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In the Matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON FERMI-2 OPERATING LICENSE REVIEW

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON FERMI-2 OPERATING LICENSE REVIEW

Nuclear Regulatory Commission
Room 1046
1717 H Street, N. W.
Washington, D. C.

Friday, July 24, 1981

The meeting convened, pursuant to notice, at 8:30 a.m.

ACRS MEMBERS PRESENT:

W. KERR, Chairman
M. CARBON
D. MOELLER
J. RAY

ACRS CONSULTANTS PRESENT:

Z. ZUDANS
I. CATTON

DESIGNATED FEDERAL EMPLOYEE:

P. BOEHNERT

ALSO PRESENT:

L. KINTER
P. BYRON
B. YOUNGBLOOD
L. SCHUERMAN
H. TAUBER
W. JENS

E. COLBERT
E. PEDERSON
L. KANOUS
E. LUSIS
L. WOODEN
T. HUANG
F. GREGOR
W. STREET
J. KIMBALL
J. KNIGHT
M. DEORA
D. LENERT
C. TAN
E. PAGE
J. GREEN
J. LANE
E. GRIFFING
W. HODGES
Q. DUONG

P R O C E E D I N G S

MR. KERR: The meeting will come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, the Subcommittee on Fermi-2 which is applying for an operating license.

My name is William Kerr. Other ACRS members are Mr. Carbon, Mr. Moeller and Mr. Ray. Consultants present are Mr. Zudans and Mr. Catton.

The purpose of the meeting is to review Detroit Edison's application for a license to operate the Enrico Fermi Unit 2 plant. The meeting is being conducted on accordance with the provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act.

Mr. Paul Boehnert is the designated federal employee for the meeting.

The rules for participation in today's meeting have been announced as part of the notice of the meeting published in the Federal Register of July 14th, 1981.

A transcript of the meeting is being kept and will be available as stated in the Federal Register notice. I request that each speaker identify himself and use a microphone so that we can record his remarks.

We have received no written statements from members of the public nor have we had requests for time to make statements from members of the public.

We will proceed with the meeting.

I call on Mr. Les Kinter of the Nuclear Regulatory Commission for introductory comments.

Mr. Kinter.

MR. KINTER: I am Les Kinter, the licensing project manager for Fermi-2 for the Nuclear Regultory Commission. To my right is Paul Byron. He is one of the resident inspectors at Fermi-2, and my branch chief Joe Youngblood to his right.

I would like to use the projector. I have a presentation to make.

MR. KERR: Surely.

(Slide presentation.)

MR. KINTER: Fermi-2 is a BWR 4 boiling water reactor being built on Lake Erie in Monroe County, Michigan. I will give you an overview of the NRC review of Detroit Edison's application to operate Fermi-2, including open items remaining after the issuance of the safety evaluation report and plans for completing our review of these open items.

Our review of the operating license application has been substantially completed. A summary of our review is reported in the SER, the safety evaluation report issued in July of this year, this month.

Based on its review, the NRC staff has concluded that Fermi-2 can be operated without endangering the health and safety of the public subject to the completion of our review of the the open items identified in the SER and the resolution of any issues raised by the ACRS in its review. The prospective decision date for issuing an operating license is November 1982.

MR. MOELLER: Excuse me. The date for the OL is tentatively scheduled as of November of '82. When would they start operating?

MR. KINTER: This is the scheduled fuel loading date.

MR. MOELLER: Oh, fuel loading date.

MR. KINTER: Yes, the fuel loading date coincides with the OL. I use them synonymously.

MR. MOELLER: I misunderstood. I thought that was the OL, the issuance of the operating license.

MR. KINTER: We try to issue the operating license slightly before it is needed for the fuel loading, but with this one we tried to issue them at about the same time. So that all the construction is complete and we can review the completed plan.

The operating license review was made in three periods. When the final safety analysis report was docketed the site characteristics and design criteria for structure systems and components were reviewed.

The review was terminated when construction completion was delayed and an interim safety evaluation report was prepared to give the status of the review.

Some review areas were completed such as meteorology, probable maximum flood, geology, seismic and quality group classification of structures, components, missile protection, design basis for category one structures and the ultimate heat sink. These are the major structures and site characteristics which do not change and have not changed since of course the plant was designed and construction started.

The open items were identified in review areas that were not completed.

After the final safety analysis report was updated in June of 1978, the review was restarted. The Three Mile Island 2 accident occurred prior to the completion of our safety evaluation report. However, a safety evaluation input had been completed for some areas as indicated in the slide.

The review was again suspended while the Three Mile Island lessons were applied from 2.

The third review period began when Detroit Edison filed its response to Three Mile Island requirements of NUREG 0737. The review of non-TMI areas was completed, including new review areas that had risen since the suspension of the review in 1979. For example, masonry walls, fracture toughness of containment pressure boundary, emergency diesel engine lubrication system and the "Q" list.

A major review area in this period was Detroit Edison's response to Three Mile Island requirements.

I would now like to address the open items that remain and are listed in our safety evaluation report.

MR. MOELLER: During which of the three review periods was emergency planning covered?

MR. KINTER: Emergency planning is being covered now under the Three Mile Island requirements.

MR. MOELLER: Is the review of emergency planning, does it include consultation with officials in Canada?

MR. KINTER: Not to date.

MR. MOELLER: Will it before it is completed?

MR. KINTER: I don't know the extent of our interaction with Canada. I know we have not interacted with them to date. I will explore that. I just don't know at the time just how far we will go.

MR. MOELLER: Is this plant closer to a foreign country than any other commercial plant in the U. S.?

MR. KINTER: I don't know the location of the other commercial plants. I was thinking about that last night, what other plants are near the border. I can't think of any that have another country's territory within the

50-mile radius.

MR. MOELLER: In fact, this has some territory within the 10-mile radius if the map is correct.

MR. KINTER: It has a slight sliver of beach on the Canadian side.

MR. KERR: I thought when you referred to the foreign country you might have been thinking of Ohio and Michigan.

(Laughter.)

MR. KINTER: The review areas that are still open are listed on the next three slides.

This slide identifies the areas that are expected to be closed in a supplement to the safety evaluation report.

The first one, Detroit Edison's statement on conformance to 10 CFR Part 20, 10 CFR Part 50 and 10 CFR Part 100, is expected to be filed by the end of the month, this month. This statement will summarize how these rules are met and reference applicable parts of the application for details.

MR. KERR: Mr. Kinter, help me. Is this part of the TMI action plan or some rule? I gather that the licensee has to go through and say we conform to 10 CFR 20, 50 and 100 in almost paragraph by paragraph?

MR. KINTER: No, this is not part of the TMI requirements. As I understand it, it is simply a request I think at the Commission level that statements on this be clarified and that statements be made that reference portions of the application where it describes in detail how the rules are met.

MR. KERR: Thank you.

MR. CARBON: Excuse me, an additional question. You indicate, if I understand correctly, that you expect these issues to be closed in about a month.

MR. KINTER: Yes.

MR. CARBON: Do you know what they are going to say? Have they discussed these verbally with you?

MR. KINTER: Yes, I know how the review is going. I am in the communication chain between the reviewers and the applicant I know in general how they are going and I expect them them to be closed.

MR. CARBON: Have they submitted the material for all of the items on this slide?

MR. KINTER: No, they have not submitted the material but for some of them. I will find out the ones they have submitted and, as a matter of fact, I have word from the reviewers that some of them are closed. They just need to write the memorandum report.

MR. CARBON: Thank you.

MR. ZUDANS: Wouldn't under normal conditions the FSAR contain all these items already? The FSAR should make reference to all of these already.

MR. KINTER: The FSAR does make reference to all these, yes.

MR. ZUDANS: So what is the need for special issue of a summary? What does this item mean?

MR. KINTER: Well, the safety evaluation report lists in I think Section 1.8 these open items. Now, in order to record the basis for closing these items we will issue a supplement to the safety evaluation report targeted for August 31st in which we will record the correspondence that Detroit Edison gives us as a basis for closing these open items and our conclusions. Some of these may not be closed. It is my judgment they will be closed at this time.

MR. ZUDANS: Thank you.

MR. KINTER: Regarding the second open issue, the characterization of ground motion for the safe shutdown earthquake and the design response spectrum was not consistent with currently accepted spectrum. Detroit Edison generated an acceptable response spectrum and made a reassessment of desi margin and equipment structures required for a safe shutdown.

Jim Knight, Assistant Director for Components Engineering in the Division of Engineering will report on our evaluation of the seismic reassessment under Agenda Item 2(e).

Detroit Edison has submitted its program for a preservice and in-service testing of pumps and valves. We expect to complete our review in time for inclusion in the supplement to the safety evluation report. At this time I do not krow of any need for additional information in this review.

The seismic qualification review team is planning to audit seismic qualification records for 25 pieces of equipment during the week of July 27th. About half of this equipment is that required for safe shutdown and therefore will be evaluated as a part of the seismic reassessment.

MR. KERR: What is a seismic qualification review team? Is that a team that you have in-house?

MR. KINTER: Yes. The seismic qualification review team is led by a member of the Equipment Qualification Branch, Arnold Lee, and there are some consultants. That is, there is another consultant that is workin with him on that but under the NRC guidance.

MR. KERR: Is this something special about Fermi-2, or does this seismic qualification review team audit all plants now?

MR. KINTER: The seismic qualification review team has been used over the past several years to review the equipment.

MR. ZUDANS: Just to make sure I understand, is it the audit of the team or the team audit? What is going to be audited, the team or the equipment by the team?

MR. KINTER: The equipment will be audited by the seismic qualification review team. It is not properly phrased.

The next item we have completed. We have completed our review of the buried pipe foundations since publication of the safety evaluation report and we have concluded that they are acceptable. This additional information which provided a basis for acceptance will be described in the supplement to our safety evaluation report.

The next item, information from General Electric to demonstrate conformance of nuclear steam supply components to Appendix G and H to 10 CFR Part 50 or to justify exemptions to Appendix G and H, will be filed by August the 1st, 1981. Information on the balance of plant components has been received from Detroit Edison.

Some aspects of leakage tests of containment isolation valves have not been satisfactorily described by the applicant. We expect satisfactory information to be filed by August the 1st.

Procedures for testing residual heat removal valve interlocks have been provided to us by Detroit Edison. This is another issue that is resolved.

The NRC staff is currently reviewing information on the loss of instrument and control power. That is I&E Bulletin 79-27. This information was submitted by the applicant since preparation of the SER was completed.

We will report on our conclusions in the supplement to the SER.

The next item is our fire protection review. Fire protection systems have been evaluated for Fermi-2, including a site trip to examine the relationship of safety related components and structures in specific plant areas to both combustible materials and fire detection and suppression systems. Fire protection for all areas except the control room are acceptable.

Detroit Edison has conducted a fire test to demonstrate that a fire external to the control panels in the control room will not result in loss of redundant shutdown functions.

We have identified four apparent deficiencies in this test which are listed in the safety evaluation report, Appendix E. These are inadequate simulation of plastic components, inadequate fire configuration, inadequate simulation of panel ventilation and inadequate simulation of the use of fire suppressants. We have requested a report to demonstrate that the tests realistically simulated the control panels.

MR. KERR: I am sorry. You have requested another test?

MR. KINTER: No, we requested a report on the results of the fire test. One of the things we wanted them to do was to answer these four areas we have identified as deficient in the safety evaluation report.

MR. KERR: So the deficiency you think was not in the experiment but in their description of it?

MR. KINTER: Well, let's look at those four deficiencies. Inadequate simulation of plastic components ---

MR. KERR: No, I understood what you said initially and what you said initially led me to believe that they had inadequately simulated something and I don't see how you can correct that by rewriting a report. So my question was do they have to redo the experiment or just rewrite the report?

MR. KINTER: That may be one of the outcomes, just to redo the experiment. But they may be able to describe their tests, the prototype tests on the panels and justify and convince us that they do adequately simulate. That is another possible outcome.

The next item, the physical security plan and their two related plans, the guard training and contingency plan, have been reviewed and minor changes were requested. We expect revisions to the plans to be filed by August the

1st.

The next item, selected emergency operating procedures have been commented on by the staff and reviewed by the applicant. A walk-through on a simulator is scheduled for this weekend and a walk-through at the plant on July 28th.

MR. CARBON: Les, I guess I don't get the connection between your walk-through and the simulator. Would you go through that again?

MR. KINTER: Well, the emergency operating procedures are first reviewed by our people and consultants and then again it is a team review. Comments are made and then Detroit Edison revises the procedures incorporating those comments which are appropriate and applicable after discussion with us.

Then the next step is to take those emergency operating procedures to a simulator and simulate the Fermi-2 plant. This will be at Chattanooga, Tennessee, in the Browns Ferry simulator. Then the operators will walk through or operate the simulator using the emergency operating procedures.

MR. CARBON: Okay, fine.

MR. KERR: How many licensed operators are there in the group that is reviewing the operating procedures?

MR. KINTER: You mean on our side?

MR. KERR: Yes, sir.

MR. KINTER: I don't know.

MR. KERR: Are there any who have been licensed operators?

MR. KINTER: I don't know the answer to that.

MR. KERR: I will ask the same question again at the full committee meeting.

MR. KINTER: The next item, we are awaiting more information from Detroit Edison on the feedback of operating experiences. Mr. Eric Pederson of our Licensee Qualification Branch in the NRC can provide comments on this if desired under Agenda Item 2(a).

MR. MOELLER: I would like to hear something about that.

MR. KINTER: About our comments on feedback of operating experience?

MR. MOELLER: Yes.

MR. KINTER: Would it like it now?

MR. MOELLER: No, under Item 2(a).

MR. KINTER: All right.

The next item is on control room design and review. This review has been completed by a Nuclear Reactor Regulation Inspection and Enforcement review team, except for five items. There are five open items resulting from that review and these are identified in the safety evaluation report, Appendix D. Three of these are expected to be completed on August the 1st. One of them is a control room evacuation alarm signal. There are some modifications that will be completed later in the fall. Processed computer software won't be completed until nearly time to start up in August of 1982.

The next item, a degraded core training program, a description has been requested by the Nuclear Regulatory Commission. They have committed to provide the training and the schedule is committed to, but we want to know more about the program.

The next item, an audit of the containment purge valve operability documents, that is reports, design reports, will be conducted to assure valves will close under accident pressure, to assure that stresses are acceptable in the valves and to assure that the valve operators are operable under accident conditions.

MR. CARBON: Was that a review of reports and write-ups or review in part of actual tests?

MR. KINTER: Well, reports on the test results or reports of analyses, as the case may be, or maybe both to show operability of the valves. There may be stress reports and stress analyses to demonstrate that the valves will hold up under those conditions. Part of it will be a review of tests of valves being closed to show that they will close.

The tests have already been run I believe. We will review the work on that.

MR. KERR: Does that complete your presentation, Mr. Kinter?

MR. KINTER: I did have a couple more open items.

MR. KERR: You have a minus four minutes.

MR. KINTER: Let me just then not go into these very much but just say that these items are to be completed prior to the operating license issuance, these five items. They are longer term items. These items are license conditions. The fifth one, the analysis of the effect on high energy line break on control systems, has been resolved. We won't make that a license condition.

MR. CATTON: The second item, test of the fuel cannel box deflection, is there something unique about Fermi-2?

MR. KINTER: No. As a matter of fact, I think it is a license condition. It is not treated as an open issue. It is not really an open issue. What this is is after long burn-up, say after the first fuel cycle ---

MR. CATTON: I understand what it is. I am just wondering why it is here. I am wondering why Fermi puts it here when we haven't seen it for Susquehanna, for example.

MR. KINTER: That is right, and I don't think Susquehanna showed you the license conditions. Perhaps that

is the reason, but it will be a license condition on
Susquehanna, too.

MR. CATTON: You go through this exercise for every plant?

MR. KINTER: Yes, all boilers, yes.

MR. CATTON: Exactly the same thing for every plant?

MR. KINTER: Yes.

MR. KERR: Mr. Moeller.

MR. MOELLER: I had two general questions. A follow-up on the matter of interacting with the Canadians. We have received this week or recently two SERs, one for Susquehanna and one for Fermi-2. The SER for Susquehanna when you deal with the meteorological section has a wind rose in it. The SER for Fermi-2 does not. That is the first time I can ever recall a wind rose not being provided. Was there some reason for this or why was it suddenly deleted because that is very fundamental information?

MR. KINTER: I am not aware of any reason for leaving it out of Fermi-2. I am sure it is in the final safety analysis report but I don't know of any reason for leaving it out of our safety evaluation report.

MR. MOELLER: Well then if you read the text carefully you can find that the prevailing wind is from the southwest which means it blows toward Canada and a wind rose would have quickly shown you that.

Also I have a question along this same line.

MR. KERR: You realize that in that area Canada is south of the United States, don't you?

MR. MOELLER: I had another question and this is something I would perhaps like to hear later today and perhaps containment is the place to do it. When the committee a month or two ago reviewed LaSalle, which is a BWR 5 with a Mark II containment. Today we are reviewing Fermi which is a BWR 4 with a Mark I containment.

Now, if you compared the routine releases of air-borne effluents from these two plants you will find that LaSalle which is a Mark II BWR 5, a more modern unit, far exceeds the amounts being released or projected to be released by Fermi-2. They are both at comparable power levels and I just wondered if someone -- well, this would be better for the staff, of someone from the staff could quickly sometime today explain to me why this is so.

MR. KINTER: I will call someone and get someone over this afternoon.

MR. MOELLER: Thank you.

MR. KERR: Does that complete your presentation, Mr. Kinter?

MR. KINTER: Yes, it does.

MR. KERR: Are there other questions?

Mr. Ray.

MR. RAY: On your first slide you indicate a reactor rating of 3292 megawatts thermal. I presume that is the basis for the OL?

MR. KINTER: Yes.

MR. RAY: In the March 9th, 1971, ACRS letter to the then Chairman, Commissioner Seaboard, on the construction permit, paragraph 3 refers to this as the initial power. It says "The applicant's analysis and our evaluation have however been based upon the ultimate power of 3428."

MR. KINTER: Yes.

MR. RAY: Is this still the characteristic of this installation?

MR. KINTER: The engineered safety features are evaluated at the higher power level but the rating of the plant is 3292 megawatts thermal.

MR. RAY: Is there ever any probability of their asking for a stretch into the 3428?

MR. KINTER: I don't think for the initial license, no. They may, I don't know.

MR. RAY: But at this time it is not comtemplated?

MR. KINTER: No.

MR. RAY: Thank you.

MR. KERR: Other questions?

(No response.)

MR. KERR: Thank you, Mr. Kinter.

This brings us to the Detroit Edison's presentation, site and plant description.

Mr. Tauber, you may proceed.

MR. TAUBER: Good morning. My name is Harry Tauber and I am the Vice President of Engineering and Construction for the Detroit Edison Company.

I would like to introduce Dr. Wayne Jens who is the Vice President of Nuclear Operations for Detroit Edison. He will be talking to you later. Wayne and I represent senior management at this presentation.

The way we would like to conduct this is to respond to all of these items that were listed by Mr. Kinter and later on we will turn that over to Mr. Colbert to go down that list item by item and cover them.

Also Mr. Schuerman will come up here and give a description of the plant and the site.

I would like to make a few introductory remarks about the company and the way we have approached this project.

First of all, we view this as a very significant and important meeting to us because we have spent a lot of years and a lot of money getting to this point. If we look a little nervous this morning we are because it means so much to us. Although we are nervous about this meeting, we are very confident about the fact that we know what we are doing and that we have designed a safe plant.

At the present time we have some $1,200 million invested in the project. The estimated cost is $2 billion. As you know, the fuel loading date is the end of 1982 with a commercial operating date at the end of 1983.

