initiation of the Auxiliary Feedwater System for PWR's (Section 2.1.7.a).

This issue is not applicable to the BWR/6 Nuclear Steam Supply System of the Black Fox Station, Units 1 and 2.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Auxiliary Feedwater Flow Indication to Steam Generators for PWR's (Section 2.1.7.b).

This issue is not applicable to the BWR/6 Nuclear Steam Supply System of the Black Fox Station, Units 1 and 2.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Improved Post-Accident Sampling Capability (Section 2.1.8.a).

NRC STAFF POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safety obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly (less than 2 hours) quantify certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel metling). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be perforemd in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e, the boron sample analysis within an hour and the chloride sample analysis within a shift.

PSO COMMITMENT

PSO will perform a design and operational review of the reactor coolant and containment atmospheric sampling system, the radioisotope analysis facilities, and chemical analyses to achieve prompt and safe sample acquisition and analysis in accordance with the position stated above. Results of these studies will be presented in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Increased Range of Radiation Monitors (Section 2.1.8.b).

NRC STAFF POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal oeprating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from a minimum of 10-7 uCi/cc (Xe-133) to a maximum of 10^5 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

PSO COMMITMENT

PSO shall provide the monitors as required in the staff position, and will document a description of the same in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Improved In-Plant Iodine Instrumentation (Section 2.1.8.c).

NRC STAFF POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

PSO COMMITMENT

PSO will provide instrumentation, training of personnel and the technical procedures for accurately determining airborne iodine concentration throughout the plant under accident conditions, with documentation to be provided in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Analysis of Design and Off-Normal Transients and Accidents (Section 2.1.9).

NRC STAFF POSITION

Analyses, procedures, and training addressing the following are required:

1. Samll break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests, (scheduled to start in September, 1979) shall be performed as a means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long-term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required: LOCA with forced flow; LOCA without forced flow);

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCA's, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

PSO COMMITMENT

As the penultimate paragraph of the above stated position of the NRC staff indicates, the requirement for additional transient and accident analyses is promoted by the need to develop more knowledge and information for reactor operations rather than a concern about the adequacy of reactor design. Information of this type is best developed on a generic basis, and as indicated below, such information will be available prior to the operation of the Black Fox Station.

PSO understands that analysis and emergency procedures or guidelines for:

1. Small break loss-of-coolant accidents;
2. Inadequate core cooling; and

3. Transients and accidents

are being generated by the operating Boiling Water Reactor Owners' Group in response to the Bulletins and Orders Task Force. These analyses are being generalized first to cover BWR/1-5 type power plants and will be extended by General Electric Company to cover the BWR/6 System generically. Each of the specific requirements stated in the above position have been identified by the Bulletins and Orders Task Force. As this assessment is completed for the operating power plants, the results will be reflected in the FSAR and factored into the Black Fox Station plant emergency procedures development and operator training. Analyses performed by General Electric will be put in the form of emergency procedures guidelines, and these guidelines will be implemented in the Black Fox Station procedures and training programs as appropriate.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Shift Supervisor's Responsibilities (Section 2.2.1.a).

NRC STAFF POSITION

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

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PSO COMMITMENT

PSO commits to comply with the staff position which provides methods to enhance plant safety and reliability. We recognize that the shift supervisor is the member of station management who ensures the safety and reliability of the plant on a daily basis. He will receive the full support of corporate management to enable him to perform his duties in a manner to provide the proper attention to safety and plant reliability.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Shift Technical Advisor (Section 2.2.1.b).

NRC STAFF POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

NRC STAFF COMMITMENTS

PSO will provide an on-shift technical advisor to the on-duty shift supervisor. The technical advisor shall have suitable experience, education and training as described in the staff position to prepare him for the duty of advising shift personnel on safe operations of the plant.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Shift and Relief Turnover Procedures (Section 2.2.1.c).

NRC STAFF POSITION

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist);

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklists); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

PSO COMMITMENT

PSO commits to compliance with the above position and concurs that it is a prudent management approach to plant operations.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Control Room Access (Section 2.2.2.a).

NRC STAFF POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access;

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

PSO COMMITMENT

PSO will comply fully with this position and recognizes the importance of access control to the control room.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Onsite Technical Support Center (Section 2.2.2.b).

NRC STAFF POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capabilty to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

A complete set of as-built drawings and other records, as described in ANSI N45.2.9-1974, shall be properly stored and filed at the site and accessible to the technical support center under emergency conditions. These documents shall include, but not be limited to, general arrangement drawings, P&ID's, piping system isometrics, electrical schematics, and photographs of components installed without layout specifications (e.g., field-run piping and instrument tubing).

PSO COMMITMENT

An onsite technical support center as described above will be [illegible] with the capability to display necessary plant status information for [illegible] who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. Various tools needed to support engineering and operational analyses shall be provided therein, such as communications and as-built drawings. The activation and use of this center shall be governed by the BFS Emergency Plan and the plant administrative procedures. A description of this center will be provided in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TITLE: Onsite Operational Support Center (Section 2.2.2.c).

NRC STAFF POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

PSO COMMITMENT

PSO will designate an area to serve as the operational support center as described in the above position. The support center will be physically separated from the control room, and appropriate communication facilities between the two will be provided. The BFS Emergency Plan and Station administrative procedurés will describe the activation and use of the Operational Support Center, as well as establish the methods and lines of communication and management control. The location of the Center will be provided in the FSAR.