The company, with that kind of an investment, believes that the viability of the whole organization depends on the success of this project and we have applied practically all of our resources to assure that it is accomplished. We have reduced our capital program, we have taken key people and assigned them to this project and are applying the resources to assure that we can complete it on time.

At the present time the project is slightly late, "slightly" being in the order of two to four months which we expect to recover by the application of the resources that I just mentioned.

We believe we have some unique talents applied to this project. First of all, the Detroit Edison Company is basically an engineering company. Its executives historically have been engineers and today they are engineers. Our chief executive officer-elect who takes over on September 1st is an engineer with a strong nuclear power background. The new president is also an engineer who has had the Fermi-1 exerience.

We have a long history of having designed and constructed our own power plants. This is true in this case partially. Detroit Edison is the designer of the Fermi-1 project. We are using others to construct it for us, although recently we have taken a more active role in its construction.

I mention that point because we will not be facing a situation where an architect/engineer who desicns and builds the plant for you leaves when the plant is completed and leaves the utility without the proper resources and knowledge that it may need. The very people who are designing this plant are the same people, or many of them, who will wind up operating this plant. That to us is a tremendous asset.

Even though we began construction of this plant some 12 years ago it is an up-to-date plant. The reason we have been in the field so long is because we shut the plant down for a two-and-a-half-year period after the 1974 financial crisis that hit utilities where Michigan was particularly hard hit because of the lack of regulatory relief. That shutdown cost us more than two-and-a-half years. It cost us perhaps four or five years. That is the reason why you see this plant coming before you now even though it is an older version of the plant and why it has been in the field so long.

So the point I would like to make is that we have the Fermi-1 experience behind us and we have the experience of having designed our own power plants for many, many years. We have an able engineering organization who are assigned to this project who have the Fermi-1 experience and many of those same people will wind up in our operations organization. Those 80 consultants that we do have when they leave we will not be left empty and without the expertise.

You will note today in the presentations we make you will be talking to Detroit Edison personnel primarily and not consultants.

With that I would like to introduce Larry Schuerman who will describe the site and the plant.

MR. KERR: Any questions for Mr. Tauber?

(No response.)

MR. KERR: Thank you, sir.

MR. SCHUERMAN: As Harry said, I am Larry Schuerman and I am Detroit Edison's nuclear licensing engineer for the Enrico Fermi atomic power plant Unit 2.

We are indeed glad to be here this morning.

What I have prepared and will now present is a brief description of the plant and some of its features.

(Slide presentation.)

Fermi-2 is co-owned by the Detroit Edison Company and two cooperatives, Northern Michigan Electric Cooperative, Incorporated, and Wolverine Electric Cooperative, Incorporated. The 1,120 acre site of Fermi-2 is located on the western shore of Lake Erie in Fringetown Township, Monroe County, Michigan.

The plant is approximately 30 miles southwest of downtown Detroit and 25 miles northeast of Toledo, Ohio. Approximately 90 percent of the land area within ten miles of the plant lies within Monroe County. The remaining ten percent is in Wayne County.

Of the area in Monroe approximately 55 percent consists of farmland. Within a 50-mile radius of the site are all or portions of 11 counties in Michigan and ten counties in the State of Ohio.

The 1980 census data showed a population of 84,600 within ten miles of the plant and 5.5 million within 50 miles.

MR. MOELLER: Excuse me, I am pursuing the point now. Does the 5 million include the people living in Canada?

MR. SCHUERMAN: Yes, it does.

MR. MOELLER: Thank you.

MR. SCHUERMAN: Fermi-2 uses a General Electric boiling water reactor of the BWR 4 class with a pressure suppression Mark I containment. Fermi-2 is similar in design to other nuclear units presently in operation, including Browns Ferry Units 1 and 2, Edwin Hatch Units 1 and 2 and Brunswick Units 1 and 2.

In addition, the Fermi-2 design has been improved in response to the lessons learned from Three Mile Island Unit 2 and industry experience. Some of these improvements will be discussed during our presentations later on today.

The designed thermal power rating is 3428 megawatts with a turbine generator producing a rated output of about 1,154 megawatts.

Two natural draft cooling towers provide the normal heat sink for the closed cycle cooling system. If this heat sink is not available, the reactor can be safety shut down and maintained using the mechanican draft cooling towers and reservoirs in the residual heat removal complex. That complex appears in the center part of the slide here. It is this structure right there (indicating).

MR. RAY: Question.

MR. SCHUERMAN: Yes.

MR. RAY: Are both cooling towers needed to be operating for full power in the plant?

MR. SCHUERMAN: Under the design basis, yes, they are.

MR. RAY: If you lose one what capability do you have left and can you operate with one?

MR. SCHUERMAN: The plant can be operated with one. We lose capability and I don't know how mu.... We will find out and get that information back to you later on today.

MR. GRIFFING: That would depend on the weather primarily.

MR. KERR: Would you identify yourself, please.

MR. GRIFFING: I am Edward Griffing, the plant superintendent for Fermi-2. My answer to the question was it would primarily depend on the weather whether you would need the maximum cooling tower capability.

MR. RAY: I am curious as to what it might be with the worst weather conditions.

MR. SCHUERMAN: I don't know.

MR. KERR: Mr. Moeller.

MR. MOELLER: When you were listing the other units that are similar to Fermi-2 were you referring simply to the reactor or to the reactor and containment combination?

MR. SCHUERMAN: To both the reactor and the containment. I believe all three of them are Mark I containments.

MR. MOELLER: You had mentioned of course the Browns Ferry units but you mentioned also Hatch Unit 2 and my understanding on it was it had a Mark II containment.

MR. SCHUERMAN: I made a mistake.

MR. MOELLER: Thank you.

MR. KERR: Other questions?

(No response.)

MR. KERR: Please continue, Mr. Schuerman.

MR. SCHUERMAN: The reactor core contains 764 fuel assemblies and 185 control rods arranged in a vertical upright cylindrical configuration. Each fuel assembly consists of an 8-by-8 array of rods, 62 of which contain fuel and two contain water.

The emergency core cooling system in conjunction with the automatic depressurization system provides the capability for high and low-pressure coolant injection and core spray. These systems provide both redundancy and diversity.

The plant's electrical distribution system is designed to provide normal and standby sources of electrical power to permit safe shutdown and to maintain the plant in a safe condition.

In addition, the power sources are adequate to accomplish all engineered safety feature functions required under postulated design basis accident conditions.

The Fermi-2 facility employes rad waste systems designed to limit the dose to the general public due to radioactive effluents to levels which meet the design objectives of 10 CFR 50 Appendix I and are as low as reasonably achievable.

In addition, the Fermi-2 facility design incorporates features which minimize the occupational radiation exposure under normal and postulated accident conditions.

A large number of experienced personnel are available at Detroit Edison to support the operation of Fermi-2. Most of the nuclear operations personnel are located nearby the plant site.

Finally, Detroit Edison mana, ment, as you have already heard, is committed to the safe and efficient operation of Enrico Fermi-2.

This concludes my prepared remarks and I would now like to turn the Detroit Edison presentation over to Bill Colbert who is the Technical Director on the Fermi-2 project.

MR. KERR: Are there questions for Mr. Schuerman?

(No response.)

MR. KERR: Mr. Schuerman, in your opening remarks you mentioned a 3400 figure as a design. What is the licensing figure in your view?

MR. SCHUERMAN: That would be the requested power level in the license.

MR. KERR: Then I might have misunderstood Mr. Kinter because I thought he used the figure 3292.

Did I misunderstand, Mr. Kinter?

MR. KINTER: No, you didn't. That was my understanding that 3292 would be the licensing basis.

MR. SCHUERMAN: I will just have to check the numbers.

MR. KERR: I don't need an answer today, but it would be nice to know what the number is.

MR. SCHUERMAN: It certainly would.

MR. KERR: Thank you.

The number you used was 34 and what were the last two digits?

MR. SCHUERMAN: 3428. That number in the writeups that you have is 3430. We may get a little bit sloppy in some of our numbers when we get to that level of power and tend to round off. But I used 3428 when I spoke a moment ago.

MR. KERR: That was the number I wanted, the number you used. Thank you.

Mr. Colbert.

MR. COLBERT: Thank you, Dr. Kerr. Good morning.

In the interest of saving some time here, and I believe we can, Mr. Tauber said we would go over the items point by point. In listening to Mr. Kinter, with a few minor exceptions, we agree with his statement of the condition of the open items. We understand the open items to be what was stated by Mr. Kinter and the status of the open items, those places where we will have the information completed as a result of I guess the close communication we have had over the last few months.

Referring to Mr. Kinter's sheet, the one where he says we are going to have the items completed in the SER by August 31st, 1981, a bit of new information that I just received yesterday would indicate that the last item on that list we may not be able to support that. I told this to Mr. Kinter this morning when we met here. That was some new information.

It has to do with one of the corporations in question that we had to get some information from. They are going on a two-week vacation where they shut down the facility.

MR. MOELLER: What is the specific item?

MR. COLBERT: The specific item is the containment purge operability of the one that he was talking about getting back the test results. That is the only one we appear to have a little schedule problem on.

There is one other item that was discussed, very quickly, which was the first protection in the control room where we will file the report on the scheduled date.

The question of whether or not we would be likely to volunteer another fire test of the panel is something else again. I presume that that item may require further discussion with the staff before we make that decision.

The page that Mr. Kinter showed talking about to be completed prior to OL issuance in November 1982, we understand each of those items. The scheduled information is as we understand it and we believe we will be able to meet all of those.

The page that was talked about to be completed after license issuance, that is licensing conditioning, as we understand the first three items on that page, one of which was questioned, are in effect a requirement for all BWRs at this point in time. We understand that.

The four item, which is analysis of multiple control system failures, the item for discussion with the staff on that one is not whether or not we have analyzed the multiple control systems failures. It would be I believe, my understanding of where we are there is a matter of the documentation of that particular item.

We, of course, have previously continued to analyze our plant versus the industry experience and versus our own questioning minds.

The other item on that page -- in fact, I guess we are now saying the first item on this whole list of which we disagree with the staff happens to be the last item on the page which is design of instrumentation for adequate core cooling, specifically the use of input thermocouples on Fermi-2, is an item is a subject for discussion a little bit later today.

MR. MOELLER: Could you remind me of the design of the modification to the diesel engines? What is that? Your first item on the items to be completed prior to OL issuance in November of '82 is "design of modification to diesel engines." What is that?

MR. COLBERT: The point which was open and which we were not able to respond to in time to get it into the FSAR was an item dealing with the keep wetness, if you will, of the engine bearings, the lubrication system in the engine. The staff has requested that we have a modified system that will prevent dry starts.

MR. MOELLER: Thank you.

MR. COLBERT: That, by the way, is underway with our vendor on those particular units.

Larry, I am up here without the agenda. Could you give me the agenda sheet, please.

MR. KERR: The agenda item is a response to SER open items. Do you need more detail than that?

MR. COLBERT: That is the agenda item we are speaking to. The SER open items I think were described by Mr. Kinter.

MR. KERR: I don't want you to overdescribe them.

MR. COLBERT: I have completed describing it.

MR. KERR: All right. I will ask if there are questions. Are there questions?

Mr. Catton.

MR. CATTON: I would like to hear a little bit about degraded core training at some time.

MR. COLBERT: That is on the agenda.

MR. KERR: Mr. Carbon, did you have a question?

MR. CARBON: I would like to ask Mr. Kinter, are there any items in this review that are in contention as far as within the staff, differing opinions?

MR. KINTER: I don't know of any items where we have instituted a professional differing opinions procedure. You know, we have a formal procedure for that.

Most of these areas are resolved within the branches. If there are differing opinions they are resolved within the branches.

On the fire test Bob Ferguson had a differing opinion and perhaps the consultant, too, on the criteria for the fire test. This is within NRC.

MR. CARBON: This is the only significant difference that you are aware of?

MR. KINTER: This is the only one I am aware of, yes.

MR. KERR: Other questions?

(No response.)

MR. KERR: Thank you, Mr. Colbert.

MR. COLBERT: I would like to call upon Wayne Jens, Vice President of Nuclear Operations, who will discuss the area of organization management on the Fermi-2 project.

MR. KERR: Mr. Jens, I want to warn you that yesterday when I asked the chief executive of another utility I was told that he had a master's degree in engineering, a MBA and a Doctor of Jurisprudence. If you can top that ---

MR. JENS: I am not even going to try, Dr. Kerr.

(Laughter.)

First of all, I would like to reinforce what Mr. Tauber just said that this is a great pinnacle in our achievement on Fermi-2. We have been at it a long time and we are pleased that we finally made it up to talk to you.

I would like to elaborate a little bit more on what Mr. Tauber talked about because I would like to talk about the structure of the nuclear operations organization and why we have so many people that will be in that organization, about 500.

In order to do that I would like to develop a little bit of the history of our involvement as a company in nuclear power. We were one of the original utilities in the early power demonstration program under the Atoms For Peace

Program. At that time we had associated ourselves with the Dow Chemical Company. That led ultimately to the development of design and construction of the Fermi-1 plant.

As you know, there were two organizations created for that, the Atomic Power Development Associates, the developer and the lead design company, and the Power Reactor Development Company that operated the plant.

Many of the people in key positions now on our project and in the company had a long association with that project. Early in 1970 during the time when we were winding down on that project and our involvement in it all of these people entered Detroit Edison, most of them did, and took these positions to help in the design of Fermi-2.

Mr. Tauber gave you a few examples of that. Our recently elected chief executive officer, Mr. Walter McCarthy played a leading role in Fermi-1 and he ended up in the project as the general manager of PRDC and then went on of course to become the project manager of Fermi-2.

I ended up at various spots in that project ending up as general manager of APDA and then I took over as the second project manager on Fermi-2.

In addition to our involvement in Fermi-2 as the lead designer, our company also initiated work on a second unit at Fermi, Fermi-3 which originally was to be a duplicate of Fermi-2. At the time it was a duplicate I was project manager and it was a simple matter to continue in that capacity.

But we converted to a BWR 6 Mark III containment during the course of that work and Mr. Fahrner then became the project manager and he is the project manager now on Fermi-2 because we cancelled that project.

Our other involvement in nuclear power during those years was the initiation of two nuclear units at our Greenwood Energy Center, Greenwood 2 and 3. Those were terminated after the termination of Fermi-3 in 1980.

Now, in order to handle all that work of course our company not only had to staff up for its direct responsibility on Fermi-2 but, because of the load, we of course hired architect/engineers to handle Fermi-3 and Greenwood 2 and 3. But that did not mean that we didn't have involvement. We specified what we wanted, we did licensing work and we evaluated the work that people were doing in the design area. So we had to have an additional cadre of people.

Following the termination of that work of course these people then became a very valuable resource to help Mr. Fahrner and Mr. Tauber complete the construction of Fermi-2 and will also be a resource in the operations of that plant, Fermi-2.

One of the other valuable things that came out of that of course is that we have one single purpose in the nuclear power business at this time in the company. That is the successful completion of Fermi-2 and its successful operation. There is no other diversion of that talent and we can concentrate on that particular effort.

It was about a little over a year and a half ago when I held a position very similar to Mr. Tauber. I was the assistant vice president of engineering construction. The then president, Walter McCarthy, came to me and asked me to evaluate what we should do about the operation of Fermi-2 in view of the experience that had been gained at Three Mile Island. So I undertook that particular job to study and make a recommendation to the company about what to do.

Up to that time it had been anticipated that Fermi-2 would be operated as a conventional plant in our production department under the vice president of operations. I felt that I needed help in order to make that study and I engaged to consultants, one consultant that was intimately involved in the TMI experiences and the lessons learned, Mr. Lou Roddis, who had also experience as the vice chairman of the board of Consolidated Edison.

In addition, I felt that I would like the experience of somebody that has been in the operating end of nuclear power plants and I engaged the former president of Yankee Atomic Electric, Larry Minnick, to help me with this. They both were invaluable resources for what we came up with.

We of course had all of these assets to work with. As Mr. Tauber said, our company is technically strong. We have 11,000 employees and 1,100 of those are professionals. We have a strong engineering research department that is oriented to problem solving for our operating plants and operating electrical system. It is not a conventional research department but it is engaged in problem solving and it has laboratories in order to do that. These resources of course are available to us in the nuclear business in the company.

We made certain conclusions, all of which led to recommendations and they have all accepted and we are implementing those. Let me just touch upon them.

We recommended that we separate our nuclear from our fossil operation into a separate organization, that we integrate all of our nuclear activities under one organization when Fermi goes into operation, that all safety related functions be controlled by this organization in a very careful way and that the emphasis in the new organization be on training, one of the lessons that I think we have learned from Three Mile Island. This meant that we must elevate the training function to a very high level in the organization.

That since we have a single purpose, we can focus our attention on it with one organizational unit and that also means that we can locate those human resources right at the plant site. That was another recommendation that is being implemented.

That we should develop, because of that, a very intimate relationship between those people that design the plant, specified it and know what they intended it to do, the criteria, the requirements, with those that will have to operate the plant.

Shortly after the accident we put together a team of operating people, including some shift supervisors and the plant superintendent, at the time to work on a safety review of the plant working with our system engineers under the direction of Mr. Colbert. That intimate relationship was demonstrated during that study of the value of it in conveying information from those that designed the plant to those that will have to operate it. We wish to continue that through this organizational structure that was recommended.

We also recommended that we buy a simulator especially for this plant and that we build a new building to house a simulator and all these people that we are going to have supporting the plant. We are involved in building a hundred thousand square foot facility to do that and that we get some commitment from the company to staff it with the best people we have available in the company. I can assure you from my experience to date that that commitment is also being met.

With that in mind I would like to show you what the organizational structure is.

(Slide.)

This was done independent of NUREG 0731, but I think for those of you who have studied it you will recognize that it is very similar. I am gratified that it is because I won't have to argue why there should be differences.

Basically you will notice that what I said about training, it is elevated to the same level as the plant itself, on the same organizational level. I think that is somewhat different than has been done in the past and it places a great deal of emphasis in the minds of everybody that training is very important.

The names in the boxes I am going to mention in a few minutes, but they also indicate who are the people presently in place. Of course, we have myself in place, we have a manager in place, we have a director of administration, the director of training, the director of nuclear engineering and a plant superintendent.

The two unfilled boxes here will be filled about the time of fuel loading. Those people are presently engaged. They haven't been identified because we have quite a few resources that we can call on in the project management organization. These are the people that really schedule the work that has to be done for an outage, the same thing as scheduling the work for a project and then to carry out the design and construction work for modifications is very similar to building the plant. So we have those resources now available but we don't want to call on them in this organization until absolutely necessary because they have a big job to do still down at the site to finish it up.

The quality assurance group, we have a group that had been reporting to the plant superintendent. We are going to take the quality assurance people that we have there and there are about 10 people and put them into our quality assurance corporate organization. They are going to develop an organization that they turn over to me about four months prior to fuel loading that meets my requirements which I have specified. We have an agreement that they will do that.

This will mean that we will inherit many of the people that are now in quality assurance for construction which will be needed for quality assurance in plant modifications and that type of work.

MR. CARBON: Question.

MR. JENS: Yes.

MR. CARBON: On the figure you just have the director of quality assurance reports to you.

MR. JENS: Correct.

MR. CARBON: On Figure 6 a manager of quality assurance reports to the executive vice president above you.

MR. JENS: Correct.

MR. CARBON: I am a little confused.

MR. JENS: Figure 6 is the corporate structure of the organization and it represents what exists right at the present time. It does in with the statement that I made that this director presently will be reporting to the manager of quality assurance until roughly the end of 1982 at which time he will then report to me and the corporate quality assurance manager will then audit our entire operation as an independent function. It won't go away, but there is going to be a direct nuclear quality assurance group reporting to me and that is this particular group.

Now let me say a few words about people because you can have the best organizational structure in the world and without good people it is kind of a hopeless process.

Let me begin with Mr. Willard Holland. One of the problems I felt I had after they accepted the recommendation and said you are it, now I had the problem of building from nothing to 500 people. I recognize that that in an organization as solid and as well established as Detroit Edison might not be very easy.

So I looked around for someone that had don' that before. Over the past several years we have created new divisions in the company and one of the leading men in that activity was not only my boss, Mr. Heidel, the Executive Vice President of Operations, but a manager that existed in the company named Willard Holland who worked for Mr. Heidel.

Mr. Holland also had the unique capabilty of having been Assistant Manager of Projection and that is who the plant people had reported to prior to this reorganization. So there was a continuity involved in that.

In addition, Mr. Holland had been Assistant Superintendent of our St. Clair plant, a seven unit fossil plant up on the St. Clair River. So he has had some operating experience in addition to this managerial capability that he had in how to manipulate people from one organization to another.

He has been an invaluable resource because I didn't recognize really how difficult it was going to be and how much I really needed him.

Let me go on to the next person on the table. As I said, we wanted the best possible man for nuclear training. We wanted to elevate that to a very important function. Well, to do that we had to have credibilty in who we put in the spot.

Within the company we had a Director of Employee Training covering all the training within the company. This man had been associated with Fermi in the early days as a member of the Human Factors Group in the design of our control room. I think we are pioneers in the utility business for having used human factors in control rooms because we have been doing it for the last nine power plants and Al Kanous has participated in that.

Furthermore, he is a real professional. He is an industrial psychologist. He has a master's degree in that field. He has completed all the requirements for his Ph.D. except a dissertation. He has had training as a behavioral scientist in the Air Force and has been with the company 30 years in various capacities. You will get to meet him because he is going to present something in a few minutes to you.

Again nuclear engineering is an important activity, the continuity of that expertise. That position is filled by Elton Alexanderson. He has a master's degree from the Oak Ridge School of Technology. He was associated with Fermi 1 and in APDA he was responsible for the summary report which I think is one of the first that was ever written. It was pretty thin in those days.

He became reactor engineer and then assistant superintendent. So he has had operating experience and that is what is important in a nuclear operating organization. He became the assistant general manager and general manager and was one of the people that has decommissioned a nuclear plant, particularly a breeder, and did it for very minimum cost, not any of the numbers are are talking about today.

For the past five years he has been the Director of Nuclear Engineering in our Generation Engineering Department. Again, a very solid citizen that will stabilize our operation.

Now, the superintendent up until four months ago had been Mr. Bill White. For some of you who don't know who Bill is, he was the last plant superintendent at Fermi-1 and was the second appointed superintendent at Fermi-2. The first man designated had passed away. Bill spent a great deal of time developing that organization and did a very valuable job in gathering into the organization some very key people, one of whom is here today and you have met him. He stood up a minute ago. He is presenting the superintendent of the plant. He acted as mentor to Mr. Griffing. Mr. Griffing is the N vy Commander that came to us four years ago. He has been involved in the start-up of the plant.

Because the timing on the retirement of Mr. White would have coincided with our fuel loading and operation, we decided just recently not to do that at that particular time to meeting this transition early. This not only gave us an opportunity for Mr. Griffing to have his presence felt as the superintendent during this very critical period but it also released Mr. White to go down to the Institute of Nuclear Power Operations to impact their group on his past experience.

Now, in the few minutes remaining I would like to address a few of the points of contention I guess that we might have had during the negotiations with the staff.

In general the staff agrees that our organizational structure, the people we have more or less meet the requirements in NUREG and the guidance of 0731.

They were concerned, and I will give you just a few of the concerns, I won't give you all of the positive things, I have already done that, that the transition from this PMO operation to nuclear operations may lead to some questions of safety.

Let me assure you that I am conscious of their concern. I would be concerned about it myself. But let me also assure you that the functions that these people will play when they make this move are essentially the same functions they are making today. The people that are doing design work will continue to do design work. We are not going to take designers and make operators out of them. We are not going to change their function and therefore if they inherit some responsibilities in the move they are just reporting slightly to a different vice president or a different manager than before.

I don't think it will impact safety because I am as concerned about safety as Mr. Tauber is or Mr. Griffing is as concerned as Mr. Fahrner. So I don't see it as a major problem.

Another issue was the BWR experience we have. We enumerated how many people, independent of the people we would send off to observe or spent some time at at BWR, how many were actually in a plant under the direct responsibility of other people. We could only name about six people. Six people is quite a few I think, but perhaps not enough. It didn't convince the staff that we had enough. So they asked us to commit to having additional BWR experience on shift which we have done.

There was only one contention left, that they would like to have us have those people there until we reach 100 percent power. Because I am concerned that we might not quite get to 100 percent or that it may take an extraordinarily long time to do it that I didn't want to have the people that we were talking about, namely, GE engineers, on our staff for that length of time.

I think that there is a way of writing that so that my concern will disappear. So I don't think it really is a concern. I certainly would say, yes, we are going to have them there until we get to high power if we can define that in some way.

The other true lesson learned I think at Three Mile Island is the one that you have identified on your agenda. How do you transfer information around to people and do it so that they really get it and understand it? Obviously it isn't an easy thing to do because it wasn't done properly at Metropolitan Edison or at Toledo Edison at the time.

We have in our system engineering group that has been engaged in the design of Fermi being doing that from the very beginning. We were ambitious when we first started off that every LER that we could lay our hands on we would look at and review and see if it was applicable to Fermi and document our review.

That became an overwhelming activity as more and more LERs were written and more plants got into operation of course. So we had to restrict our reviews then to bulletins, circulars and any other summary information that outside organizations such as recently NSAK and INPO would give send to us.

It is our plan of course to continue that activity and we will organize in Nuclear Operations to do that. We are involved in doing that right now. Now, we are not going to look at every LER because again we just couldn't afford to do it and we just don't have the manpower to do it. We are grateful that we as an independent industry have undertaken an activity that at INPO and NSAK to get that done. As I understand it, we recently have transferred all of that responsibility now in our industry to INPO, the Institute of Nuclear Power Operations.

MR. CARBON: How large a group do they have on that?

MR. JENS: Well, we actually have a Detroit Edison man that has been doing that out at NSAK. As I understand it, there were about 30 or 40 people. Now, there was a similar but a smaller group at INPO and they are going to have to transfer some of those people, Dr. Carbon. So I am not 100 percent sure how many INPO people are presently in place because the principal work was really being done out at NSAK.

MR. CARBON: But it should like on the order of 40 people?

MR. JENS: I would expect that you would have to have at least that many to accomplish that job for the industry.

Incidentally, our company is a very strong supporter of these two organizations. I think they are absolutely essential to our success.

MR. MOELLER: Well now, you can transfer the responsibility or the actual day-to-day nuts and bolts review of the LERs to INPO or NSAK, but then whom do you have back at the ranch who is analyzing what they come up and making sure that it is applied to your plant?

MR. JENS: Let me attempt to describe that. As I mentioned, we are going to try to preserve somewhat the same methods we have been using in the past. We have had a group of what we call system engineers, engineers that don't get involved with the detailed design but understand the operation of the system.

These system engineers have been assigned various LERs as they come in by the head of the system engineering group for their particular speciality. He determines who should review this. Then after the review is made he would then review the review to make sure he concurs. If then this leads to a change in design, it is fed into our normal change process system.

That same method will be employed in the nuclear operations organization. Instead of having the head of system engineers do that, however, we are going to designate what we had called, and it is probably a misnomer, an LER coordinator. I think it is too narrow a connotation.

What it is is an operational information coordinator. He will receive not only the plant LERs, our own information, which now would become an important activity for us, at the moment we don't have that kind of information being fed in, it is only external, but he would cover both internal and external.

He would make an independent determination whether our own internal ones need review outside of the plant staff and the plant staff would make this review through our committed on-site review organization which consists of the chairmanship of the plant superintendent plus all of the section heads that report to him commonly called OSRO by our acronym.

They would review every LER. If that LER needs an independent review they would give it to the coordinator who is in the nuclear safety and plant engineering group. He would then determine who should review it independently. He can also determine that if they didn't ask for a review he is going to have it done anyway. That independence will be residing in his authority. If it is an unreviewed safety question he has to review it and this group has to make that review. If that is an unreviewed safety question, of course it goes to our independent review and audit group called IRAG.

Both of these groups had been in place prior to the recent amendments to our FSAR and we have added these particular groups because being committees they cannot really do very much analytical work. They can review analytical work but they can't do any independent analytical work. It is this group that will do independent analytical work, full-time people in that group.

MR. CATTON: Do I understand right that OSRO is made up of the directors across that line?

MR. JENS: No. It is made up of the director's section heads as we call them that report to the plant superintendent, the operations engineer, the technical engineer, the rad chem engineer, the reactor engineer -- help me out.

MR. GRIFFING: The maintenance engineer.

MR. JENS: And the maintenance engineer.

They gather on every LER to approve it and review it. If it leads to a design change or a recommendation for a design change, the plant itself cannot undertake that. That has to then be given to this group. If it is a procedural change they can proceed unless it is an unreviewed safety question and then why it comes over to this group.

MR. CATTON: What do the LERs have to do with operator error and so forth? That sort of ties it into training. Is there anybody from your nuclear training box?

MR. JENS: Well, they are invited to s_t in on these meetings. At the moment we haven't formalized it. But if there is a training issue it is up to our coordinator here to make sure that that information is imparted to our training director.

MR. CATTON: So the others need to recognize that if it is a traning issue then they would pass it on?

MR. JENS: Well, there would be two people that would have to recognize it. Either one could recognize it, the OSRO chairman or anybody in OSRO, plus the reviewer of the LERs, the independent reviewer of the LERs which we had called the LER coordinator, the operations information coordinator. So you have two places where that can be recognized.

MR. CATTON: Do you make sure of the EPRI note pad?

MR. JENS: Yes, we do.

MR. CATTON: Is that within this OSRO as well?

MR. JENS: That would be also within the authority and within the responsibility of this coordinator. He is also going to be our note pad coordinator. Now, that doesn't mean if that workload builds up that it is a single person. That could be a group of people.

MR. CATTON: What about this next one up, this IRAG?

MR. JENS: Those are hopefully people, not quite of your caliber but similar, outside consultants and also experts from within the company and from within our own organization.

MR. CATTON: Then the Nuclear ꜀fety Committee.

MR. JENS: All right. Let me address that a moment.

MR. CATTON: I am just moving up here.

MR. JENS: I understand. I was going to get to that.

MR. ZUDANS: Could I ask a question?

MR. JENS: Certainly, Dr. Zudans.

MR. ZUDANS: According to what you say under that picture, this will only handle your own LERs.

MR. JENS: No. The picture was structured that way, but I wanted to make it very clear that when we talked about LERs it included all external information coming into this system as well. All of the external information will be coming into this particular group for either analysis or dissemination to others within the organization. They are a control group of information to make sure it is analyzed and properly distributed and properly acted upon.

MR. ZUDANS: Didn't you just say before that you will rely on INPO and NSAK and to the individual LER review?

MR. JENS: If you consider that a moment, there is no way that an external group such as INPO or NSAK can make a judgment that a particular item applies to our plant, our specific plant.

What they can determine is that this appears to be a generic operating problem that you must address to see how it applies to you. So that every one of these summary LER reports that we get from INPO or NSAK must be analyzed by us to see how it impacts us. Somebody has designated this is an important thing for you to consider, but we still have to consider it. So it doesn't flow to anybody except this coordinator who makes sure that it is properly analyzed for applicability to us and who it applies to.

MR. KERR: Mr. Carbon.

MR. CARBON: If I may, one more question on that. I appreciate the problem of the manpower in looking at all of the LERs because I know there are tremendous numbers of them. I guess I am sort of a suspicious character and I tend to think that my own people in my own organization will do a better job than someone else. Do you expect to do any kind of review of bulk LERs or are you just going to look at ones that INPO says you need to? You said that you are going to rely on INPO, but I wonder if you really have hidden back there the thought somewhere that you are going to do at least some screening yourself?

MR. JENS: Well, we have a number of suspicious characters like you, Max, in our organization, namely, me.

I have reviewed them myself, and I know Bill Colbert looks them over as well as other people. But it is not part of the formal process. In other 'ords, I am not going to commit a group of people to review LERs. I am prepared to commit to this sort of thing.

That doesn't prohibit anybody in our organization from reading and trying to understand and making recommendations. We are a very open organization from that standpoint. But I think to undertake a much broader view is very difficult.

Bill.

MR. COLBERT: I would like to speak from personal experience for a moment on the incredible size of the job, suspicion or not.

My experience over several years of doing this was that there are a very few kernals in a great big bunch of chaff. In fact, I welcome the efforts put forth by the industry to weed through that.

When I did it, frankly they would come in in hugh piles. I would sit by the wastebasket and ---

MR. KERR: Mr. Colbert, I am convinced.

MR. COLBERT: Thank you, sir.

(Laughter.)

MR. JENS: Let me get on with two other points and then I will ---

MR. MOELLER: Another quick question. This LER process and so forth is already underway. You are not waiting until the plant starts operating?

MR. JENS: Well, it has been underway in the project management organization. One of the deficiencies that we are trying to correct as quickly as possible is the process of taking information from the PMO and passing it on to Nuclear Operations in closing the loop. So we are trying to fix that up while there are two organizations operating.

We are now writing the proper procedures and we have designated a person to undertake this activity so he can get geared up for it. It is for our ultimate organization.

MR. MOELLER: Thank you.

MR. JENS: There was a question about the Nuclear Safety Committee. As Mr. Tauber indicated, our company is very concerned about Fermi and I guess it is no wonder in view of what is happening down at Metropolitan Edison and GPU as far as the financial impact of that particular accident.

Our board of directors obviously are the ones most concerned and they want a very intimate involvement with the operation of Fermi-2. We have traditionally had people on our board that have had nuclear experience, Ken Nichols, for example, the first General Manager of the AEC, was on our board and Bob Backer, one of the first members of the AEC, was on our board.

We will have to find some adequate replacement for him and then the board would constitute a subcommittee to review and set policy as far as our operation is concerned. So it is going to be an overview policy determining function that our boa d wants to undertake and we are very prepared for that and welcome that because we need all the understanding we can get at that level. We have already a very good understandin all the way up the line, as Mr. Tauber said.

I think it is so essential that those people that have to provide funds today understand what we are faced with.

Now, one other point. I failed to really recognize Bill Colbert here. Bill is going to be the Assistant Director of Nuclear Engineering and will occupy that particular position in our organization. Bill Colbert is a rather unique individual because he was one of the shift supervisors at Fermi-1.

It is very rare, I think, that you find somebody that has had hands-on operating experience and that has taken the kind of responsibility to design a power plant the way Bill has. To have him available to Nuclear Operations is going to be an invaluable resource.

MR. CATTON: Somehow I missed something. Who did you say the people were that make up the Nuclear Safety Committee?

MR. JENS: Well, the Nuclear Safety Committee, I put it into the pass-out and you will see that it consists of many groups. The one group that we have designated who will have reporting to it the SPAs plus about five additional people that will do this kind of work.

MR. CATTON: Up at the board level? Up at the top.

MR. PEDERSON: Oh, here. These would be members of the board of directors. I am sorry.

With that I am finished. I am sorry I took more time but I felt that I wanted to tell you everything that I told you.

MR. KERR: Are there questions?

Mr. Moeller.

MR. MOELLER: Say that again. I am not sure I understand who the Nuclear Safety Committee is.

MR. JENS: It is a subcommittee of the board of directors. We haven't constituted it yet. It will be in place probably around fuel loading time. It will probably have about three members.

MR. MOELLER: What background will have to know about nuclear safety?

MR. JENS: It depends on who we finally nominate.

I gave you some examples of previous people that have been on the board and we will have to find proper people for that spot.

MR. ZUDANS: No members from any of the line organizations below the board?

MR. JENS: No. There may be consultants that the subcommittee has, but there will be no people, as far as I am concerned, from the operating organizations.

MR. KERR: Other questions?

(No response.)

MR. KERR: Thank you, Mr. Jens.

The NRC staff comment on Item 1 is an agenda item. Do you have comments, Mr. Kintner?

MR. KINTER: Yes. Mr. Eric Pederson has some prepared comments to make on compliance with NUREG 0731 and also he can comment on feedback of operating experience.

MR. KERR: He is going to do that within five minutes.

MR. PEDERSON: I am Eric Pederson from the Licensing Qualification Branch. A team of NRR and I&E made a management audit in May of this year at Detroit Edison. We listed first the corporate office and discussed the organization and supports for the Fermi-2 plant.

For the next two days we went to the plant site and also discussed organization and the plant staff. We had interviews with various key people. We also looked at some important procedures.

Based on this audit there we had some concerns. We had a meeting then with mangement there. In summary our concerns are the following.

We found that there was a lack of proper staffing. At that time there were 160 people in the nuclear organization. At plant operation there were supposed to be 500. We also found that many of the key positions were still open.

Edison has since then made the following commitments that they will have an organization in place four to six months before fuel load and all the key positions would be filled six months prior to fuel load and also 90 percent of the staff will be filled four months before fuel loading. We find this acceptable.

We also looked at the training organization and found it at that time weak and fragmented. Training for licensed and unlicensed personnel we thought was behind schedule. We asked again for the training schedule and the status of the training.

We have since received detailed information on both the schedule and the training status. We have found that it meets the requirements in ANSI 3.178 which is actually more than we really require. We require that they meet ANSI 18.171 and Regulatory Guide 1.8.

MR. KERR: Excuse me, Mr. Pederson. Let me see if I understood you. I thought you said that when you visited you found the training program weak and fragmented. Then they submitted some paper and everything was okay.

MR. PEDERSON: It was not that simple.

MR. KERR: Well, it didn't seem to me that it would be that simple, but that is what I thought I heard.

MR. PEDERSON: They have detailed their training program and the schedule of their training quite well since then. We have in writing these documents and we have reviewed these documents since our May visit.

MR. KERR: What did you review when you visited the site that was so weak and fragmented and then became clearer after?

MR. PEDERSON: Well, again, at that time they gave an oral presentation but we didn't have any documents in our hands.

MR. KERR: Had they give you those documents it probably would have been okay?

MR. PEDERSON: That would have helped quite a bit, yes.

MR. KERR: Thank you. I understand.

MR. PEDERSON: Another concern was lack of commercial operating experience of the plant staff, and especially the operators. We do not require that all key personnel have extensive commercial operating experience. However, we certainly believe that such experience, especially during the first six months of operation, should be available.

As Dr. Jens referred to, Edison has now committed itself to have one additional person per operating shift with at least three months of BWR operating experience at the start of fuel loading and extending up to at least nine months. We have required that it should be up to 100 percent power. However, I am sure that we can talk about that and maybe 80 percent would be sufficient.

MR. MOELLER: You said six months just a moment ago. Do you presume they will reach 80 percent within six months?

MR. PEDERSON: We have required up to 100 percent full power.

MR. MOELLER: Well, you also though said for the first six months or so.

MR. PEDERSON: That is what Edison has proposed.

MR. MOELLER: Oh, Edison proposed that.

MR. PEDERSON: Another concern that we had was the number of license candidates in the pipeline. Edison has now commited itself to five shifts at fuel load. They have at present 33 candidates in training and they will have eight additional in training by September of this year, again their commitment, making a total of 41. An additional eight candidates will also be training by January 1982. This we find acceptable.

The last item we had concern about was again 1-C-5, the procedures for feedback of operating experience to plant staff.