NRR Lessons Learned Task Force

Short-Term Recommendations

TI1 ⁻: Revised Limiting Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability (Section 2.2.3).

NRC STAFF POSITION

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the ncessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4, and 5 of operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours;

2. Within 24 hours, bring the plant to cold shutdown;

3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established;

4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function;

5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee;

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of steps 3 and 4, above, along with a basis for allowing the plant to return to power operation. The senior corporate executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power.

PSO COMMITMENT

As indicated in the NUREG-0578 discussion preceding the position stated above, the Lessons Learned Task Force recognized that this position should be implemented through the rulemaking process provided for under the Administrative Procedures Act. This approach was emphasized in Dr. Mattson's letter of July 18, 1979 to Mr. Denton, atta... During the July 20 meeting with PSO, Mr. Denton stated that any commitment to the position must await the rulemaking process.

In view of the foregoing, no commitment to the above position is required of .SO at this time. PSO does agree to comply with any requirement ultimately determined by the rulemaking.

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

July 18, 1979

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Roger J. Mattson, Director
TMI-2 Lessons Learned Task Force

SUBJECT: TMI-2 LESSONS LEARNED TASK FORCE
REPORT (SHORT TERM) NUREG-0578

Enclosed is the first report of the TMI-2 Lessons Learned Task Force. It contains a set of short term recommendations to be implemented in two stages over the next 18 months on operating plants, plants under construction, and pending construction permit applications. There are 23 specific recommendations in 12 broad areas (nine in the area of design and analysis and three in the area of operations). The 23 recommendations would provide substantial, additional protection which is required for the public health and safety.

All but one of the 23 recommendations have a majority concurrence by the Task Force. The exception is the recommended requirement to provide capability to install an external recombiner at each reactor plant for post-accident hydrogen control, if necessary following an accident. The majority of the Task Force recommends that this matter deserves further evaluation in conjunction with other hydrogen generation and control questions being reviewed by the Task Force for its final report.

Three of the recommendations appear to require changes in existing regulations for which the Task Force recommends immediately effective rulemaking. They are: 1) inerting of MK I and MK II BWR containments that are not already inerted; 2) provision of the capability to install an external recombiner for plants that do not already have recombiners (minority view); and, 3) revised limiting conditions of operation in operating licenses for total loss of safety system availability through human or operational error. The Office of Standards Development has agreed to develop the required Commission papers and carry through with these rulemaking actions.

The 23 recommended actions were discussed with the Regulatory Requirements Review Committee (June 22, 1979), the Commission (June 25, 1979), the TMI-2 Subcommittee of the ACRS (July 11, 1979), and the ACRS (July 12, 1979). In addition, meetings were held with various groups in the Office of Nuclear Reactor Regulation in the course of the last few weeks to discuss technical aspects of specific portions of the recommended actions and the implementation alternatives.

The Task Force recommends that time not be taken to request and evaluate public comments on these short term requirements prior to their promulgation as licensing requirements or rules because they are safety significant matters that require prompt application to operating reactors and operating license applications in the late stages of review. Other TMI-2 accident review groups and the Lessons Learned Task Force are continuing to evaluate the longer term implications of the accident. Any public comments on the short term recommendations that are received after their issuance (just as in the case of the earlier IE Bulletins) can be factored into those continuing evaluations.

Having identified the 23 specific recommendations for short term action, the Lessons Learned Task Force will turn to the broader, more fundamental regulatory questions which should be addressed in the longer term (some of them likely to require evaluations that extend beyond the life span of the Task Force) before other regulatory actions are recommended. These longer term interests of the Task Force are described in Section Three of the report. The Task Force intends to develop its final recommendations and issue a final report in early September 1979. The topics to be addressed in the final report could affect the future structure and content of the licensing process to correct deficiencies identified by the TMI-2 accident and to further upgrade the level of safety in operating plants and plants under construction. The Task Force does not believe that allowing new plants to begin operation in the next few months will foreclose further design changes that may be shown to be desirable by its continuing review of the accident.

On July 11, I solicited the comments of the principal NRR line organizations on the final draft of the report and its central conclusion regarding the necessity and sufficiency of the short term recommendations for continued operations and licensing. General support for the conclusions of the Task Force report was expressed by all of the principal NRR line managers. We have reviewed and considered the detailed comments supplied by the various NRR organizations in the course of their review. Where appropriate, we made clarifying changes in the language of the report. The principal substantive change occurred in the form and schedules of the implementation section (Appendix B). Some of the comments addressed matters that the Task Force has deferred for consideration in its final report. There are significant differences of opinion within the staff on two of the Task Force recommendations, as follows: a) the need for recommendation 2.2.3 concerning rulemaking for revised limiting conditions for operation (some agree with the recommendation and others prefer more stringent enforcement actions using existing regulatory machinery) and b) the need for the minority Task Force recommendation 2.1.5.c concerning rulemaking for backfit of recombiner capability (some support the minority recommendation, others do not). Having considered these comments and made changes to the report where appropriate to reconcile them with the intent of the Task Force, I recommend that you:

a. direct the immediate implementation by DPM, DOR or B&OTF, as appropriate, of all the short term recommendations, except the three rulemaking matters, through the issuance of licensing positions to operating plant licensees, plants under construction, and construction permit applicants.

Harold R. Denton

b. request the formulation of immediately effective rules by the Office of Standards Development for action by the Commission on the three rulemaking matters.

Another matter that needs to be considered by you in deciding upon the additional requirements for near term CP and OL decisions and for operating reactors is improvements in licensee emergency preparedness.

Roger J. Mattson

Roger J. Mattson, Director
TMI-2 Lessons Learned Task Force

Enclosure: as stated

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne
ACRS (20)
Policy Evaluation
SECY
L. V. Gossick, EDO
S. Levine, RES
R. Minogue, SD
V. Stello, IE
M. Rogovin, Special Inquiry
J. Fouchard, PA (20)
C. Kammerer, CA (20)
NRC PDR

RESPONSE TO

INSPECTION & ENFORCEMENT BULLETIN 79-08.

IEB 79-08 Task 1

Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 03/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action;

b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 5a of this bulletin); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available;

c. All licensed operators and Plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documentedin plant records.

PSO COMMITMENT

Public Service Company of Oklahoma has established a Technical Advisory Committee (TAC) to assess the events at Three Mile Island, Unit 2, and to apply the lessons learned to its Black Fox Station Project. This committee was established at the direction of the President and Chief Executive Officer of the Company and reports its findings and recommendations directly to the Review and Audit Committee. These findings and recommendations will then be implemented by the Review and Audit Committee.

The TAC has been directed to utilize PSO and consultant resources to fully review the interim and final results of the various investigations. These presently include:

- USNRC's "Lessons Learned Task Force"--NUREG-0578
- The President's Commission on Three Mile Island
- EPRI--Nuclear Safety Analysis Center

IEB 79-08 Task 1--Continued.

- Generic vendor programs
- Atomic Industrial Forum TMI Policy Committee
- NRC Special Investigation (Rogovin)

The TAC and its consultants have already assessed issuances of the ACRS and regulatory staff and presented a preliminary assessment to the NRC Staff in our June 15 submittal. It is aware of the activities of various other legislative and regulatory investigations and will assess future recommendations from them.

The assessment and resulting program was predicated on the advice, and guidance set forth in the various letters, from the ACRS (particularly their letters of April 7 and May 16, 1979), and IE Bulletin No. 79-08, dated April 14, 1979. In addition, S. Levy, Inc., a participant in both the post-event safe shutdown activities of TMI and the EPRI investigation, has been retained to keep PSO continously informed of any new developments arising from the ongoing investigations by EPRI and other organizations.

The objective of the TAC and its consultants is to ensure that the Black Fox Sation design, construction, operating procedures, staffing and training program, and emergency response plan incorporates the benefits of the TMI investigation to the fullest extent practicable.

The effort is directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three-Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

Prior to completion of operating procedures and training instructions for operation of the Black Fox Station, these procedures and instructions will be reviewed to assure that operational personnel are instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions, and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available. (See also commitments made under IEB 79-08 Task 5).

The Manager, Black Fox Station and the Manager, Nuclear Training are assigned to the TAC to ensure that operational experience is considered in the TAC reviews and to provide continuity for implementation of TAC findings into operator license and station supervisor/management training. A key objective of the TAC is to review administrative mechanisms to ensure that lessons learned are incorporated into the station training programs.

Findings and recommendations from the TAC will be documented in the Project files and conformance with each specified commitment will be incorporated into this documentation system.

IEB 79-08 Task 2

Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate containment isolation, whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

PSO COMMITMENT

At the time of final design, i.e., FSAR submittal, and prior to completion of operating procedures, containment isolation initiation will be reviewed to assure containment isolation of all lines whose isolation does not degrade needed safety features or cooling capability upon automatic initiation of safety injection. This isolation may be automatic or manual, and any necessary manual actions will be covered by appropriate procedures.

IEB 79-08 Task 3

Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure by which this action is taken in a timely sense.

PSO COMMITMENT

At the time of final design, i.e, FSAR submittal, and prior to completion of operating procedures, the functioning of the auxiliary heat removal systems that are used when the main feedwater system is not operable will be reviewed. Both automatic and manual actions will be assessed for adequacy, and any necessary manual actions will be addressed by procedures to assure timely actuations.

IEB 79-08 Task 4

Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

PSO COMMITMENT

At the time of final design, i.e, FSAR submittal, and prior to completion of operating procedures, all uses and types of vessel level indication for both automatic and manual initiation of safety systems will be reviewed. Redundant instrumentation which the operator will have to give the same vessel level indications will be identified and factored into operator training, instruction, and procedures.

IEB 79-08 Task 5

Review the action directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g., vessel integrity);

b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evaluating plant conditions.

PSO COMMITMENT

Prior to completion of operating procedures and training instructions, actions directed by these instructions will be reviewed to ensure that:

a. Operators are directed not to override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions;

b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evaluating plant conditions.

IEB 79-08 Task 6

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also, review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

PSO COMMITMENT

At the time of final design, i.e., FSAR submittal, PSO will review all safety-related valve positioning requirements and positive controls to assure that valves remain positioned in a manner to ensure the proper operation of engineered safety features. In addition, prior to completion of related procedures, the procedures for maintenance, testing, plant and systems startup, and supervisory periodic surveillance will be reviewed to ensure that safety-related valves are returned to the correct position following necessary manipulations and are maintained in the proper position during all operational modes.

IEB 79-08 Task 7

Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting, or other relase of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and;

b. Whether such systems are isolated by the containment isolation signal;

c. The basis on which continued operability of the above features is assured.