We have reviewed the writeup which was a response again to the action plan item. We find that this writeup has made the right steps to meet the NRC position on this item. However, we also find that the procedure we have reviewed up to now, and we have in writing again from Edison, only concerned itself with LERs and there is nothing in the review of other operating information from publications such as I&E bulletins, circulars or notices and pertinent NRC information.

Also, the procedure does not indicate how operating experience is incorporated into training and retraining programs.

Now, having listend to Dr. Jens a few minutes ago, it looks like with his statements being put into writing that I am sure we can approve the new procedure.

Summary and conclusions.

Basic in our review of the information given in the FSAR and the various amendments and our trip to the corporation and plant, we can make the following conclusions.

The corporate and plant organization structure for operations and the total number of personnel planned to staff the organization are acceptable.

The phased transition of personnel from construction activities to operations will provide valuable continuity and carryover of knowledge to the operations phase. However, the applicant should effect this transition in a manner that does not compromise the quality of construction or safety of operations.

The qualification education experience requirements for the corporate technical staff supporting the nuclear plant operations and the resumes of the key personnel filling those positions have been found acceptable to us.

Also, the applicant has made acceptable provisions for manning the plant operating shifts again by adding this extra person both to the shift staff and to the plant staff.

The plant management staff will be provided with individuals having sufficient operating experience to safely operate the plant.

That is all I have.

MR. KERR: Thank you, Mr. Pederson.

Are there questions?

(No response.)

MR. KERR: I declare a ten-minute break.

(Whereupon, a short recess was taken.)

MR. KERR: My agenda shows operator training to be discussed by Kanous.

MR. KANOUS: I would like to express the appreciation of myself and the entire audience to Dr. Kerr for the humanitarian break that you just gave us. I for one needed that.

(Laughter.)

I am Lords E. Al Kanous, the Director of Nuclear Training for the Detroit Edison Company and I have the privilege of discussing with you four topics, operator selection, use of simulators in training programs, training programs for serious accidents including those beyond design basis accidents andthe selection and aining of maintenance personnel.

We have as a matter of policy chosen to select a mix of operating personnel to staff the plant, particularly those for the license group. The mix includes experienced fossil plant operators with experience in large-scale centralized control unit system operations.

Fortunately, some of the people who are in the group have had that kind of experience also going back into the nuclear Fermi-1 experience.

These people to enter the job have to be either an active supervising operator having qualified for that position or have to be a candidate for that position. One of the requirements that the Edison Company has in its fossile stations for that position is to have a stationary engineering license for the City of Detroit at the second class level which implies or connotes that they have already completed a rather extensive and demanding training program in stationary engineering. The City of Detroit has had for years the model for that kind of licensure in the country.

The other part of the mix is ex-Navy nuclear personnel whom we have a policy of selecting at the essentially eight-year level from the nuclear Navy. All of these people have been so far qualified as engineering watch supervisors and have by that process demonstrated their trainability as well as their superior competence. Those men have all had experience operating reactors although not large commercial reactors.

The experience that we have had with these people up to now in-house is a very systematic and orderly evaluation of past training experience and accomplishments both company required and voluntary and an evaluation of the reactions of their supervisors that they have worked for in connection with operating in a control room, their emotional stability, their ability to work under stress, their knowledge of systems and their ability to communicate in an operating situation. All of these kinds of things are evaluated according to a process which has been developed over a period of about six or seven years.

The experience with all of that in terms of the progress that these men have made through training at this point suggests that we have, and I don't want to toot our horm, but I think it is an outstanding crop of men who are highly motivated in spite of the fact that the process that they have been going through is a very severe and demanding one and that it has extended over almost a barbarous length of time.

We are as a company involved in the POST study, the Power Operator Selection Test Study, which is without question the largest scale attempt in the history of industrial phychology to develop a valid bearer meeting the requirements of EEOC's method of doing a prescreening of operators using psychological tests of one sort or another and the evaluation of biographical data.

The results of tha study, which began in 1978, prior to the concerns in Three Mile Island and other things, are just about ready for publication at this time. Seventy utilities across the country have been involved in that and our company is one of the pioneers in trying to get that off the ground and get the funding from Edison Electric Institute which has funded it through the company. It is about ready now to be published and evaluated in terms of how each company will apply the results.

I am happy to tell you that the results of that research indicate that we have in hand a valid measure of success in operating plants not only now but are in operating training. There are some 250 people across the country in various kinds of nuclear utilities where the operators have in fact had operating experience and their performance has been evaluated. We propose to utilize that as an augmentation of our selection procedures when we go.

Turning now to the use of simulators in the training program, Dr. Jens mentioned that we have committed, a contract has been let with the Singer-Link Corporation to build us a plant specific state-of-the-art simulator for Enrico Fermi-2.

So I would like to talk about this use of simulators in two contexts, the first of what we are doing now and have been doing and then what we will be doing and what we have put on a request for a proposal to develop with respect to that simulator.

MR. CATTON: Before you get into the simulators could you kind of give me the flavor of your philosophy of the training of the operators. You can train them to just push buttons or you can train them to understand their system. You can do all sorts of things. I don't really know where you are coming from other than you have put a lot of time into it.

MR. KERR: Did you understand the question, Mr. Kanous?

MR. KANOUS: I believe so, Dr. Kerr.

The answer to the question is that to operate a power house you must necessarily be trained to understand the system and its parts, the interaction of its parts and all these kinds of things. You must in addition be trained to know how to operate the plant, how to preoperationally inspect a pump in the correct manner according to whatever the procedures are or whatever. So there really is a necessary mix.

It is not possible, in my judgment, to do an adequate job of training on either end of that spectrum all by itself. It must be both. Our philosophy is to teach the people how to perform their work and understand the system with they work in order to correctly perform their work.

MR. CATTON: Would one of your trained operators, for example, be able to calculate net bulk at suction head?

MR. KANOUS: Yes, sir.

MR. CATTON: Good.

MR. KANOUS: He would have to do that in order to get his stationary engineer's license because that is a requirement of that license in the fossile plants as well as in the nuclear.

MR. CATTON: And things like the mast valves on a system, he could do that?

MR. KANOUS: Yes, and he could understand the process well enough to utilize the process computer to act as an assistant in crunching the numbers.

MR. CATTON: That is even better.

MR. CARBON: Will you expect to have quite detailed procedures telling this operator how to respond in essentially every case that you can imagine?

MR. KANOUS: Well, again, a mix. The training has been focused in terms of sympton evaluation. Our operating personnel have over the years been inputting to the training that we not only have to be told in certain instances that you must do it by the numbers but you must understand how the numbers were arrived at and what produced that procedure, how did it get that way, why is it that way.

MR. CARBON: When the operator is at work in the control room and something comes up and there is a procedure that says do so and so, will he have any leeway except to follow that procedure in detail as a general philosophy?

MR. KANOUS: I guess my view of that is that is why we have the operator in that control room in the first place. When that circumstance occurs the first thing he has to do is evaluate for himself what is happening, what is going on here and then to determine the appropriateness of the procedure that he has to operate the plant. That is he is there, to make that kind of an evaluation.

MR. KERR: Mr. Carbon, do you want an answer to your question because I am not sure that the training division alone can give an answer to this question. If you really want to explore it I would think you would want to ask some of the other people as well.

MR. CARBON: You have a real good point. I guess part of what I am asking is does the training division expect to train the operator on the basis that he will have the leeway to make decisions or is he going to need to follow the procedures in detail? What are you training him for?

MR. KANOUS: We are training him to understand the system, to understand how the procedures were built, the design basis that produced that kind of a procedure and so on. I guess I would defer to the superintendent as to how much leeway he is going to give them with respect to when that operator is in responsible charge.

MR. KERR: Mr. Griffing, do you understand the question Mr. Carbon is raising.

MR. GRIFFING: Yes, I believe I do. We are expecting our operators to have respect for both aspects following the specific procedures that we will provide them, that we have provided them and that they will be trained with under Mr. Kanous, and also to have respect for judgment. The shift supervisor in particular is going to be expected to have judgment. We have made an effort in our training program to provide some course material in the thermohydraulics area and those types of studies that would help the shift supervisor and the operators make judgments.

MR. CARBON: Let me concentrate on the operator. Do you expect him to exercise judgment and deviate from a procedure, or do you expect him to follow a procedure?

MR. GRIFFING: I think at the operator level, the control board operator, he will for the most part be following his procedures. The procedures themselves, though, for example, our emergency operating procedures, are symptom oriented and they provide the control board operator with some flexibility and more than just a rote follow your number down a certain event. We do not expect to put the operator in that type of mode for an emergency. He will have a symptom approach and he will be putting the plant in a safe condition regardless of the origin of the difficulty.

MR. CARBON: Fine. Thank you.

MR. KERR: Please continue, Mr. Kanous.

MR. KANOUS: Turning back to the use of simulators we have the present program which is an approved program by the Operator Licensing Branch being conducted for us on the

Browns Ferry simulator by the General Physics Corporation for operator certification.

In the program for the operators there is an additional program that appears just prior to the taking the licensing examinations given by the Operator Licensing Branch which we have referred to as the simulator refresher program.

In the original plan prior to Three Mile Island that was to be one week and it was essentially a refresher to bring you back up to snuff with respect to operating the plant as simulated. That has been augmented by one week and that week deals specifically with issues that have resulted from the various studies reflecting operator behavior and plant behavior as a result of Three Mile Island. So the additional week is attached.

Also, the requirements, INPO guidance and others, with respect to the shift technical advisors which we are committed to has in it the requirement for simulator exercises and that also will be done for the STAs.

Finally, the original plan we had called for the use of simulators, at that time an appropriate simulator outside the company or in the requal program. It was at that time thought that the plan would emcompass about one week per year of simulator training. That has since been modified now.

The Fermi-2 plant specific simulator is contracted to meet all of the current guidance with respect to how well that conforms to the regulations. In the contract it is to meet as of the time of delivery all at that time existing guidance. So we are again working ahead a little bit trying to see what is going on. The Singer-Link, Guss Warner, the Technical Director for Development is a member of the Standards Committee on Simulators and participates actively in that organization. We have therefore a close perception of what is going on.

In future programs we will at that time of course, since we will be hopefully after fuel load, we will be doing operator certification, shift technical advisor training on the simulator and requal training for any and all personnel who have to have to be requalified periodically.

We propose to add, because of the way the simulator is being built, several additional programs that we have already felt are useful to train some people in the simuator who are non-operators, at least non-licensed operators.

The first group of these is the non-licensed operators that we have specified in our contract for the development of programs to use with non-licensed people, as a part of their training as non-licensed operators to get them into the control room for looking around and get them over into the simulator to have an appreciation of how the work they do out in the plant interfaces with that that is done in the control room. We are not going to teach them at that point to operate.

Similarly with the maintenance personnel we have had some experience sending maintenance supervisors to the Browns Ferry simulator to take an abbreviated operator training program and for those people that has been an eye-opener. They have developed a greater appreciation for how their job fits in and the fact that the maintenance people and the operating people are closely interdependent with respect to operation of the plant.

Similarly I&C personnel tasks, and we haven't identified these, but again we have had the engineers and the technical supervision in the instruments and the control group take simulator training and they have found it valuable with respect to their understanding of how the plant operates and how their jobs fit in and how they must do it to help the operator rather than what is done in some other places where they seem not to be helping in some way.

Finally, there are some management and supervisory tasks that can be supported by simulator training to have people gain a better understanding of exactly what is involved in being an operator.

I have completed the management training at Browns

Ferry myself and that also was an eye-opener for me. I have a greater appreciate now than I had in the past although it was very high before.

Lastly, the simulators are being specified to provide outputs from the simulator computer to some additional training spaces in the EOF we call it currently such that signals can come out of the simulator and go to classrooms where we can exercise in a proper environment things that relate to the emergency plans and drills and exercises to provide a proper interface of the simulator with the control room. We have specified that to be done.

We don't know what kind of control displays, et cetera, might be placed in those rooms, but we have specified that the cabling be available in the building so we don't compromise the building and have cables all over the place.

MR. MOELLER: There is a difference in your present program in the future in that you don't list operator refresher training. Where is that then covered in the future programs?

MR. KANOUS: In the requal area.

MR. MOELLER: I notice in the present program you have requalification.

MR. KANOUS: Yes.

MR. MOELLER: So you are just moving that over into that area?

MR. KANOUS: Yes. The original plan suggested that prior to taking the initial license, since there has been a long span of training, that we would have to take the people back to the simulator for a week and then allow them to write their license in a hot licensing mode. You have a different problem and you are now requalifying the people who had their previous license using the simulator or whatever mode is necessary and training your follow-on personnel. That is why that is shown that way.

I would like to change a little bit, if I would be permitted to do so, and turn now to talking about the selection and training of maintenance personnel.

The Detroit Edison Company has for years had a commitment to training corporately. The first training program for significant personnel systematic orderly discipline, the kind of things we currently expect of nuclear programs, was developed in 1928 for system operators. There have been programs systematic orderly documented for fossil plant operators going back into roughly 1937 and formalized in 1958.

About ten years ago the corporation on the basis of some studies done by behavioral scientists and others looking at job satisfaction and that kind of thing came up with the idea of job enlargement which I am sure you are familiar with.

One area in which that seemed appropriate to us was the maintenance area, especially power plant maintenance work. We looked around the world and we found that a different philosophy than the one-man/one-craft philosophy was being applied in other utilities, particularly the Electricite de France.

So we examined the possibility at that time of what benefits might be obtained for both the men and the company of changing our concept of maintenance from a one man, one craft, to what we now call general maintenance training. And we have developed a program to train people in fossil and other maintenance organizations to be general maintenance journeymen.

This produces an increased scope of the work, so you are not just an electrician or not just a structural steel worker, you are not just a mechanic. We have a primary skill and secondary skill system. It has resulted in increased job satisfaction and it has produced a higher level of work.

MR. KERR: Do the craft unions in Detroit know about this?

MR. KANOUS: Yes, sir. Our guys are represented by the Utilities Workers Union of America. And one of the reasons it took us a fair length of time to get it where it is had to do with work with those folks -- work with our folks.

So our union is with us in this thing, reluctantly in the beginning, enthusiastically now. It also produced more money for their men. The general maintenance journeyman is paid one job classification higher than a simple mechanic or electrician.

(Slide.)

If you would like additional information on how that progra" is constructed. And in fact that it was designed in its original build with the idea of the Fermi plant in mind, that we had had it based on task analysis, job-relevant skills, orderly programs, systematic documented evaluation, and I can provide you with information about that if you wish.

People come out of the program, for example, with the primary skill designator of electrician and secondary as a mechanic, primary as a mechanic and secondary as an electrician. What that essentially means is, in the primary skill he is able to not only perform the work but lead others in its performance and be involved with the safety factors. As a secondary skill, he performs all the skills but is not permitted to lead. All of them are, furthermore, trained to help in any job classification.

This program is based on task analysis. It is in modular format. It is a modified self-paced program. People progress at the rate which they can progress at and slow down when they have to. But it is not simply a laissez faire program. There are certain schedules that must be met and the attainment of those schedules produced pay increases, which has a very high motivational effect on the personnel.

The evaluations are performance based, i.e., if the problem is to be able to align a coupling, the performance is to align a coupling and demonstrate that you can, without assistance, align that coupling, or any other of a number of maintenance tasks. It is not simply writing it down on a piece of paper, although there are cognitive tests as well.

Now, we get to the Enrico Fermi maintenance selection process, and at the top of my slide it says that at the Enrico Fermi plant all workers will be journeymen. There will be by policy no apprentices trained in the Fermi plant. What we will do is to select, using a constrained bidding system.

In the Detroit Edison we have had since 1948 a job fit system. We advertise openings, people bid, and then we select from among the bidders. And our contract allows us the privilege, rights, however you like -- I am not a union relations man -- to select on the basis of seniority and qualifications. The qualifications are primary. If two men have equal qualifications, the senior man gets the job, not based on seniority solely.

The GMJ mechanic must want to work at Fermi. He bids on the job. He is oriented toward coming to the Enrico Fermi plant. And then finally, because we have the right to evaluate their performance, we select among the high performers.

After they get -- being a general maintenance journeyman in the rest of our company is not sufficient, is not regarded as sufficient to perform duties in the nuclear plant. So that they get special specific Fermi training. One kind of training that they get I mentioned earlier with respect to use of the simulator. We try to provide training on the big picture: What is a nuclear plant, how does it work, what are the systems that are in it?

They get a version of what we call the generic systems and procedures program. We also have a program which teaches them new applications of old skills. A journeyman has precision instrument training, for example. The use of precision instruments are in the nuclear plant applied in special ways. Sometimes they have to be done in glove boxes. They have that sort of thing, so they have specific training applied for that.

There are some tools and equipment that exist in the Fermi plant and will exist in the Fermi plant which exist nowhere else, and they must qualify on those kinds of equipment. And then there are new skills due to the nuclear environment, the nuclear environment meaning the physical environment and the mental environment, if you will, of working in a nuclear plant.

And then finally, the development of proper working habits. One of the things that is necessary in a nuclear plant is a step increase, if you will, of work habits, and an understanding of the consequences of the kind of work that maintenance people do. That is also provided. That is part of when we select the workers. One of the things we evaluate in terms of their accomplishment in the past is how do they work, what work habits do they have.

I have a slide that I was told last night is pretty busy, and it sure is.

(Slide.)

You have a copy of it. It is called attachment one. This gives you a general overall look at the training program as it exists currently for maintenance personnel. I am not -- if you look at it across the top, we are not simply talking about our hands-on maintenance guys, but we are also talking about welding engineers, shop foremen, plant maintenance foremen, system foremen, engineering technicians, engineers, maintenance engineers, right across the board.

If you look down on the right you see the kinds of training that I have been speaking of up to now: emergency training, security plan, fire protection plan. All personnel trained in first aid, cardiopulmonary resuscitation, general training in plant administrative procedures, and specific training on maintenance procedures, and just run down the line.

As you get to the bottom you begin to see very specific kinds of things to the nuclear plant, such things as control rod drive hydraulic system maintenance, the Dexter valve reseater. Dexter valves are applied all over the plant and there is a specific device used only in the nuclear plant to reseat those valves. They are trained to operate that device efficiently.

Very specific training and qualifications to perform if they ever get out on the job.

MR. MOELLER: Now, each of these skills, or some of them, I presume, are covered in a single course. Others are covered in several courses or several are covered in one course, is what I am trying to say. And your first line there says "general." I presume it meant to say "employee."

You are simply saying all of these people are general employees?

MR. KANOUS: There is a specific training process that is referred to as GET in the industry, general employee training. That is what that is.

MR. MOELLER: That is what that is, okay.

MR. KANOUS: They get that. But in addition, as you look down the -- so that they cover at the first level, first cut level, so that they are able to come into the plant the first time they have to have that. They have to have a knowledge of radiation protection, health physics, that sort of thing, emergency plan, what sirens and signals mean, et cetera.

That gets them into the plant the first time, and beyond that we have to augment that. In the health physics area you have a training program that relates to something called the radiation worker. Okay, that is an augmentation. We are trying to build the programs, we are building the programs on a modular basis.

The general employee training program would work for a journeyman coming into the place for a first time. They all have common needs there. Then he has specific needs going off that way and I have some other different ones going off in another direction.

We have been asked to address the issue of serious accidents beyond design basis accidents or mitigating core damage, and it was earlier indicated that the staff had some questions about the content of the course, the outline of this course, and that will be provided to them by the date that was mentioned by Mr. Colbert.

But to give you some idea of the kinds of topics that are covered -- and I would like to say that it is not covered in a specific -- there is not a course, bounded course that talks about that problem. This business is now interspersed throughout many different elements of the training program, emphasizing those tasks that may in fact have an impact if we have an accident.

(Slide.)