PSO COMMITMENT

At the time of final design, i.e., FSAR submittal, and prior to completion of operating procedures, the operating modes of all systems designed to transfer potnetially radioactive gases and liquids out of the primary containment will be reviewed to assure that undesired pumping, venting, or other release of radioactive gases and liquids will not occur inadvertently.

In particular, the impact of resetting of engineered safety features instrumentation will be examined to ensure that such an inadvertent radioactive liquid or gas release will not result from this resetting.

Each of the above systems will be reviewed to assure that:

a. Interlocks exist to prevent transfer when high radiation indication exists, and;

b. Such systems are isolated by the containment isolation signal.

IEB 79-08 Task 8

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service;

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing:

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

PSO COMMITMENT

Prior to their completion, maintenance and test procedures for safety-related systems will be reviewed to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service;

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing;

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from or returned to service.

IEB 79-08 Task 9

Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time, an open continuous communication channel shall be established and maintained with NRC.

PSO COMMITMENT

Prior to completion of the emergency plan and implementing procedures, NRC notification shall be incorporated to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at the time of NRC notification, an open continuous communication channel will be established and maintained with NRC.

IEB 79-08 Task 10

Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

PSO COMMITMENT

At the time of final design, i.e, FSAR submittal, and prior to completion of operating procedures, operating modes and procedures will be reviewed to assure that they are adequate to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

IEB 79-08 Task 11

Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items above.

PSO COMMITMENT

Those issues that need to be addressed by technical specifications as a result of implementing IEB 79-08 task items 1 through 10 shall be incorporated prior to completion of the technical specifications which will be submitted with the FSAR.

RESPONSE TO

SELECTED ISSUES ON EMERGENCY PREPAREDNESS

Emergency Preparedness

i. Regulatory Guide 1.101 Emergency Planning For Nuclear Power Plants.

The BFS PSAR, Section 1.9 reflects a commitment to revision 0 of this regulatory guide. For the purposes of design and development of operating procedures, PSO will use Revision 1 dated March, 1977. Full implementation will be demonstrated at the time of FSAR submittal.

Discussions with the regulatory staff have indicated that revisions to the uniform action level criteria will be forthcoming as a result of the experiences at TMI. PSO will utilize these criteria in development of the BFS Emergency Plan.

ii. Improved Sampling and Instrumentation Capability.

These issues are covered in NUREG-0578 TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations as issue 2.1.8. PSO has addressed these requirements in our response to that section.

iii. Emergency Operating Center.

The BFS PSAR § 13.3.3 identifies a secondary Emergency Control Center located away from the generation complex, but within the site boundary. This center will serve as the focal point for radiological emergency response, i.e., an emergency operating center, by being the coordination point for local, state, and federal authorities involved. Appropriate plant status and meteorological data will be read directly from instrumentation placed in the center.

iv. Improved Offsite Monitoring Capability.

As a part of its evaluation of the events at TMI, PSO commits to reevaluate the necessary capabilities of offsite radiation monitors. The number and location of thermoluminescent dosimeters (TLD's) will be studied, as well as possible use of continuous radiation monitors with remote readout. PSO also commits to closely monitor forthcoming regulatory guidance in this area to assure that appropriate capabilities are promptly factored into the BFS design and operation plan.

v. Adequacy of Protective Action Planning.

PSO is evaluating the current regualtory requirements for emergency planning in light of the events at TMI. Since April 1, 1979, our techincal staff has had several meetings with Oklahoma State Department of Health, Division of Occupational and Radiological Safety personnel who have been designated by the Governor, State of Oklahoma, as the prime state agency respondent.

The State of Oklahoma does not presently have in effect an emergency response plan. The attached letter dated June 20, 1979 from George Nigh, Governor, State of Oklahoma, to Joseph Hendrie, Chairman, U. S. Nuclear Regulatory Commission, explains the State's status in preparing such a plan, and receiving NRC approval. As stated therein, PSO personnel are working closely with the State in review of the draft. We are fully prepared to assist the State in timely final development and submittal to NRC approval.

Concurrently, PSO is establishing target tasks for the BFS Emergency Response Plan development. The plan will be submitted with the FSAR in support of the application for operating licenses.

Our understanding from recent discussions with the Staff is that protective actions in the future may be planned out to a radius of 10 miles rather than out to the radius of the Low Population Zone (LPZ) of 4,000 meters as reflected in the BFS Preliminary Safety Analysis Report and Environmental Report.

Accordingly, we have reviewed the applicable discussion from the ER (§ 2.1.3.1) on the popluation projections within a ten-mile radius of the site. Also studied were PSAR tabluations of regional incorporated community statistics and population projections of the two communities within the area. Finally, we examined the PSAR figure relating to emergency evacuation routes for the ten-mile area.

The only significant population concentration within the ten-mile radius area is the town of Inola. The area is primarily rural and is expected to remain so during the lifetime of Black Fox Station. The 1980 estimated population of Inola is 2900 with projections increasing to 4600 by the year 2020.

There are three other small communities within ten miles of Black Fox Station, in addition to Inola as shown in ER figure 2-1-6. They are New Tulsa (eight miles WSW), Fair Oaks (nine miles WNW), and Tiawah (ten miles N). New Tulsa and Fair Oaks populations are expected to increase only marginally. Much of the Tiawah 1980 estimated population of 125 is located beyond the ten-mile radius while the 2020 population is expected to be only 321.

The accompanying ER Table 2-1-1 shows that the overall population density within the ten-mile radius of the Black Fox Station is small--less than 15,000 in 1980 and less than 24,000 in 2020.

PSAR Figure 13.3-3 shows the potential emergency evacuation routes. Major routes such as state highways 18 and 33 and U. S. Highway 69 are identified. In addition, since Oklahoma is uniformly divided into square mile sections, each of the perpendicular lines forming uniform squares on the figure represents a transportation route.

As a result of our review, we have concluded that implementation of protective measures such as evacuation is feasible over the lifetime of the station based on population estimates and evacuation routes.

vi. <u>Periodic Testing</u>.

PSO comments to periodically conduct local emergency plan testing to assure that the plan is fully functional and kept up-to-date with regard to local population location and transportation routes. In addition, we recognize the benefits of an integrated PSO/State/NRC test to fully check communications and to insure correct agency interaction. We will support the practice of integrated testing.

STATE OF OKLAHOMA
OFFICE OF THE GOVERNOR
212 STATE CAPITOL BUILDING
OKLAHOMA CITY, OKLAHOMA 73105

GEORGE NIGH
Governor
405 / 521-2345

June 20, 1979

Mr. Joseph M. Hendrie, Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Hendrie:

I share your concern with regard to states having adequate radiological emergency response plans in operation which support fixed nuclear facilities. I appreciate your kind offer to assist in preparing such a plan through the mechanism of the Federal Interagency Regional Advisory Committee and your agency.

The Occupational and Radiological Health Service of the Oklahoma Department of Health, in cooperation with the Oklahoma Office of Civil Defense, has recently completed a preliminary draft of Oklahoma's radiological plan. Copies of this draft have been circulated to my office, several State executive agencies, the NRC Office of State Programs, and Public Service Company of Oklahoma for comments. Following revision in accord with these comments, the plan will be circulated for comment to these State agencies, local officials, the public, and the NRC. Our current schedule calls for a final version of the plan to be ready by early 1980. We fully intend and expect to receive NRC concurrence to the final plan several years prior to the now anticipated operational status of the Black Fox Station in 1985.

Sincerely yours,

George Nigh

George Nigh

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2.1.3.1 <u>Population Within 10 Miles</u>. A map of the 10-mile area of the BFS Site is presented on Figure 2.1-6. The map is overlayed with concentric circles, centered on the central plant complex with radii of 1, 2, 3, 4, 5, and 10 miles, and with radial lines forming 22-1/2 degree sectors centered on the 16 cardinal compass points. Table 2.1-1 presents the corresponding projected residential population within each annular and radial sector segments for the expected first year of plant operation (1983) and by census decade beginning with 1990 through the end of the anticipated plant life (2020). The largest cumulative population density for this area through the year 2020, occurs within the 4-mile radius area, in which the town of Inola is located. The 10-mile radius area is primarily rural and is expected to remain as such during the period of plant operation. Base data and methodology of population projections are presented in Subsection 6.1.4.2.

The town of Inola is the only significant population concentration within the 10-mile area. The 1974 estimated population of Inola is 1176 with projections presented in the Community Development Plan, Inola Oklahoma, increasing to 4200 by the year 2000 (6). There are three other small communities within 10 miles of BFS in addition to the town of Inola. The other communities are New Tulsa (8 miles WSW), Fair Oaks (9 miles WNW), and Tiawah (10 miles N). New Tulsa and Fair Oaks are incorporated entities in Wagoner County while Tiawah is unincorporated and located in Rogers County. New Tulsa and Fair Oaks populations are not expected to increase significantly according to projections by the Oklahoma Employment Security Commission (7). Much of the Tiawah current, estimated population of 95, is located beyond the 10-mile radius (8).

2.1.3.2 Population Between 10 and 50 Miles. Figure 2.1-7 shows the region within 50 miles of the reactor locations in northeast Oklahoma with concentric circles drawn at 10-mile radius intervals and with radial lines defining sectors centered on the 16 cardinal compass directions. The projected populations for 1983, 1990, 2000, 2010, and 2020 for each annular and radial sector segments are presented in Table 2.1-2. The methods for estimating population distribution are described in subsection 6.1.4.2. The nearest population center (as defined in 10 CFR 100) at the time of startup of Unit 1 is Tulsa, Oklahoma with a 1970 census population of 330,350 (9). The nearest boundary of the densely populated area of Tulsa determined by interpretation of July 1974 aerial photographs is located 13 miles west of the Site. This distance is 5.2 times the low population zone radius of 2.5 miles.

The segment within 50 miles of BFS with the largest projected population is the segment containing Tulsa, Oklahoma, which is the west sector, between 20 and 30-mile radii. The largest projected cumulative population density area is within 30 miles of BFS, in which the city of Tulsa is located.

Regional incorporated community statistics are presented in Table 2.1-3. Data presented are the name of the community, county in which the community is located, distance and direction from the Site, and the 1970 census population. Location of the above communities in relation to the Site are shown on Figure 2.1-8.