And generally, we think that these -- this program, the general characteristics of the program, meet all the characteristics of the NUREG's. The letter talks about INPO guidelines. It will in fact meet all of those types of things. We recognize this is a very important area for training, particularly of operating personnel.

The topics that are covered are core cooling mechanics, potentially damaging operating conditions, the consequences of gas-steam binding on core cooling, the recognition of core damage --

MR. CARBON: How are you going to recognize core damage?

MR. KANOUS: Sir?

MR. CARBON: How do you recognize core damage?

MR. KANOUS: Using the kind of installed instrumentation that is in the plant, which these gentlemen -- radiation indications, chemistry, a number of ways, all of which I am not intimately acquainted with. Certainly the developers of the training programs and the implementers of those programs will be.

Hydrogen hazards, monitoring critical parameters during accident conditions, in-core instrumentation, ex-core instrumentation, the process computer and how it can be used to access information. That is an augmentation, incidentally, that is being done since Three Mile Island. Formerly we talked about using the computer only to provide operating information. That computer also provides additional information that can be accessed. We are teaching them how to make that access and use the information.

High radiation sampling, radiation hazards and radiation monitoring responses, the criteria for operation under these conditions and cooling mode selection, infrequent abnormal emergency operating procedures, gets into this problem, the thermal dynamics and heat transfer. Our men all get some of that as station engineers.

We have augmented that in a course taught at the college level. Half the people have already completed it.

Recriticality potential, and finally their role in the emergency plan. We believe that the program listing those topical areas meets the requirement of the NRC, and I guess if I had this available I could have given to Eric.

MR. KERR: How about the requirements of Detroit Edison's requirements?

MR. KANOUS: I think at a minimum it probably does.

MR. KERR: Thank you. Are there other questions?

(No response.)

MR. CARBON: Does that conclude this presentation?

MR. KANOUS: That concludes my presentation, sir.

MR. CARBON: I would like to address a question to Dr. Jens or Mr. Griffing. The emphasis of discussion here on operator training seems to me to be certainly worthwhile. Other organizations are doing the same thing and I applaud it.

But it seems to me that the breakdown at TMI was more at the level of the designer, the engineer, the supervisor, the management people, the people who should have been analyzing what could go wrong. Now, I want to ask you, are you giving equal attention or more attention to trying to upgrade things at that level since TMI, as much or more attention as you are to the operators and the maintenance people and so on?

MR. KERR: Who wants to handle that question?

MR. KANOUS: Can I address part of that for a moment?

MR. CARBON: Sure.

MR. KANOUS: One of the things -- the Commission -- I have been given the task of the director of nuclear training. It essentially says that I am responsible first to define, however that is done, with the appropriate people, the qualifications that are required for anyone who operates, maintains the plant or supports its operation, And then where we are finding deficiencies in those qualifications to provide the mechanism by which that training requirement gets satisfied. That includes the kind of folks that we are talking about.

MR. CARBON: I am not questioning what you are doing and what you have presented here. But your discussion had pertained primarily to operator and maintenance people.

MR. KANOUS: Only, if I may, only because when I was given the agenda on the topics it said discuss that.

MR. CARBON: I am not questioning that. I am asking management --

MR. KANOUS: I hope this is not an argument here.

MR. CARBON: No. I am asking management if there is equal attention or more attention being given to what I personally believe was more the trouble at TMI, rather than the operator.

MR. JENS: First of all, Ed and all of his section heads have been put through certification training. The maintenance engineer, obviously the operations Enrico Fermi, the radchem engineer has had that. And that is a step in that direction.

Furthermore, the transfer of the people from the design down at the plant, I talked about, certainly another step in that direction. The STA program that we are involved in and supporting is a step in that direction. We had prior to what I think is still some confusion as to what the requirements would be about shift supervisors and their training, we had engaged the Memphis State people into upgrading the educational requirements of the shift supervisors. And as I indicated, this ultimately will lead to a movement of people from the operation up into management, because they will then have that academic background.

So I think we are giving that attention to it. But I think we are somewhat unique because of what I tried to convey, that we have a large number of people with the background that are moving into the operations area, and as Mr. Kanous said they are going to receive training in various areas as well.

MR. CARBON: I guess the thrust of my question, though, is still aimed more, not at the shift supervisor or the maintenance people or operators, but more the nuclear engineers, the people who should be analyzing accidents, the people who are responsible for seeing that you have instrumentation that gives unambiguous readings, the people who are responsible for the technical and engineering analysis to keep situations like TMI from happening in the first place, rather than --

MR. GRIFFING: I hoped I had addressed that by the way we have organized. At least it was my intention to do that. But we are putting a tremendous technical resource into support of the operation of the plant at very high levels. Maybe I do not reawlly understand what else you think we could be doing.

MR. CARBON: I am not sure. Maybe it is simply communication. Let me ask this. I suspect you are spending a tremendous amount of money for a simulator here, and I suspect it is probably worthwhile. But it is a big chunk. It is a big change.

Have you made or are you making similar changes at the engineering, the designer, the analyst level, in increasing the emphasis on analysis, on the nuclear engineer trying to analyze and prevent accidents, an equal emphasis to what you are doing for the operators and so on?

MR. GRIFFING: Dr. Carbon, I believe there are several things that we have done which demonstrate -- we have, I believe, demonstrated that involvement of the engineers at the design and system analysis level and the plant staff in our safety review task force that we undertake, which was a great learning experience from my viewpoint. And I believe I can speak for the system engineers. They also learn from that experience.

Our STA program is essentially a qualification which we will expect many of our engineers to acquire, not just a small group that are dedicated to backing up the shift supervisor within five minutes, but essentially a career development checkoff, a point that they will pass in perfecting their abilities as professional engineers as they move up in the chain of management.

MR. CARBON: Well, thank you.

MR. KERR: Other questions?

MR. ZUDANS: Just a simple one. Do you have any training program to engineering staff that you hire, for any of them? By training I mean training in nuclear operations, assigning them to a simulato course or something like that.

MR. KANOUS: Yes, we do.

MR. KERR: As a matter of fact, he mentioned that as one of the uses of the simulator.

MR. ZUDANS: Well --

MR. KANOUS: Yes, we do.

MR. ZUDANS: That would be the answer to your question, Max.

MR. CARBON: No, I do not think so. I do not think simulators will in any way take care of my concern. A simulator, it is a robot. It is something that we set up based on --

MR. ZUDANS: That is only one part.

MR. KERR: If you and Mr. Zudans want to discuss this, I want to get on with the meeting. If you want to ask something -- if you want to continue questioning the Applicant, I am in favor of that.

MR. CARBON: No, no. I will stop.

MR. CATTON: Do you encourage your engineers to go to short courses, like the one that was held at MIT or elsewhere?

MR. COLBERT: Bill Colbert, technical director.

MR. CATTON: I believe those are the kinds of courses you are referring to, Max.

MR. CARBON: I am not acquainted with them.

MR. COLBERT: I am having a little difficulty with Mr. Carbon's question.

MR. KERR: Please answer Mr. Catton's question, because --

MR. COLBERT: In the context of trying to answer the other one -- all right. Your question was, do we have -- do we go to various seminars and so forth. Yes, we do, and in fact that takes place. It takes place on the basis of primarily need. There is again -- I do not think this business of system engineers has come through. The system engineer is responsible -- a project engineer, essentially, in an area needs to know -- that is, in his area -- updating information.

We make an effort to get people to various conferences, seminars that are applicable to his part of the job.

MR. CATTON: Thank you.

MR. KERR: Are there other questions?

(No response.)

Thank you, Mr. Kanous.

Let me ask one question. How does the organization, once you have gotten an operator through the training program, how do you select those that you are going to permit to take the licensing examination and if the pass it will then ultimately become operators? What is your selection process, as contrasted with the licensing process?

MR. KANOUS: In the first place, to get into the organization at all --

MR. KERR: I recognize that.

MR. KANOUS: That is a hurdle based on -- our current philosophy is to have all of the people make an attempt to qualify at the highest level of licensing that they can attain. Our --

MR. KERR: I am not expressing my question well. Given that they may qualify at the highest level that they can attain, how do you decide then whether you want them operating your plant?

MR. KANOUS: On the basis of a demonstrated competence.

MR. KERR: That is a good general answer, but do you give examinations, oral, written?

MR. KANOUS: Yes, sir.

MR. KERR: Who makes the decision as to whether they will finally become operators?

MR. KANOUS: The person who makes the decision ultimately they will finally become operators is sitting back there. He is the superintendent, Dr. Jens. I do not. I conduct -- my people conduct all the training programs, give all the examinations along the road, various kinds of assessments that are required. And the collection of that data leads to my recommendation that the man is competent.

MR. KERR: Can you respond, perhaps, Mr. Griffing, to what process you use, if there is a process other than seeing whether the man passes the licensing exam? I mean, are your qualifications different than that, higher than that?

MR. GRIFFING: We have --

MR. KERR: I recognize that you can make a selection process before licensing. You can say, we are not going to recommend anybody for licensing unless we would be willing to have him. But at some point you make the selection. Where do you do it and how?

MR. GRIFFING: We have not completed that effort yet because we are still in the program process. But evaluation of the individual's performance under stressful conditions and in the simulator exercises is something that I am very concerned with monitoring and using as a guide to assigning those individuals in the plant to an operating shift.

An example of that is my concern for the emergency procedure training that we are engaged with right now and being evaluated by the NRC. I am going to the simulator to specifically monitor the performance of our operators as they walk through these procedures.

MR. KERR: I asked the question in the context of what is done in other areas, and for example, if I understand the process correctly, an airline does not determine whether its pilots shall fly on the basis of whether they have an appropriate license. There are requirements beyond that. Not just any licensed pilot who comes in off the street is permitted to fly a commercial airliner. So they have qualifications which at least in many cases they think are more stringent than those that are required, for example, by the FAA.

I am simply trying to get an idea of whether you intend to do the same sort of thing and how you plan to do it. Is it a personal selection made by you as plant superintendent? Is it a committee thing? Is it a collective or --

MR. GRIFFING: There is definitely a collective nature to it, to the recommendation for a person to take a license in the first place and continue in that mode. We have to monitor our personnel for their daily performance to see that it does not go -- degrade or change in some abnormal manner.

I expect to use techniques like evaluation boards, which is an experience that I used and saw used in the Navy very effectively. In short, we will be using some subjective evaluations of the abilities of our licensed operators to man their positions, and on a continuing basis.

MR. KERR: Thank you.

I guess we have no more questions of Mr. Kanous. Thank you.

MR. KANOUS: Thank you.

(Slide.)

MR. LUSIS: I am Elwood Lusis. My position is assistant technical director, Fermi 2 Power Plant. I am addressing the control room and the design reviews of the control room.

As Dr. Jens mentioned, the human engineering factors are now being used by Detroit Edison in the control rooms. For a number of fossil plants preceding Fermi 2, we have used the task force approach for designing the control rooms.

In the Power Conference 1968, a team of Detroit Edison engineers presented a paper in which they said, amongst other things, analysis of operating mishaps which had originally been classified as human error, revealed on closer study certain features of systems as designed which appeared to have played a significant role in contributing to the mishap.

To deal with this problem of complex plants being operated from a location remote of the equipment, the Detroit Edison Company assembled for Fermi 2 a task force that consisted of several operators, the representatives from engineering disciplines, a human factors consultant, who was our company psychologist and presently the training director, and system engineers as the individual systems were discussed and analyzed.

The systems engineers have been named several times. It is a somewhat unique thing for Detroit Edison Company in assigning people that have intimate knowledge on a system or number of systems, and they are responsible for the functional assurance that the system will perform throughout all the design, testing and initial operation functions.

The control room task force spent some 18 months. One of the first decisions that they came up with was a room layout, the control room layout, and it was based on the consideration that you want to assemble the information so that it would be most accessible to the operator.

(Slide.)

The operator's desk is in the middle of the room. The front horseshoe is wrapped round the operator. The emergency operation panels are on the left side of the operator. The boiler operation is in front of him. The turbine and the feedwater, auxiliary panels of various systems are around the back of the operator.

A full-scale mockup then was constructed of this entire panel and as individual systems were developed they were laid out on this mockup. When the control room task force agreed on how the system is laid out, then the operating members of the task force performed mock operation and they tried normal operation, startup and shutdown of the system, and emergency operations.

That was then the approved section that remained in that position until all of them were laid out. Before the sections went to the manufacturers, another review was done by the systems engineer task force that walked through the entire control room and reviewed the configuration and the relation of the systems as they were laid out.

Some of the basic principles that the task force came up with in laying out the panels were: extensive use of mimics that show in the control room layout -- we have a general layout picture of the control room .

(Slide.)

You cannot see too much on these for the mimics. I have a more detailed layout of one section that shows emergency core cooling.

(Slide.)

You can see that the components here are laid out to represent a rough diagram of the system and that helps the operator to locate the valves that he is interested in operating.

This shows also another principle that was engaged in and that is shape coding, using only pushbuttons for valve operations. Other motors are operated by CMC switches. So that the operator immediately can see that he is going to reach for a pushbutton when he operates a valve.

Another shape coding that you see -- I guess it will be on the other slide --

(Slide.)

The indicators are a different shape. This is a rather small slide of it. Your team that was visiting the plant saw this. The flow indicators are a different shape than level indicators, temperature indicators. Pressure gauges are a different shape. So that just by looking at the -- a glance at the shape will indicate to the operator that he is looking at a particular function, to help him a little bit in an emergency so he does not make mistakes.

The control room after it was built has been reviewed. There is one more slide I can show you on the use of mimics.

(Slide.)

We made a live mimic. This is actually a live mimic that indicates the isolation of the primary containment. These lines are going to light up when the valves are open and you have a flow through the lines. The valves that I am not showing on this mimic were additional isolation.

A color coding of various components has been consistently implemented in the control room , and the principle is that, the same as in most utilities, the red shows energy flow and green shows no flow of energy. That is across the board in all locations in the panel.

The reviews that were made have been threefold. Right shortly after Three Mile Island, when it was evident that the problem at least partially was in the control room , we decided to review our control room to see how our design would withstand an emergency like that. For the guidelines we used the EPRI document that spelled out the human engineering principles. Our review revealed no discrepancy in the application of the human engineering.

After that Detroit Edison Company participated in the BWR Owners Group to develop a control room review procedures and checklists, and the control room was subsequently reviewed by the Owners Group team. The approach used by the BWR Owners Group in reviewing control room s is that the team consists of people from other than the utility that is being reviewed, and the home utility is just supporting the action. The decisions and the critique is done by the other people.

In this review again there were no major problems found in the control room. They basically agreed with our previous review.

Subsequent to that an NRC team came out to the control room and reviewed it in considerable detail. No major problems were found by them either.

There are open items that have been left that Mr. Kintner mentioned earlier. Those open items are the temperature and humidity and noise levels of the air conditioning system. Because the air conditioning system at the time of the review was down, the NRC team could not review it. Subsequently that has been reviewed and witnessed by the resident inspector and the report is on the way to the staff.

Procedures for --

MR. KERR: How does one review the noise of an air conditioner? Stand and listen to it?

MR. LUSIS: No. The noise level was actually measured by instrumentation and records made of that.

MR. KERR: And there is a standard?

MR. LUSIS: Yes.

Procedures for modifications of panels. We actually did have the procedures, but we did not produce them for the NRC team. They are going to be submitted now to the team for their review.

The third item that we are supposed to take care of before the first of August is color coding standards, and again that was just a matter of getting our company standards together and packaging them for the NRC to review.

There are two other items: evacuation signal, which is going to be procured, installed and operational before the 15th of October this year; and process computer programs. Process computer programs are basically the GE standard, and as commonly practiced they will be available shortly before the fuel load, because we want to incorporate all the ongoing things that are modifying in the plant.

It has been gratifying to see that a panel -- the task force effort that went into developing the panel has given us panels that ten years later have withstood the critique of our peers on several levels.

MR. KERR: You say that is encouraging?

MR. LUSIS: It is gratifying for us to see.

MR. KERR: I think it is discouraging to see that no more progress has been made in ten years.

MR. LUSIS: We felt that we were ahead of our time at the time.

(Laughter.)

MR. KERR: Are there questions?

(No response.)

Thank you very much -- Mr. Moeller?

MR. MOELLER: In terms of your control room design, the NRC has a manual, I believe they call it, that they are developing on control room design. Have you compared your design to the recommendations in their manual?

MR. LUSIS: We have compared our design to the degree that that manual has been settled on. And really, the manual is not much different from what the first principles were that were initially developed by NSAC/EPRI, and that is the one that we in detail compared ourselves against.

MR. MOELLER: Thank you.

MR. LUSIS: The contour is indicated in this.

MR. KERR: That is enough.

MR. LUSIS: Okay.

MR. KERR: Mr. Moeller is satisfied.

MR. LUSIS: Fine.

MR. KERR: Other questions?

(No response.)

MR. KERR: Thank you very much, sir.

MR. LUSIS: The next topic on the agenda is habitability for serious accidents, that is habitability of the control room. And that will be addressed by our system engineer, Dick Beaudry.

(Slide.)

MR. BEAUDRY: My name is Dick Beaudry, systems engineer, Detroit Edison.

We feel our control room has been well designed to be habitable following a seriouse accident, even following a LOCA. We have built inlets. Each inlet is built on opposite sides of the reactor building. Each inlet has dual radiation monitors. The radiation monitors have automatic sequencing such that they will turn on only that inlet which has only the lowest level of radiation.

We have complete redundancy of all active components in the control train, filter train. We have both a makeup filter and a recirculation filter for the control of radioisotopes.

We have calculated the radiation doses to personnel in the control room --

MR. KERR: Excuse me, Mr. Beaudry. Your first statement said, I think, that you believe that your control room is capable of taking care of a design basis accident.

MR. BEAUDRY: Yes.

MR. KERR: The parenthesis below your agenda item says "beyond DBA."

MR. BEAUDRY: I wanted to cover the design basis accident quickly and then go on.

MR. KERR: Oh, okay. Thank you.

MR. BEAUDRY: I figured it was easier to go from A to B, Dr. Kerr.

MR. CATTON: It is hard to read.

MR. BEAUDRY: We have calculated the doses to the control room personnel, and also the staff has calculated the doses. Our values show up here in the FSAR on the left-hand side. The staff values are on the left-hand side.

(Slide.)

We have done it both the realistic analysis and what is the nature and concern for the design basis.

Am I in your way, Dr. Kerr?

MR. KERR: Well, I cannot see through you, but I do not know if you are in my way or not.

(Laughter.)

MR. BEAUDRY: The design basis accident numbers both from the staff's and our own viewpoint are under the NRC guidelines. The basic criteria we use for this is Reg Guide 1.3, releases from the core which are 100 percent of the noble gas and 50 percent of the iodines.

In addition to this, down on the bottom of the slide we have what we call the direct dose to the control room personnel and that comes from fission products scattered throughout the reactor building and coming through the walls of the control room and the ceiling and the floor of the control room. As you can see, this dose is very low.

Now, we also feel that if we had an accident that is in excess of the LOCA, design basis LOCA --

MR. KERR: Let me make sure I understand. The doses that you have calculated are calculated on the basis of Reg Guide 1.3, sources?

MR. BEAUDRY: That is correct.

MR. KERR: So already you are far beyond the DBA. The DBA is a LOCA, and you do not get anything like this on a LOCA, not if the ECCS works.

MR. BEAUDRY: That is right. But it is -- the requirement for designing the control room system is the releases in Reg Guide 1.3.

MR. KERR: No, I recognize that. But I was just trying to understand your statement, which refers to these as post-LOCA control room doses. It seems to me if what you had was a LOCA and the ECCS operated you would not get doses anywhere near this, if these are the doses you calculate with Reg Guide 1.3 source.

MR. BEAUDRY: The ones on the right were the Reg Guide 1.3 releases.