2.1.3.3 Transient Population. The transient population within a 5-mile radius of BFS central complex include school and church attendees, commercial and industrial employees, recreational facility employees and users, and public

TABLE 2.1-1

AREA RESIDENT POPULATION AND PROJECTIONS
(Ref. Figure 2.1-6)

Sector	Year	Radial Distance from Reactor (mile) 0-1	1-2	2-3	3-4	4-5	5-10	10-Mile Total
N	1970	3	3	25	56	22	254	363
	1983	0	4	36	80	31	362	513
	1990	0	5	42	94	37	425	603
	2000	0	7	56	126	49	570	808
	2010	0	8	70	156	61	707	1002
	2020	0	10	84	189	74	857	1214
NNE	1970	3	0	51	8	42	310	414
	1983	0	0	297	417	60	443	1217
	1990	0	0	305	501	70	519	1395
	2000	0	0	401	572	94	696	1763
	2010	0	0	426	609	117	863	2015
	2020	0	0	440	633	142	1046	2261
NE	1970	0	8	243	674	44	222	1191
	1983	0	135	908	974	160	310	2487
	1990	0	140	932	999	191	352	2614
	2000	0	184	1223	1311	232	427	3377
	2010	0	197	1296	1389	263	496	3641
	2020	0	208	1339	1435	293	569	3844
ENE	1970	0	5	8	33	47	210	303
	1983	0	7	287	76	67	289	726
	1990	0	8	298	90	79	322	797
	2000	0	11	390	114	105	363	983
	2010	0	14	416	134	131	402	1097
	2020	0	17	434	159	159	438	1207
E	1970	0	8	11	8	14	194	235
	1983	0	11	16	11	20	266	324
	1990	0	13	18	13	23	295	362
	2000	0	18	25	18	31	327	419
	2010	0	22	31	22	39	357	471
	2020	0	27	37	27	47	384	522
ESE	1970	0	8	11	14	0	227	260
	1983	0	11	16	20	0	334	381
	1990	0	13	18	23	0	381	435
	2000	0	18	25	31	0	440	514
	2010	0	22	31	39	0	499	591
	2020	0	27	37	47	0	551	662

TABLE 2.1-1 (Continued)

Sector	Year	0-1	1-2	2-3	3-4	4-5	5-10	10-Mile Total
SE	1970	3	3	5	5	18	194	228
	1983	0	4	7	7	29	312	359
	1990	0	5	8	8	34	369	424
	2000	0	7	11	11	41	442	512
	2010	0	8	14	14	48	514	598
	2020	0	10	17	17	55	588	687
SSE	1970	3	8	17	18	0	335	381
	1983	0	11	24	29	0	540	604
	1990	0	13	28	34	0	794	869
	2000	0	18	38	41	0	952	1049
	2010	0	22	47	48	0	1107	1224
	2020	0	27	57	55	0	1268	1407
S	1970	3	14	3	3	0	442	465
	1983	0	20	4	5	0	712	741
	1990	0	23	5	6	0	840	874
	2000	0	31	7	7	0	1007	1052
	2010	0	39	8	8	0	1171	1226
	2020	0	47	10	9	0	1340	1407
SSW	1970	0	0	0	10	26	285	321
	1983	0	0	0	16	42	459	517
	1990	0	0	0	19	49	541	609
	2000	0	0	0	23	59	649	731
	2010	0	0	0	26	69	755	850
	2020	0	0	0	30	79	864	973
SW	1970	0	0	0	3	10	495	508
	1983	0	0	0	5	16	797	818
	1990	0	0	0	6	19	940	965
	2000	0	0	0	7	23	1128	1158
	2010	0	0	0	8	26	1311	1345
	2020	0	0	0	9	30	1501	1540
WSW	1970	8	0	0	5	8	596	617
	1983	0	0	0	8	13	960	981
	1990	0	0	0	9	15	1124	1148
	2000	0	0	0	11	18	1349	1378
	2010	0	0	0	13	21	1568	1602
	2020	0	0	0	15	24	1795	1834

TABLE 2.1-1 (Continued)

Sector	Year	0-1	1-2	2-3	3-4	4-5	5-10	10-Mile Total
W	1970	0	3	8	0	0	810	821
	1983	0	5	13	0	0	1,305	1,323
	1990	0	6	15	0	0	1,539	1,560
	2000	0	7	18	0	0	1,846	1,871
	2010	0	8	22	0	0	2,145	2,175
	2020	0	9	24	0	0	2,456	2,489
WNW	1970	3	3	3	0	3	23	35
	1983	0	5	5	0	5	37	52
	1990	0	6	6	0	6	44	62
	2000	0	7	7	0	7	52	73
	2010	0	8	8	0	8	61	85
	2020	0	9	9	0	9	70	97
NW	1970	3	0	17	25	19	612	676
	1983	0	0	24	36	27	874	961
	1990	0	0	28	42	32	1,025	1,127
	2000	0	0	38	56	43	1,374	1,511
	2010	0	0	47	70	53	1,703	1,873
	2020	0	0	57	84	64	2,064	2,269
NNW	1970	0	0	39	61	44	291	435
	1983	0	0	56	87	63	416	622
	1990	0	0	65	102	74	487	728
	2000	0	0	88	137	99	653	977
	2010	0	0	109	170	122	810	1,211
	2020	0	0	132	206	148	982	1,468
GRAND TOTALS	1970	29	63	441	923	297	5,500	7,253
	1983	0	213	1,693	1,771	533	8,416	12,626
	1990	0	232	1,768	1,946	629	9,997	14,572
	2000	0	308	2,327	2,465	801	12,275	18,176
	20[illegible]	0	348	2,525	2,706	958	14,469	21,006
	2020	0	391	2,677	2,915	1,124	16,774	23,881

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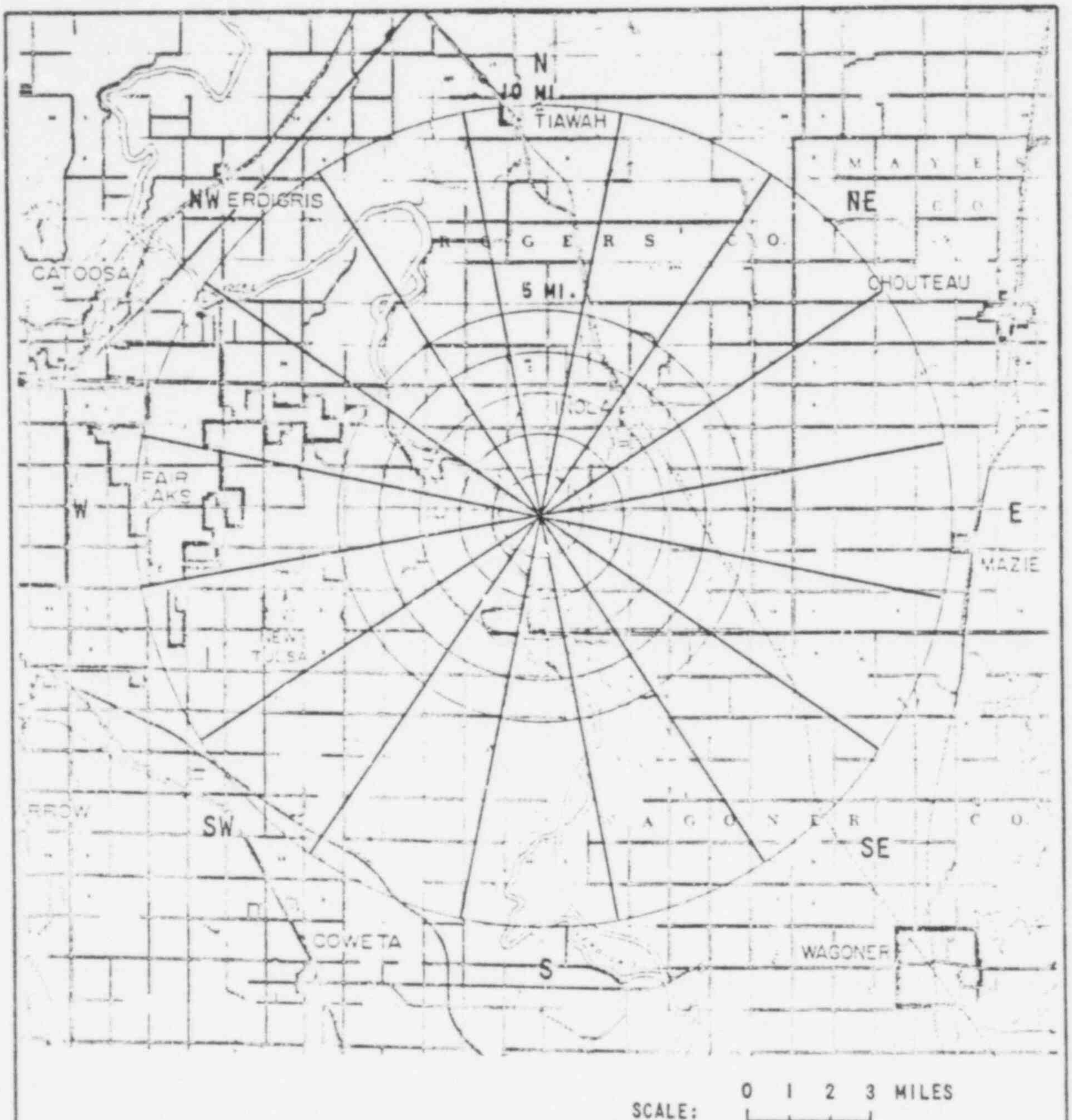

BFS 10-MILE RADIUS AREA MAP
(REFER TO TABLE 2.1-1)

FIGURE 2.1-6

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TABLE 2.1-1

REGIONAL INCORPORATED COMMUNITY STATISTICS

City	County	Distance and Direction	1970 Population
Inola	Rogers	3 miles NE	948
New Tulsa	Wagoner	8 miles WSW	17
Fair Oaks	Wagoner	9 miles WNW	23
Tiawah*	Rogers	10 miles N	95**
Catoosa	Rogers	12 miles WNW	970
Chouteau	Mayes	13 miles ENE	1,046
Coweta	Wagoner	13 miles SSW	2,457
Broken Arrow	Tulsa	14 miles WSW	11,787
Claremore	Rogers	14 miles NNW	9,084
Wagoner	Wagoner	15 miles SE	4,959
Red Bird	Wagoner	16 miles S	230
Porter	Wagoner	18 miles S	624
Pryor	Mayes	19 miles NE	7,057
Owasso	Tulsa	20 miles WNW	3,491
Tullahassee	Wagoner	21 miles SSE	183
Haskell	Muskogee	22 miles SSW	2,063
Foyil	Rogers	22 miles N	164
Bixby	Tulsa	23 miles SW	3,973
Locust Grove	Mayes	23 miles ENE	1,090
Okay	Wagoner	23 miles SE	419
Tulsa	Tulsa	23 miles W	330,350
Collinsville	Tulsa	24 miles NW	3,009
Jenks	Tulsa	25 miles WSW	1,997
Taft	Muskogee	25 mils S	525
Oologah	Rogers	25 miles NNW	458
Peggs	Cherokee	26 miles E	82
Salina	Mayes	26 miles ENE	1,024
Hulbert	Cherokee	26 miles ESE	505
Adair	Mayes	28 miles NE	459

*Tiawah is an unincorporated area within 10 miles of the plant site. It has been included in this listing because of its proximity to the plant site.