MR. KERR: So you would not get anything like that if you just had a LOCA and the ECCS worked.

MR. BEAUDRY: Right.

MR. MOELLER: This assumes, though, cor inment integrity is maintained, the leak rate is very ... .005.

MR. KERR: Yes. It assumes that one doe this calculation -- when one does the DBA calculation --

MR. BEAUDRY: The assumptions here are the same as we would assume when we do our offsite dose analysis and so on for chapter 15.

MR. KERR: Yes.

MR. BEAUDRY: We feel that even if you went beyond this basis and looked at even a worse accident that the control room doses would still not be very much on.

MR. KERR: What do you mean by you feel? Have you done some calculations or --

MR. BEAUDRY: I will get into that.

MR. KERR: Okay.

(Slide.)

MR. BEAUDRY: For example, the whole body dose that we just covered in the previous chart, that whole body dose is due almost entirely to the noble gases released from the core. Since we have already assumed 100 percent is released, we hardly believe this can increase any more.

MR. KERR: Suppose one had a breach of containment and the noble gases got outside of containment?

MR. BEAUDRY: That is obviously another ballgame.

MR. KERR: So when you were believing you did not -- you did not consider that believable?

MR. BEAUDRY: Well, the ground rule of my analysis is that everything is working that was working before. The only difference is a release from the core. That is the question.

MR. KERR: Okay.

MR. BEAUDRY: All right. Thyroid dose --

MR. MOELLER: Is that the standard approach that the staff currently requests if they want something beyond the DBA? Does the staff assume no containment failure beyond or no leak rate beyond that --

MR. KERR: I assume the question is addressed to

Mr. Kintner. Do you understand the question, Mr. Kintner?

MR. KINTNER: Yes, the question is, does the staff require anything beyond the DBA by way of analysis. I would like to ask Frank Extulewicz from the Accident Evaluation Branch to comment on that.

MR. EXTULEWICZ: The answer --

MR. KERR: Would you identify yourself, pleae.

MR. EXTULEWICZ: I am Frank Extulewicz, Accident Identification Branch.

The current staff analysis as given in Standard Review Section 6.4 requires only the calculation of the Reg Guide 1.3 type of LOCA evaluation for control room habitability, nothing beyond that.

MR. MOELLER: When in the environmental impact guide, where they are doing plant design accident calculations, that does not include anything relative to doses in the control room?

MR. EXTULEWICZ: I personally do not know of a control room evaluation done for the class 9 accident in the environmental report.

MR. KERR: Thank you, sir.

Please continue, Mr. Beaudry.

MR. BEAUDRY: The dose to the thyroid is due entirely to iodines. Since we have already assumed 50 percent of those are released from the core, we would contemplate that that number would only increase by a factor of two at the most, and even if that were to be the case the resultant number would still be under the NRC guidelines or just at that level.

The direct dose to the concrete if that were as a matter of fact our control room is based upon a release from the core that is greater than the Reg Guide 1.3 requirements, namely we assumed that an additional one percent of the solids were released from the core when we did our basic control room design. And even with that additional release, we calculated that the control room personnel dose, once again through concrete, would only rise to about .6 of a rem, which is still quite low compared to the guideline values.

That is about it.

MR. KERR: Thank you. Further questions, Mr. Moeller?

MR. MOELLER: What is the maximum percent of makeup air from outdoors that you can bring into your control room?

MR. BEAUDRY: We bring in during this sort of a situation 1800 cfm through the inlet makeup filters.

MR. KERR: I think the question is not restricted just to an accident situation, but what is the maximum you can get.

MR. MOELLER: Presumably, in the accident you might not bring in any. I mean, you know, just go on recirculation, just enough makeup to keep the pressure at atmospheric. But what is the maximum percent of the total air flow that you --

MR. KERR: Mr. Lusis indicates he may want to respond to this.

MR. LUSIS: Ed Lusis, 100 percent makeup. You can have a 100 percent makeup.

MR. KERR: So you could have the total ventilation of the control room be outdoor air coming in?

MR. LUSIS: Yes. That is not the normal, but you can have such a situation.

MR. MOELLER: Thank you.

MR. KERR: Thank you, Mr. Lusis.

Other questions of Mr. Beaudry?

(No response.)

Thank you, sir.

MR. BEAUDRY: Okay.

MR. KERR: That brings us to instrumentation to follow the course of a serious accident, hopefully.

You do not look like Mr. Wooden to me.

VOICE: I would like to introduce this next subject.

Detroit Edison Company has been a member of the

BWR Owners Group and the licensing review group, which is a subset of the BWR Owners Group. Those organizations have gone on the record as questioning the applicability of thermocouples in a BWR core.

We understand that yesterday this subject was discussed at the Susquehanna meeting and we are prepared today with a representative from GE to present their position on the need for or lack of need for BWR incore thermocouples. And with that, I would like to introduce Rick Hill of General Electric, who will carry the presentation, concentrating on the incore thermocouples.

MR. KERR: Do you endorse his position 100 percent?

VOICE: Your question, sir?

MR. KERR: Do you endorse his position, the position we are going to hear, 100 percent? Or do you want to wait until you have heard it until you decide?

(Laughter.)

VOICE: I have heard -- Detroit Edison Company generally endorses this position. The term "100 percent," yes, I guess the answer is yes.

MR. KERR: Thank you.

MR. HILL: Thank you. Again, my name is Richard Hill with General Electric Company. I am here to discuss for you our evaluation of BWR incore thermocouples.

First I would like to present an overview of what I am going to talk about. I would like to be able to define the issue as we see it for you more clearly and discuss the conditions under which the thermocouples are postulated to be useful, and discuss evaluations of the BWR design, operation, risk and impact, with a short summary.

The issue as we see it stems from the purposes that are suggested for the thermocouples. Those purposes, according to Regulatory Guide 1.97, are to monitor core cooling and to provide diverse indication of water level. Our concern and the utility's concern in this matter is that the thermocouples will not actually provide a solution to those purposes or fulfil those purposes, with the exception of one narrow condition and that narrow condition is during some slow boildown condition in the core with no injection. When the core begins to uncover, at that point the thermocouples would be able to register the fact that there is superheat being produced in the core.

Outside of that one narrow condition, we do not believe that they will work and satisfy those purposes.

MR. CATTON: Do you believe the thermocouples may be useful in serious accidents beyond the DBA?

MR. HILL: Outside of that narrow condition -- and that narrow condition is outside of the DBA. That condition is no ECCS, in which you are faced with some sort of a transient or accident where you are losing your core inventory. And we will discuss that in a viewgraph a little more thoroughly next. But that is outside the range.

MR. KERR: Excuse me, Mr. Hill. You mentioned Reg Guide 1.97. I take it you have also now seen the supplement to the LaSalle SER, which I read these calculations show that the core exit thermocouples will function independent of location in the incore instrument assemblies and can provide operationally meaningful data with respect to inadequate core cooling.

MR. HILL: I would disagree that they would provide operationally meaningful data. I have reviewed the LaSalle SER supplement, the calculations there, and I can say that GE in general agrees that the thermocouples will register the superheat and that is what those calculations were performed to measure, that the thermocouples would measure superheat at present.

Our disagreement is the fact that those are operationally useful, that is operationally useful information. Again, it is only for that one narrow condition when you are in the process of having no injection, because a BWR with one ECCS pump running you prevent core uncovery, and so you would not be in that condition unless you had lost all ECCS.

MR. KERR: Mr. Zudans?

MR. ZUDANS: A question. Are you able to tell us where are the thermocouples in the primary coolant system at present, if any?

MR. HILL: There are no thermocouples in the BWR.

MR. ZUDANS: You do not measure primary coolant temperature.

MR. KERR: He did not say in the core. He said in the primary system.

MR. ZUDANS: In the primary system.

MR. HILL: There is only one place that I am aware of where there are thermocouples and that is in the CRD housings. We also have them, I believe -- that is to monitor the temperature of the CRD's. And we have them in the recirculation pump motors to monitor temperature there. Those are the only two places that I am aware of.

MR. KERR: Someone yesterday said that they were in a feedwater inlet line. Maybe that is counted as part of the primary system. Are they in the feedwater inlet line?

MR. HILL: I am not aware of any.

MR. ZUDANS: Are you trying to tell me that you do not measure the steam temperature that goes to the turbine?

MR. HILL: I am not trying to tell you that. There may be thermocouples in the balance of the plant somewhere downstream.

MR. ZUDANS: It is still a primary system. It is the same coolant that goes to the reactor.

MR. HILL: I would defer that to Detroit Edison if they have someone that is familiar with the balance of plant.

MR. KERR: Did you understand Mr. Zudans' question? It was whether you measure temperature going to the turbine, I believe. Was that the question?

MR. ZUDANS: Yes, that is right.

MR. KERR: Mr. Lusis?

MR. LUSIS: I would like our instrument systems engineer, Larry Wooden, to answer that.

MR. WOODEN: Larry Wooden, IoperatiC systems engineer for Detroit Edison.

Yes, we do monitor both the main steam temperature and the reheat temperatures on this turbine cycle, and I believe they are data logged and they are computed in the plant performance programs. There is another additional temperature measurement associated with the steam flow measurements, actually for correction factors and the method that the actual flow is measured.

Is that an adequate answer?

MR. KERR: Do you want to know if they are measured with thermocouples or did you just want to know if it was measured?

MR. ZUDANS: Just measured is okay.

MR. WOODEN: They are thermocouples. They are thermocouple measurements.

MR. KERR: Thank you.

MR. ZUDANS: This is all the measurement of primary coolant temperature at any location that you know of?

MR. WOODEN: No. We measure with RTD, we measure the recirculation flow temperature at the discharge of the recirculation pumps, which is actually more indicative of the primary coolant temperature during normal operation.

MR. ZUDANS: Recirculation discharge point is also measured?

(No response.)

MR. KERR: Any other questions on that, Mr. Zudans?

MR. CATTON: Where are the RTD's located that measure the steam temperature?

MR. WOODEN: There are thermocouples -- there is a set near the actual flow measurement in the steam line. We have a flow measurement.

MR. CATTON: Where are they located?

MR. WOODEN: Downstream of the excess flow, near the main steam isolation valves in our scope of supply --

MR. CATTON: Thank you.

MR. KERR: Thank you, Mr. Wooden.

Continue, please, Mr. Hill.

MR. HILL: All right. Additionally, our concern is that for this one narrow condition it does not appear to be cost effective and will cause high radiation doses, especially in an operating plant.

MR. KERR: What is the cost anticipated to be?

MR. HILL: Can I save that question until I get to my impact slide?

MR. KERR: Yes, sir.

MR. HILL: Thank you.

The solution that we view for this is that the present BWR level monitoring instrumentation is redundant. We believe that it is sufficient and adequate to monitor level above the core as well as in the core in the fuel zone. Other instrumentation is provided that can be used as a diverse indication that water level has gone below the core and that you are in a condition of uncovering the core. And we will talk about that instrumentation.

(Slide.)

This viewgraph will require just a little bit of explanation on some of the terms. First, "systems unavailable." Again, we are talking about the condition where you have a slow boildown or some sort of loss of inventory and the core is being uncovered. The systems unavailable here are in a sense increasing in order of degradation.

The term "useful" here, it means useful for operational purposes. And the term "work" here is characterized as work to meet the purposes that the NRC has suggested that the thermocouples should meet, and that is diverse water level indication.

And as you can see, until you get down to the point where all ECCS -- and let me add what is not on here, that is all feedwater, condensate or any other mechanism for adding water to the core -- until all of those really have been exhausted and are not available, only at that point would we claim that the thermocouple then would register the superheat, because at that point you would be uncovering the core and it would give you some indication. And that case would be the same regardless of whether or not you have water level monitoring, and that is what I will discuss in the evaluation, the operations evaluation later.

The guidelines that BWR's have allow the operator to take actions to put the plant in a safe condition even without a water level monitoring at all.

(Slide.)

From a design evaluation standpoint, BWR's have multiple sources of water, and not to confuse you, but comparing the double counting systems: two types of systems are core spray type systems and injection or flooding type systems. Within those categories of systems, we have high pressure type systems and low pressure systems which can be activated through the use of ADS under a high pressure condition.

We have much flexibility in this type of an arrangement. Any core spray at all will prevent the thermocouples from working, as we characterize the word "work" here, and therefore it would nullify any superheat in the area of where the thermocouples would sense and the thermocouple would not register a temperature increase, but would continue to register saturation temperature.

The BWR measures level directly in the reactor vessel, and this is key in understanding the redundancy in the level instruments and the reliability we believe that they have.

MR. CATTON: The level is actually measured in the shroud, isn't it, outside the shroud, not in the vessel itself?

MR. HILL: The shroud is inside the vessel, you are correct. It is inside the vessel. It is outside the core, inside the shroud but inside the vessel.

MR. CATTON: So you have to assume that you will never get into steam binding problems to have it always be a meaningful measurment.

MR. HILL: That is true. I do not believe we have to assume. We have analyzed exhaustively in response to the Bulletins and Orders Task Force questions that were posed after the Three Mile Island accident. I believe the staff has agreed with that, and we believe and they have concurred that our water level system is a highly reliable type of system. And there does not appear to be any of these mechanistic means of a postulated scenario.

MR. CATTON: Under most circumstances.

MR. HILL: Under any postulated circumstances we have been able to look at.

And in addition, radiation and hydrogen monitors, along with sampling, are a diverse means for that one narrow case because you have to realize you are looking at a case where the water level is down below the bottom of the core, is continuing to decrease, and under those conditions you are going to be generating hydrogen and hydrogen presence will be a positive indicator that your core is in that condition, as the sensing of superheat.

MR. KERR: To what radiation monitors does that last sentence refer?

MR. HILL: That would apply to any general radiation monitoring within the containment, as well as hydrogen monitoring within the containment.

MR. KERR: You have the capability, don't you, to monitor each of the individual incore neutron detectors?

MR. HILL: Yes.

MR. KERR: Is there any way that those detector readings can demonstrate whether or not they are surrounded by water?

MR. HILL: That subject has been reviewed.

MR. KERR: I would have thought so.

MR. HILL: And the answer that we have come up with is, nc in an unambiguous fashion. You can postulate certain conditions where they possibly might. I am not here prepared to discuss in depth the neutron monitoring, but that is a mechanism which has been put aside as not being an unambiguous monitor, if water level happens to be where the neutron monitor is.

(Slide.)

From an operations standpoint, the operator is going to restore water level long before it reaches the top of the core and you start to have fuel uncovery. In addition, the symptomatic guidelines that are provided for boiling water reactors will allow that operator to take actions in a safe direction whether he has ECCS systems or not, and that includes whether he has any injection capability, including feedwater or condensate or not. They specify actions for him to take in a safe direction whether he has water level monitoring or not, and with a combination of those losses the guidelines still provide him with guidelines to take action in a safe direction.

The thermocouple indications will not provide him with any new information that he does not already have in his ability to follow those symptomatic guidelines.

MR. CATTON: What about things like the extent of core damage?

MR. HILL: We do not believe that thermocouples are going to be able to adequately monitor or even come close to monitoring any extent of the core damage. Once you have recovered water level in the core, the core is going to be sitting there at the saturation temperature.

MR. CATTON: But wouldn't the peak temperature that you measure during this process tell you something about the extent of core damage?

MR. HILL: I do not believe so, because what you are measuring is superheat, and if superheat is not going to be -- if you look at even the analysis the NRC did, the rough analysis, that is not going to give you a good indication of exactly what temperature your core might have gone to.

MR. KERR: Mr. Hill, I do not see why you tell me you are measuring superheat, because it seems to me what you are measuring is the temperature of the thermocouple and that temperature might be associated with the superheated steam passing the thermocouple, but it seems to me there could be a situation in which there was no steam at all passing and it could therefore reflect something about the absorption of thermal radiation that the thermocouple would see.

So I do not see how one can say unequivocally it is just measuring superheat.

MR. HILL: I guess I would tend to agree with you. It would measure some radiation. But as long as there is water in the core there is going to be steam through the core.

MR. KERR: That is quite true, but suppose one has a situation where there is not any water in the core, which I think might be included in the question that Mr. Catton was asking.

MR. CATTON: When there is water in the core I do not think we are concerned. I get the feeling that you keep directing this towards the case where the water is in the core.

MR. HILL: Where the water is below the top of the core and going down.

MR. CATTON: As long as the core is wet, I do not think any of us have concern. It is when the core is dry that we have concern.

MR. HILL: If the core is completely dry.

MR. CATTON: Not even completely dry.

MR. HILL: Well --

MR. KERR: Maybe 98 percent dry.

MR. HILL: We have done some studies that show that as long as there is about one food of water left in the bottom of the core, you are going to have enough steam, you know, there to provide what we would consider --

MR. KERR: What about six inches?

MR. HILL: You might have a tough time.

MR. KERR: Okay. So it might --

MR. HILL: There is obviously some point where you are going to say the core is dry.

MR. CATTON: In order to make those studies you have to make certain judgments, things like countercurrent flow limiting, steam binding, all sorts of things that we really do not know all that well. So we are operating in kind of a fictitious world wher you talk about your studies.

MR. HILL: I would counter that by saying that for the types of studies, the type of events that you are looking at the thermocouples being of any use for -- and that is a slow transient or a loss of inventory -- you are not going to be experiencing these phenomena. That is, the design basis type of event or a phenomenon that would probably be a design basis event.

After a design basis event, it is not going to be a surprise that the core is going to be voide because that is what the analysis does show. So I do not believe that we are involved in some phenomenon here that is difficult to analyze.

Okay. In continuing, along with the operations evaluation, we believe that false indication from the thermocouples, however, could mislead the operator into taking some inappropriate action. In the fact that you put instrumentation into the control room, you are asking the operator to look at that instrumentation.

MR. CATTON: Could you give an example of this inappropriate action he might take as a result of a thermocouple?

MR. HILL: I can give you an example of what we are going to do in the way of analysis to try to determine --

MR. CATTON: I care about the inappropriate action that he might take.

MR. HILL: You have asked a question that fits in with my next viewgraph.

MR. CATTON: Good.

MR. HILL: Because what we are attempting to do in a risk evaluation is to look at both the positives and the negatives that could result from having thermocouples installed.

(Slide.)

The negatives obviously are what inapropriate action could the operator take, and we do not have the answer right now to that because the study is in progress. But the study is assuming that the thermocouples are there. We are going back into a previous probabilistic assessment that General Electric has done and look at the various fault trees and event trees and look at how the operator has been modeled in each area.

MR. CATTON: I am really flabbergasted by that. Could you make a wild guess at what a --

MR. KERR: I can tell you. I mean, he might see that the core is melting, he thinks, and run like hell.

(Laughter.)

MR. CATTON: Oh, okay.

MR. HILL: Let me see if I can characterize something even better. if you are looking at a near-coremelt scenario, some near-coremelt sequence, the core does not really melt, but you have come close to it, if the operator sees that the failed thermocouple is registering he might make some assumption based upon that that could lead to a worse case, by turning one system off or doing something to another system, focusing his attention on a system that has failed because he feels he is in trouble, instead of monitoring the system that is working fine -- it is just going to take a while to bring the core back up -- therefore leaving that flow indication of one system alone and then eventually letting the core go down.

So you know, I do not have in my hip pocket the five things that the operator could do wrong if he had thermocouple instrumentation that had failed.

MR. ZUDANS: You are not talking about a single thermocouple here. There are 16 of them. That scenario described does not sound realistic if you did your job right.

MR. HILL: In the same way, it does not sound realistic that the 18 or so transmitters for water level instrumentation --

MR. ZUDANS: Nobody is questioning the reliability of the feedwater level indication system. That is not the issue.