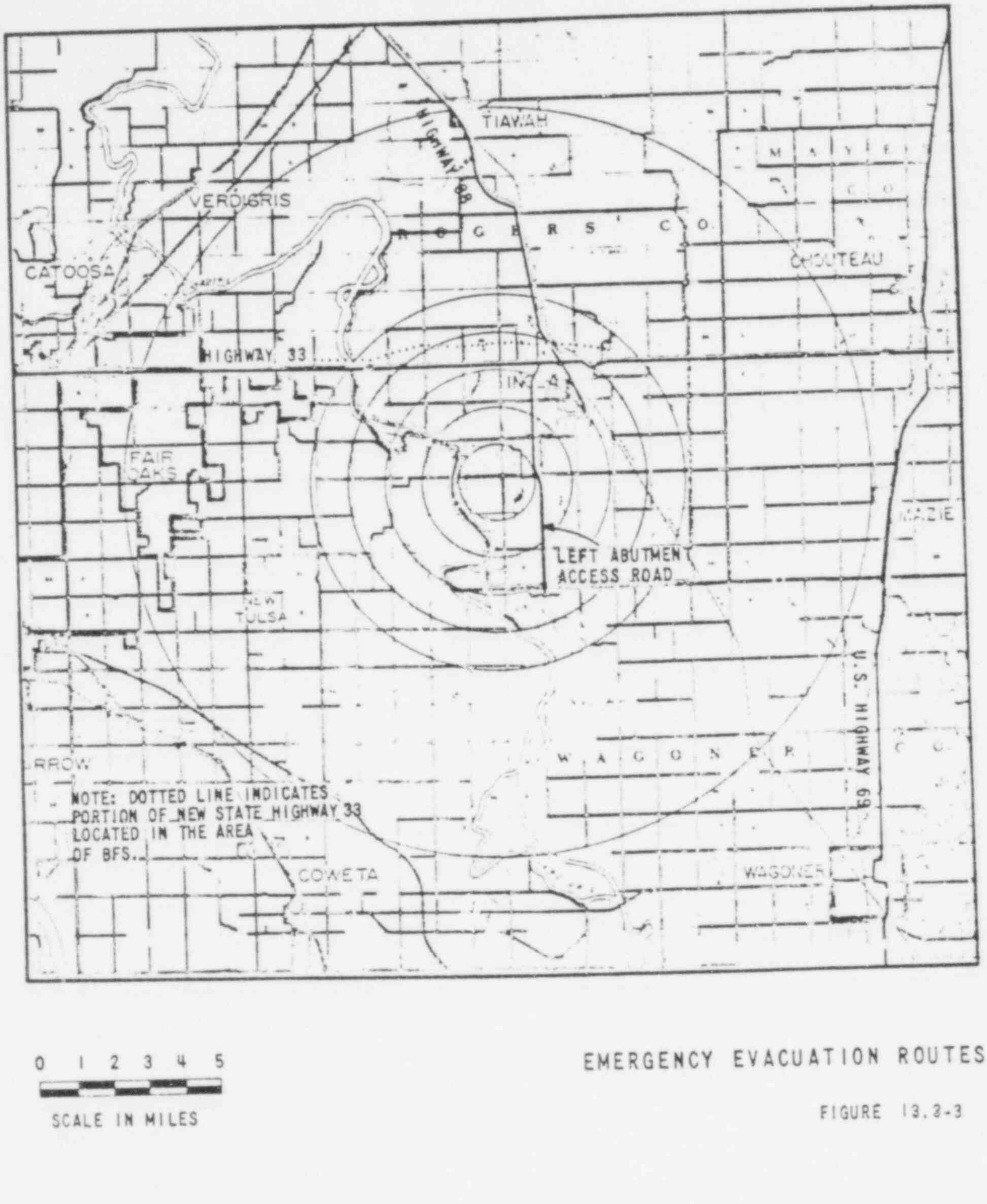
**Tiawah population is estimated from dwelling counts on the County Highway Map insert.

TABLE 2.1-3

LOCAL COMMUNITY POPULATION PROJECTION & DENSITY

	YEAR	ESTIMATED POPULATION	DENSITY*
Inola (3 mi. NE)	1970	948	237
	1974	1,176	345
	1977	2,050	512
	1980	2,900	725
	1983	3,080	770
	1990	3,700	925
	2000	4,200	1,050
	2010	4,450	1,112
	2020	4,600	1,150
Tiawah (10 mi. N)	1970	95	127
	1974	106	141
	1977	116	155
	1980	125	167
	1983	135	180
	1990	159	212
	2000	213	284
	2010	264	352
	2020	321	428

*Residents per square mile.



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0 1 2 3 4 5

SCALE IN MILES

EMERGENCY EVACUATION ROUTES

FIGURE 13.3-3

N