MR. KERR: I am sorry, Zenons. If one asks for a diverse system, surely it must be being asked for because the staff thinks the existing system is unreliable. That is the only possible reason to ask for it. So somebody is questioning in some sense the reliability of the existing system.

MR. ZUDANS: Now that you say that, then, I guess that is the motivation.

MR. KERR: I can think of no other reason to ask for it.

MR. ZUDANS: Maybe staff could tell us.

MR. CARBON: I was just going to comment, we are not supposed to argue among ourselves.

(Laughter.)

MR. KERR: That does not refer to the Chairman.

(Laughter.)

MR. CARBON: Excuse me. I stand corrected.

(Laughter.)

MR. KERR: Also, I was really instructing him, no arguing with him.

(Laughter.)

Please continue, Mr. Hill.

MR. HILL: Thank you.

Additionally, we are assuming the fact that the thermocouples will respond in this type of an event with a time lag in the way of minutes, and what we mean by a time lag there is a time difference between when the thermocouples would respond and when the operator would see hydrogen or radiation or some other indication that he has core problems.

We intend to assess improvements in operator actions, as I have suggested already. This could be in the way of operator miscallibration of instrumentation either in water level or in any of the responding ECCS systems, any unavailability of a particular system due to misalignment or something. We are going to stretch our imagination, if you will, since we are not dealing with real operations but we are dealing with a risk study and saying that if the operator sees that these thermocouples are registering he will immediately remember the fact that whatever the system unavaiability was caused by in the way of a misaction, maintenance or testing, he will recognize that and correct it.

In addition, we will try to assess the change in coremelt frequency as a result of digesting those fault trees and the approximate effect that that would have on early fatality risk. And you will see in the next viewgraph what we intend to do with that early fatality risk.

We have this projected for the end of August completion. As it says here, risk reduction that we expect would be very small. And any analysis of this sort of course could include a potential that would see the negative factors, show the risk might increase from the fact that the operator could be responding to false information.

(Slide.)

That is why we are doing the study, to take a look at both the positives and the negatives, Mr. Zudans, from the impact. And this goes back to your question, Dr. Kerr. The radiation to maintain and install -- and this number is an operating plant number -- maintenance man-rem doses on a per-year basis would be somewhere possibly around ten man-rem figure. This is 100 man-rem.

From a cost standpoint -- and this is a barebones minimum type of cost that we did rather quickly. It does not necessarily include qualification of equipment and it does not -- which could be substantial -- and it does not necessarily include some BOP areas like having to add additional penetrations.

MR. CARBON: I was not clear on the 100 manrem per plant. That is for a lifetime?

MR. HILL: That is an operating plant installation type of number.

MR. CARBON: To install it?

MR. HILL: Yes.

MR. CARBON: And then how much to maintain it?

MR. HILL: On a per-year basis, our judgment is somewhere around 10 manrem.

MR. CARBON: 10 more. Thank you.

MR. HILL: Yes, per plant.

Safety goal comparison. We took a look at the ACRS safety goal and the proposed cost-benefit criterion. That criterion is there to provide a goal or a judgment beyond which it would not be reasonable to continue to try to increase the reliability of the plant for an expenditure of the money. And if you convert the ACRS criterion with the BEIR-III linear dose coefficient you are going to end up with $100 per manrem, which is consistent with others that have been proposed, like the AIF safety goal.

And just doing a simple calculation on this, you realize that the plant is about $6,000 per manrem, rather than the $100. And in parentheses here I put "occupational," because obviously the safety goal is talking about a societal number and the number we have calculated here is --

MR. KERR: Mr. Hill, I sure wish you had not put this slide up because it really makes me wonder about the other calculations you have done. Surely the staff is not asking for thermocouples just to protect the workers of the plant.

MR. HILL: No.

MR. KERR: And that is what you are dealing with here. The benefit -- I think the principal benefit that one looks for is increased safety to the public.

MR. HILL: That is where we are headed.

MR. KERR: And that is what you have to --

MR. HILL: If you will give me a second. I put this number here to give you a callibrated idea, and I put it right here so that everybody could see it, not that I was trying to confuse it.

The results of our --

MR. KERR: No. But you see, they are both costs. The 600K plus the 100 manrem are additive costs, they are not divisible costs. The cost of installing this thing includes the 600K and the 100 manrem.

MR. HILL: Okay. Understand where I am headed from -- where I headed to --

MR. KERR: I just do not want you to go any further when you are going in such a direction.

MR. HILL: When we finish our risk evaluation, sir, we will be able to develop a delta society risk.

MR. KERR: I hope you don't do the arithmetic the way you have done it on this. It just is not right.

MR. HILL: Would you agree, then, that if we divide the cost over the delta of the societal risk we will have the appropriate number that I am trying to express here?

MR. KERR: Well, there are two costs here. One is the 600K, the other is the 100 manrem.

MR. HILL: Yes, sir.

MR. KERR: Now, I do not know what you are going to do with that or what you are trying to calculate. But if we agree that both of those are costs then we are together. But I do not think that slide shows any benefit.

MR. HILL: Do you understand what we are going to put in there once we have the risk study completed?

MR. KERR: I assume you are going to put in the benefit in terms of decreased risk.

MR. HILL: That is right. That is where we are headed. I am sorry if this slide has confused you on that matter. It was our intention --

MR. KERR: I do not think the slide confused me at all. I hope it did not confuse GE.

(Laughter.)

MR. ZUDANS: Isn't this slide completely wrong in terms of the BWR statement? What is the safety goal? The cost of dollars for reduction of certain exposures. And you added the exposure and you added the dollars and divided them with each other.

MR. KERR: That is precisely what I was trying to say and I must not have said it very well.

MR. ZUDANS: You said it well, because even I understood it.

(Laughter.)

MR. CATTON: In your risk study, are you going to somehow incorporate benefits of more information to the people who are responsible for the emergency plan? For example, the core is heating up, it may melt in an hour, or we may have vessel penetration in two hours. This kind of information I think would be very valuable in reducing the impact on the local people.

Are you going to build that in or are you going to --

MR. HILL: To the extent that that type of thing has already been modeled into a risk study -- and I have to admit that that has not extensively been modeled --

MR. CATTON: With temperature measurements to a certain extent you are going to have some of that information. Without them you are not. And if you cannot get that built in you are not going to make a true -- get a true measure of the value.

MR. HILL: However, I believe if you have the hydrogen detection you have the same information that you are studying to degrade your core, as far as an emergency.

MR. CATTON: Not necessarily.

MR. KERR: As a matter of fact, hydrogen generation tells you what has happened to the cladding, and that may not be all you need.

MR. CATTON: That is right.

MR. HILL: The last point I want to bring up in this impact slide is the increased potential for forced outages, and I do not believe that that is necessarily a small one. Until the system has been designed and looked at carefully, we cannot really describe the reliability of it.

MR. CATTON: Can't you make a guess based on the reliability of the excore thermocouples in PWR's? I would assume yours are going to be somewhat similar.

MR. HILL: The trouble with the PWR experience there is -- we do not have that data readily at hand and neither do any of the other industry organizations like NSAC or INPO. The excore thermocouples -- the core exit thermocouples in PWR's, as we understand it, are not safety grade.

MR. CATTON: They are pretty reliable.

MR. HILL: We do not have that information and have not been able to gain that information. We have been able to look at the thermocouples that we have on our own plants and other applications that I mentioned.

MR. CATTON: I suggest that you ought to take a look if you are going to try to come to a conclusion.

MR. HILL: When you take a look at that system from an analytical point of view, I would probably agree that the thermocouple is not going to be a highly unreliable element. Whatever switching network you have or anything else that is in the way of an active component, that will probably be the less reliable component and therefore would be controlling the reliability of this system.

MR. KERR: Please continue.

MR. HILL: By way of a summary, we do agree that thermocouples will for that one narrow condition where the water level is within the core region.

(Slide.)

The BWR, as I have said, has multiple sources for injection, flooding, spray. The BWR has redundant and sufficient water level monitoring instrumentation already that other instruments will provide a diversity for the purposes that the NRC has suggested and for which purpose they did their additional calculations in the LaSalle SER.

The thermocouples are not going to be beneficial for operations. The operator is going to act independently with his procedures regardless of any thermocouples that might be installed there, but he is going to be acting according to the water level instrumentation that he has. And the operator can act safely even if he does not have that instrumentation or has multiple degradations in his ECCS systems.

The doses and costs are significant in this case, and we are not convinced that there is going to be a reduction in risk. And we have to take a very close look at any negatives that might come out of that from a risk standpoint, as well as there is a potential for increased forced outages.

That concludes my presentation. Questions?

MR. KERR: Mr. Catton.

MR. CATTON: The incore radiation monitors, do you ever have to take them out to replace them? Is there a maintenance program associated with them

MR. HILL: Yes, there is.

MR. CATTON: Now what is the manrem cost in doing that?

MR. HILL: I do not know the manrem cost absolutely. If you are asking, is the delta between that and core thermocouples adequately measured by the --

MR. CATTON: I have a feeling that you are giving the full manrem dose to the installation and maintenance of the thermocouples, rather than the incore radiation monitors that are already in there.

MR. HILL: No, the numbers that we presented -- that I presented are not biased numbers to try to make the thermocouples look bad. They are numbers of what it would cost in addition to any regular maintenance or replacement that would have to take place in radiation monitors.

MR. KERR: Mr. Moeller.

MR. MOELLER: Is the topic, the Reg Guide 1.97 requirements, is that next on the agenda or has that already been covered.

MR. KERR: Mr. Colbert, did you want to comment on that question?

MR. COLBERT: Is I understand the question, is there something more to -- is there something more to Reg

Guide 1.97 we are prepared to present besides this. And the answer is yes.

MR. MOELLER: I will wait.

MR. KERR: Any other questions of Mr. Hill?

(No response.)

Mr. Hill, you talked with the staff at some length about this up to this point. Why do you think they want thermocouples?

MR. HILL: Up until including a couple of weeks ago when we met with Mr. Rubinstein, I had a difficult time trying to clear up why they wanted thermocouples. I have made the statement before, I will make it again, that if it is diversity they are looking for in a water level measurement, diversity in a water level measurement can be provided actually within as short a time lag, a few minutes, with hydrogen detection as it could be witg thermocouples.

The staff has said that their position is summarized in the LaSalle SER as best as they can do sc, and it is not clear to me, after having read through that SER, truly what the purpose is.

MR. KERR: Thank you, sir.

Mr. Carbon?

MR. CARBON: Let me ask the staff if Mr. Wang is expecting to discuss this same subject under the third item here?

MR. KINTNER: Mr. Larry Phillips will discuss the subject.

MR. KERR: In the presentation or discussion yesterday, the impression I got was that the staff's position was fully explained in the LaSalle SER, and that Mr. Phillips did not have anything to add to that. Am I incorrect or do you now have something to add?

MR. KINTNER: I will let Mr. Phillips respond to that.

MR. PHILLIPS: Larry Phillips of the staff.

There is nothing particular to add to that. I do have some comments which I would like to make, and I would also -- you indicated yesterday you would like to have some additional capability on answering certain questions here, and I would tell you what we are prepared to answer.

MR. KERR: Okay. It may be, then, that what we should do -- are you finished, Mr. Hill?

MR. HILL: Yes, I am.

MR. KERR: What I suggest at this point is that we have the presentation from the staff before we go to the rest of 1.97, if that is okay with you.

MR. COLBERT: Fine.

MR. KERR: Mr. Phillips, if you are going to make a presentation, rather than review the LaSalle SER, I would suggest that you respond to questions on that if there are questions and that you make whatever comments you want to make.

MR. PHILLIPS: First let me say that I am not going to belabor these slides, many of which the Committee has seen before.

(Slide.)

Thermocouples actually were first considered under Reg Guide 1.97, and we also considered them under II.F.2, their usefulness in connection with II.F.2 instrumentation for detection of inadequate core cooling. In that consideration, we did hear from General Electric two or three times, and before we reached any firm decisions, or at least to the degree that we are now, I heard essentially all of the information that was presented here today. And the numbers have now been refined in some cases and the information has been packaged a little better.

But all that information was available to us.

(Slide.)

In connection with II.F.2, the staff has recently taken a position on what is required for the various reactor types based on what has been proposed and what has been considered, and included in that position, as you can see on the lower right-hand side, is that for General Electric reactor types we do require core thermocouples.

(Slide.)

The basis for our requirements for core thermocouples for the BWR, as you can see on the lower right side, and as I said, was NUREG-0737, which was inadequate core cooling instrumentation. It is not specifically called for in there. There is a phrase that provides that thermocouples will be designed to the requirements of attachment 1, PWR thermocouples.

MR. KERR: Mr. Phillips, I think it would be more helpful to me and perhaps to the rest of the Committee if, rather than telling us the legal basis for the requirement, you sort of emphasize the technical bases.

MR. PHILLIPS: That is my next slide.

MR. KERR: Okay.

MR. PHILLIPS: And we spelled it out in the LaSalle SER, our conclusions. NUREG-0591 and Reg Guide 1.97 also specifies the requirements for incore thermocouples.

(Slide.)

The technical basis for the BWR thermocouples -- let me say first, as was indicated in the LaSalle SER, the staff did perform calculations and we agree that the thermocouples while core spray is coming in will monitor saturation temperature, that he usefulness of the thermocouples are primarily when there is no safety injection or very degraded safety injection and the core is becoming uncovered.

We would take that one step further on item 3 and say that even before that, if the core is partially uncovered, as indicated by your water level instrumentation, it is certainly useful and reassuring to know that the spray is getting to the core and that you are reading saturation temperature.

Secondly, it monitors core cooling effectiveness, that is when we -- when the level drops and we have superheated the cladding, our calculations show and General Electric calculations show that superheat will be detected. I think the primary disagreement is possibly in the lag in that detection, and that seems to be primarily on the basis of the assumptions that went into the analysis.

We do have Peter Anderson from our research staff, who worked on our calculations, who will discuss in any detail that you might want those calculations. He is prepared to answer questions on them.

However, I would say that even if we accepted the GE analysis I do not believe that it would change our conclusions.

MR. CARBON: I do not understand that. If their analysis is correct and it is not complete, but if it shows that these things are highly un-cost effective, wouldn't that change your decision?

MR. PHILLIPS: The cost effectiveness -- I am speaking of the analysis of the effectiveness of the thermocouple in terms of detecting superheat as the core becomes uncovered. There we are only speaking in terms of the difference in the lag of the detection of the thermocouple versus the fuel cladding thermocouple, and I believe that is primarily on the assumptions in the amount of steam cooling possibly that GE assumed in their calculation.

We considered the cost of the thermocouples and the dosage very early in the game. As I say, the numbers may have changed somewhat, but those did receive consideration.

MR. KERR: How high would the dose have been before you would have been concerned about it?

MR. PHILLIPS: I cannot answer that question.

MR. KERR: Well, I do not see -- you just said you considered the cost. If you did not have any goal, how could you have considered it? You must have considered it acceptable, so what would you have considered unacceptable?

MR. PHILLIPS: I believe Jack Rosenthal of the staff can possibly help on this.

MR. ROSENTHAL: Jack Rosenthal. I am in the instrumentation and control systems branch.

When discussing the doses for installation and maintenance, we recognized that one had to maintain CDM, control rod drive mechanisms, one had to maintain the normal instrumentation system, and the numbers that -- we felt that this was a small contribution to the total dose of people who have to go into that area anyway. And it was based on the perception that we were talking about a small fractional increase in the dose that one gets by virtue of the design of the boiler.

And I think that the -- that was the basis for the --

MR. KERR: You are talking about 10, not 100, aren't you? The 10 per year, which was the estimate of maintenance. What about the 100 for installation? That turns out not even to be a small fraction of the annual dose that is typical of boilers.

At what level would that have had to be before you would have considered it serious?

MR. ROSENTHAL: Okay. As I recall, one did not have firm numbers. One -- we were looking at incremental dose.

Second, talking to our radiation people who participated in this, it was some concern that the installation doses were realistic, especially for the newer plants. But we did not have numbers.

MR. KERR: Some concern? You mean you did not think they were realistic?

MR. ROSENTHAL: We thought that if one tried hard there are ways of installing temporary shields and devices to reduce the do -s.

MR. KERR: How much?

MR. ROSENTHAL: I do not remember the numbers.

MR. KERR: Okay.

MR. CARBON: Do you consider 100 manrems significant? I mean, from what you have said I cannot understand a bit of what your true thinking is.

MR. ROSENTHAL: The maintenance dose is a small fraction of the total dose that people get working on those sort of systems. That was one consideration.

MR. CARBON: That was not my question.

MR. ROSENTHAL: I do not have numerical values for you with me here today.

MR. CARBON: My question was do you consider 100 manrem significant?

MR. ROSENTHAL: Yes.

MR. CARBON: And you said earlier that you had a perception that the installation radiation dosage would not be very large, I believe, or something like that. What was the magnitude of that perception? One manrem, 100 manrem or what?

MR. ROSENTHAL: I would have to bring down the people in our radiation assessment branch who advised us. I cannot answer your question.

MR. KERR: Thank you.

Mr. Zudans?

MR. ZUDANS: Larry, let's see your item 3. I like to go item by item, except for the first item, and Ivan will do that.

Monitor operability of core spray. You say you have nothing else available for that?

MR. PHILLIPS: What I am speaking of is monitoring it is in fact being delivered to the core.

MR. ZUDANS: Don't they have a pressure reading and flow meters and things of that nature?

MR. PHILLIPS: They have flow meters and pressure readings and things of that nature, that is true. I believe that you could postulate conditions whereby those might be misleading as to whether it is actually getting to the core.

MR. ZUDANS: This could also be misleading, because you could be reading temperature that is not at saturation because of blockages and what-not, and the bulk of the core would be all right because you are pumping that water in there.

MR. PHILLIPS: The core would be well covered, I think, under the requirement. That is, with 16 thermocouples, four per quadrant, each one at a different radio location.

MR. ZUDANS: At any rate, it is not unique. It is additional monitoring capability.

MR. PHILLIPS: That is correct.

MR. ZUDANS: Okay.

MR. CARBON: Before we leave that topic, we have had core spray systems on all the BWR's, I think, for a decade or more. What has led you now at this time to conclude that we need some device to monitor their operability?

MR. PHILLIPS: Well, TMI-2 I guess led to the whole question in that we got into a condition where we were not providing adequate cooling to the core, safety injection. It got into the whole realm of multiple failures and being able to detect symptoms of things that have gone wrong.

So essentially where we did not consider such multiple failures before and would consider that the redundant design would effectively get the water there, we now have a different philosophy.

MR. ZUDANS: I would like to talk about item 2 if I can.

MR. CARBON: Yes. Let me finish here.

MR. KERR: We have to get item 3 out of the way.

MR. CARBON: There have been numerous tests, I believe, of the operation of the core spray system and I think they have been reviewed by the staff in the past. Do you feel that those reviews have been inadequate, that people perhaps made a mistake in the past in not saying that they are requiring more redundancy in measuring core spray? I guess I am still troubled or unable to understand really your basis here.

MR. PHILLIPS: The answer is no. We feel those are -- those reviews are good and we feel that the core spray will work. We still are requiring, after TMI, instrumentation to detect symptoms of inadequate core cooling, and one of the symptoms is whether the core spray is getting there or not. That leaves -- that is connected with their approach to inadequate core cooling, and I think in terms of the approach we have a gap between the time that the water level falls in the core and between the time that your damage has already occurred, and that is essentially what the radiation monitors and hydrogen detectors tell you, is that the damage has already occurred.

II.F.2 requires monitoring the approach to inadequate core cooling, and the thermocouples assist in this.

MR. CARBON: I will leave it there, but I guess I still have some questions in my mind.

MR. KERR: Are you on 3 or 2?

MR. CARBON: I will make a comment on both, and on 1.

MR. KERR: I have a precedent. If you're not going to talk about 3, then Mr. Zudans is next.

MR. ZUDANS: 2. On 2, if I read the way you write it, it kind of tells me that you have a continuous type of information and you will be able to kind of monitor the condition. Isn't in fact it true that thermocouples will give you bang, bang, either one end or the other, they would not tell you very much in between, very similar to the other two items you just named, hydrogen monitors and radiation monitors?

If I am wrong, please explain.

MR. PHILLIPS: All right.

(Slide.)

Well, this slide -- something seems to be wrong here.

MR. KERR: I do not know. I understood it better before.

(Laughter.)

MR. PHILLIPS: Well, this is the thermocouple temperature. This would be for a thermocouple located in the thimble at 80 percent of the elevation. And this would be after -- this is the drop in actual level and measured at an assumed boiloff rate at some decay heat level. And this is the clad temperature in the center of the bundle.

And this is a function of time, and as the level is dropping your clad starts heating up, and your thermocouple will follow essentially like this (Indicating). So I would say it is not a dropping off thing. You are measuring a rise as a function of time, and if you get it turned around, your level starts back up, you are going to get a quench and you are going to get a fall as a function of time.

MR. ZUDANS: If you would, on this same plot, the other two indicators' behavior, wouldn't they show some similar behavior, the radiation monitor and the hydrogen monitor?

MR. PHILLIPS: Well, the hydrogen monitor would not show anything until your clad temperature reached approximately 1800 F. and where you start getting a significant amount of --

MR. KERR: 700 K. is not very far from 1800 F.

MR. PHILLIPS: Well, 1800 F. is being rather generous, because that is just where you -- that is just where you start to get a very small amount of hydrogen generation.

MR. KERR: You are --

MR. PHILLIPS: Your radiation monitor is not going to show you anything until you actually get to clad failures.

MR. KERR: And you are saying that having the thermocouple there would tell you to do something that you would not otherwise know about?

MR. PHILLIPS: No. I am saying if you show an earlier indication of approach to inadequate core cooling, and yes, it could give you -- it could tell you to do something like --

MR. KERR: No. As I said, like you otherwise would not know to do?

MR. PHILLIPS: I believe I could answer that yes.

MR. ZUDANS: What about the chance that you would -- I do not know about the time scale. You said the time scale is under question, right?

MR. PHILLIPS: Yes.

MR. ZUDANS: There is an agreement on the time scale? The whole thing happens in two minutes?

MR. CATTON: There is a disagreement, I believe, about two minutes.

MR. ZUDANS: To be useful, this is such a narrow window that you may even miss it, even if you had those things.

MR. PHILLIPS: Well, I do not understand what you means. Two minutes happens to be the scale that shows for this particular temperature rise.

MR. ZUDANS: You recognize that you would have to be watching specifically during those two minutes?

MR. PHILLIPS: No.

MR. ZUDANS: If you miss that point you have other indications, right?

MR. PHILLIPS: No, it continues to go up.

MR. ZUDANS: Yes, but then all hell will break loose, right? At that point you will have a thousand other indications.

MR. PHILLIPS: Unfortunately, this is in degrees Kelvin. I do not think in degrees Kelvin.

MR. KERR: Just multiply it roughly by two.

MR. ZUDANS: I am just wondering whether this window is big enough for the operator to be able to react to it and call it a useful help. Because if you go further in time on this scenario, you will have other indications very quickly.

MR. PHILLIPS: Well, it depends on what is happening, too. Of course, I mean, you may have a situation that you could have a fluctuating level. This assumes a falling level.

MR. ZUDANS: And this is only for two percent power, is that right?

MR. PHILLIPS: Right.

MR. ZUDANS: If the power level is different --

MR. PHILLIPS: It is not going to be much different. We assume we have a condition -- it is more than assuming. If your level instrumentation is working, you have scrammed the reactor long before you get down to here.

MR. KERR: Gentlemen, this is a fascinating topic and I am willing to give up lunch to discuss it, because that is where we are heading.

MR. ZUDANS: I have finished.

MR. KERR: Mr. Moeller?

MR. MOELLER: Back on an earlier discussion of the cost of the installation of such a system, it seems to me that the staff has, particularly in the post-TMI era, required a number of backfits of various systems on various reactors. And I think I would find it very helpful if we could have provided to us in a written report some time in the near future the basic approach used by the staff in deciding whether the cost, meaning financial and health, occupational health impact, is worth the installation of a given system.

And obviously, that depends upon the risk reduction that that system supposedly is going to bring about. And if indeed we could have such a report, in which you would include some information on the length of time over which you amortize these costs, that too would be helpful, because presumably on this system if 100 manrem is correct, on a new plant it might be well worth it, but on a plant that ~ y had five years yet to operate it might not be justified.

So some idea of the formula and approach that you use in reaching these conclusions would be very helpful.

MR. KERR: Are there other questions or comments? Do you have any further presentation, Mr. Phillips?

MR. PHILLIPS: No. I did want to say that we also have Jack Rosenthal here, who has spoken, already spoken, and who is prepared to address any specific questions, many of which you were asking yesterday concerning design requirements with respect to Reg Guide 1.97. And particularly I would like to point out that the PWR thermocouples are safety grade, and in accordance with the requirement they are required to be upgraded to safety grade, and he can address what is needed in order to make thermocouples safety grade, if you want to get into that, and also in connection with the testing.

MR. KERR: Mr. Catton?

MR. CATTON: I would just like to make a comment about the level. I think if the only reason for your temperature measuremen would be diversity in level measurement, I really cannot see that they are necessary. On the other hand, you have the requirement from Reg Guide 1.9, to follow the course of the accident beyond the usual

DBA. It seems to me then that the temperature is the only way you can follow it.

MR. PHILLIPS: Let me say that it is a close call. The staff considered it and we reached a judgment.

MR. CATTON: That is based on 1.97.

MR. KERR: Saying something is required in Reg Guide 1.97 leaves me cold, because I have seen several versions of Reg Guide 1.97, and I would hope that some day it will improve. What we need to find out I think is whether the information provided is useful and it reduces risk, and if it does not then it ought to get out of Reg Guide 1.97 and if it does it ought to be there and maybe a number of other places.

And this is the reason I do not have very much patience with somebody showing me that it is in NUREG-075-whatever, and 197, and 1.13. That is irrelevant.

What we need to know is, does it give the operator something he needs in an emergency situation and does it thereby reduce risk. And if it does, does it cost little enough that one wants to use it, it seems to me.

MR. CATTON: You are right, and in my mind where it is at, if you know how quickly you are getting in trouble, does it help you. If it does not --

MR. KERR: We do not know at this point whether it does or not. Larry thinks it does and ay be right.

MR. CATTON: I am not sure Larry has looked at it.

MR. KERR: He has been living with it for some time, I think.

MR. PHILLIPS: Yes, I have.

MR. KERR: And GE thinks it does not, and Detroit Edison endorses GE's position 98 percent.

(Laughter.)

MR. KERR: Any more questions?

(No response.)

Thank you very much, Mr. Phillips.

Mr. Colbert, are you going to make the presentation on any other thing you want to say about 1.97?

MR. COLBERT: Yes, Dr. Kerr. The items besides -- the items in addition to incore thermocouples that are in Reg Guide 1.97, I would like to have Larry Wooden, our assistant engineer, IoperatiC, address the remaining part.

MR. KERR: Thank you.

MR. WOODEN: My name is Larry Wooden. I am Detroit Edison's IoperatiC systems engineer.

Following the TMI event -- we previously had thought we had pretty good degree of compliance with Reg Guide 1.97, and of course following TMI everything changed. And as a result, we were forced to modify our existing post-accident instrumentation system.

Basically, we addressed all the concerns in the NUREG-0737, and with the exception of the incore thermocouples we think our design presently is in pretty good agreement with Reg Guide 1.97. And I just want to touch on some of the changes that were made in order to address the NUREG and some that were made in fact to address some of our own internal concerns following Three Mile Island.

(Slide.)

This is some of the major instrumentation systems that were affected, that were directly associated with the reactor vessel and the primary containment integrity itself. First and I guess paramount is the fact that we are adding a positive indication for safety relief valve open and closed.

Actually, the first major adjunct is the safety relief valve indication, of which there is 15 in our plant, and we chose to use the GE method of measuring the discharge pressure in the line. The sensors will be located in the drywell and they will communicate directly with the annunciator sequencer recorder in the control room to alert the operator of the exact status of the safety relief valves.

The next major item that is a very expensive installation is the high-range in-containment monitors, which are single failure-proof and to the best of our ability made to be completely in compliance with 323 and 344.

In addition to that, we were forced to modify the ranges on the wide-range level measurement of the torus water level. This is in order to address the range requirements of Reg Guide 1.97 and also stipulated in NUREG-0737. And we agreed basically with the fundamental change. It does degrade the day to day operation of the instrument to some degree, but in the overall run it probably is a good thing to do.

Here we have modified an existing single failure-proof drywell pressure measurement system to incorporate the guidance of NUREG-0737 in that we have expanded the range to the design point, four times the desig point of the containment itself, and also included a narrow range so that the operator is able to more closely track the day to day operation of the plant.

We previously had a hydrogen-oxygen monitoring system and we have upgraded that to some degree and made a minor change in the range in order to meet the NUREG requirements. But we found that basically the system that we had procured and installed in early '75 met the new requirements.

MR. MOELLER: You say you upgraded it. What did you do to it?

MR. WOODEN: Well, at the direction of our internal safety review task force, we made some significant changes in the piping configuration inside the drywell to make the sample somewhat more uniform and representative. And we also procured a spare high-speed sampling pump because we determined it would be a long-term item, in that the importance of this measurement, that would be in our best interest to have, you might say, a completely acceptable replacement spare available if some unfortunate thing like the Three Mile Island event would ever occur at the Detroit Edison site.

And up until that time probably some of us did not take the LOCA scenario as seriously as we should have.

MR. MOELLER: Have you looked at -- I mean, it is good to have a reserve pump. But have you looked at trying to get a pump that is much more reliable than those currently being used?

MR. WOODEN: We did not because it was our considered opinion that we had exercised -- well, all the efforts we could expend in that direction, because actually the pump was procured, finally procured, the design, almost two years after the rest of the system was ready to operate, because of our requirements for a glanulus non-shaft pump that is completely enclosed in a capsule to maintain the pressure integrity.

And this particular aspect of the job was handled by one of our more experienced people that have worked on Fermi 1 for a number of years and had used his past experience with the unsatisfactory performance of blowers in this application to try and get the most satisfactory system, you know, pumping design for this application. That is why we did not really consider that to a great degree. We thought we had the best.

MR. MOELLER: We heard earlier today, of course, about the efforts of the Applicant to monitor LER and feed the information back into the system. And I would simply point out that I hope you will give attention to the noble gas monitors, iodine, particulate, and the hydrogen and the oxygen monitors, because the data show that from five to ten percent of all of the LER's at operating BWR's are failures of these monitoring units.

MR. KERR: Please continue, Mr. Wooden.

MR. WOODEN: The last but not least is the post-accident sampling system, which is of course a new adjunct to all the pressurized and boiling water reactors. Our particular system is, after many gyrations, has turned out to be a basic GE design, again in concert with the Owners Group because we felt we wanted a system that, you know, would be more similar to all the other plants, that we would gain operating experience with it more rapidly and determine its effectiveness in the shortest possible time, because this is the kind of an area where you hopefully will never use it but you would like to have something as good as anybody else in the world has got, so that when you are in a group like that you get a lot of operating experience and feedback a lot faster.

This system has the capability of taking coolant samples. It can take containment air samples, suppression pool water samples and suppression air space samples to a safe location. And I believe that we will be able to, even though we did not commit earlier because of a misunderstanding, we believe that we will be able to adequately remove a sample, a concentrated sample, and perform all the analyses on the coolant that are required in Reg Guide 1.97, Revision 2.

(Slide.)

The next major area that has been modified on our plan -- again, all these systems basically, with the exception of the ones that were added, were either completely designed and almost totally installed or else partially installed.

In this case in order to meet the guidance of the NUREG for the iodine and gaseous effluent monitoring of the stacks of the plant, we had to essentially scrap the existing equipment, at a fairly large expense to us because we had bought what we thought was the Cadillac of the industry. At the time we had Gulf Atomics and we chose to go to the Eberline system.

It is a modern concept, digitally based microprocessors. It is located near the sampling area. And of course, we will accrue some advantage, but it still is a very expensive undertaking because of the scope.

We have a number of stacks on this particular ventilation design, so that we were forced to replace more than the usual one or two monitors. As you see, we have replaced the standby gas monitors with two Sping 3's, being an accident range monitor. That is an additional component that is added to the Sping 3 to increase the range, plus we have a special provision for particulate and iodine sampling.

On the reactor building, we have replaced the existing monitor with a Sping 3 with a high range, to attain the 10^5 range that is required. Even though this particular monitor will trip the ventilation system at a much lower radiation, we have provided this in case of under duress it might be necessary to run that reactor building ventilation system. We would then in that case have provided accident range to 10^5, as required in the NUREG.

The radwaste building ventilation system has a Sping 3 monitor. The service building and the turbine building exhaust have a Sping 3 monitor. In each case, all of the monitors trip the particular ventilation system off on high radiation, so that we probably will never attempt to run these systems under the degraded condition, wherein the radiation is actually above the magnitude that is allowed to keep the ventilation system operating.

That is basically all the changes that we have accomplished to date. Most of them are under way. The majority of the equipment is procured, and hopefully we will not have to make too many additional changes to meet -- satisfactory to comply with Reg Guide 1.97, Revision 2, requirements.

MR. KERR: Are there questions for Mr. Wooden?

(No response.)

MR. KERR: Mr. Wooden, what is a Sping 3 monitor?

MR. WOODEN: That is actually the model that Eberline --

MR. KERR: It is a brand name?

MR. WOODEN: Right.

MR. KERR: That is all I wanted to know. I thought perhaps it was some acronym.

MR. WOODEN: It is basically their model name. It is one of the two in the industry that is currently being used. Gulf Atomics has a wide-range version also.

MR. KERR: Thank you very much.

A matter of logistics. I have asked the next presentation -- or asked th he next presentation be cut in half, if we can do that, that on seismic design discussions. And at the end of that I propose to break 30 minutes for lunch, because I do not propose to stay in Washington over the weekend and I would like to finish up this process. So I hope I am not interfering too much with your lunch habits by restricting today's lunch to 30 minutes.

Mr. Gregor?

MR. GREGOR: Good afternoon. My name is Frank Gregor. I am the assistant to the technical director, Fermi 2 Project, Detroit Edison.

I would briefly like to review the seismic design of the Fermi 2 plant, and in particular the recent seismic reassessment that was conducted. On March 12 of this year we received a request for information from the NRC staff to re-evaluate our Fermi 2 seismic design basis. We were basically given the option to utilize a Regulatory Guide 1.60 type spectrum anchored at 19 g or develop a specific spectrum site specific for rock site at magnitude 5.3 earthquakes.

In follow-up of that request we met with the staff on March 27, 1981, and at that time we committed to the form -- to utilize a different site spectrum, which would be

Fermi 2 unique. This spectrum would envelope the Lawrence Livermore spectrum and utilizing in addition a spectrum developed by our seismology consultant, Western Geophysical, who developed the site specific rock site spectra. And I have a slide to briefly show the spectra.

The Lawrence Livermore spectra is shown here. (Indicating). It is the solid line. The Western Geophysical spectra is shown as a dotted line here (Indicating). And the site specific spectra that was chosen for the Ferm 2 re-evaluation was drawn to envelope both of those spectra and in general tracking the Regulatory Guide 1.60 spectra shape.

This spectra also included far field effects of strong motion earthquakes and is anchored at 15 percent g. When comparing this new spectrum to the old design -- to the original design basis spectrum for Fermi 2, one can see, even though, that the zero frequency or zero period acceleration value of 15 percent g remains basically the same, but due to the shape difference in the dominant structure frequencies the increase in acceleration value is roughly about 60 percent.

(Slide.)

For the vertical earthquake, the vertical acceleration, we determined to use a factor of -- multiplying factor of two to the original design basis earthquake. This was done in the absence of adequate historical records for vertical acceleration earthquake histories.

We did compare -- I have a slide to show this briefly.

(Slide.)

We did compare our regional Fermi 2 vertical acceleration values, which are based on four individual time histories from earthquake records, and superimposing here the Regulatory Guide 1.60 spectra. And in ratioing these two spectra, we found roughly a multiplying factor of 1.6. However, due to the valleys in the synthetic time histories or the average of those four time histories, as you can see here there are valleys, they tend to drive the average spectra down.

Our structural consultant, Sargent Lundy, advised us to up this modifying factor to 2. So that is basically the basis for that number, and that was used in the re-evaluation. And also, we did apply those spectra directly to the floor response factor.

In respect to the OBE, no specific change was made. Our seismology consultant did, however, conduct the historical search of actual earthquake recordings in the central stable regions, and on that basis we found good for our OBE as a minimum for 100 to 300 years, which is well within the design life of the plant -- or outside the design life of the plant, which is 40 years.

(Slide.)

With the new criteria at hand --

MR. MOELLER: When you say 100 to 300 years, of course they are both longer than the 40-year lifetime of the plant. But what is the probability of an event that occurs once every hundred years occurring during the 40-year life of the plant?

MR. GREGOR: I have to refer that to our seismological consultant. Can someone from Western Geophysical answer that?

MR. MOELLER: If it has a short answer.

MR. GREGOR: Dr. Holt from Western Geophysical, come to the microphone, please.

MR. HOLT: My name is Richard Holt. I am with Western Geophysical.

We did not carry out a formalized probabilistic study. It was simply a study plotting historical events. I do not think, Dr. Moeller, we have the answer for you.

MR. KERR: Let me give a simplified answer, then. The probability of an event which happens once every 100 years happening in 40 years is .4. That is a rough estimate.

Mr. Ray?

MR. RAY: I have a corollary question. If you cannot give us a probability, can you tell us when the last one occurred? In other words, when does the measuring time for 100 to 300 years start? When the plant was built, 100 years ago, or 300 years ago?

MR. HOLT: The historical period was 1776 to 1976. It was 200 years of data.

MR. KERR: But I would -- the answer to your question, Gerry, is it starts at the time you start operating the plant.

MR. RAY: Well --

(Laughter.)

MR. KERR: You know, you measure from zero to 40.

MR. MOELLER: Why don't you start it shortly after you shut the plant down?

(Laughter.)

MR. KERR: Because that does not count.

MR. RAY: I would say it starts with the last earthquake you measured in the period.

MR. KNIGHT: Jim Knight from the staff. A shot at perhaps perspective.

We are talking about the operating basis earthquake?

MR. KERR: Yes.

MR. KNIGHT: As far as we are concerned, the probability is one will occur. It is designed for it. That is the assumption.

MR. GREGOR: Thank you.

(Slide.)

MR. KERR: But you start counting at the beginning of the plant life, and from there to 40 years the sum probability is one, right?

MR. KNIGHT: Yes.

MR. KERR: That is a non-mechanistic analysis.

Please continue, with all those irrelevancies --

MR. GREGOR: With that criteria in hand now, we proceed with the re-evaluation program. And basically, we proceeded as follows: We determined the shutdown scenario that will be associated with the earthquake and the loss of offsite power situation. We determined then the systems, structures and components that are required for cold shutdown of the plant.

This assessment was done not in combination with the loss of coolant accident. In all other respects of the re-evaluation, however, we did use the ground rules of Reg Guide 1.60, 1.61, the standard review plan, et cetera.

The evaluation was completed in May of this year. The preliminary report was filed with the staff to provide the detailed information that was developed, and it was followed up by a final report that was docketed last week at the public document room.

I would like to briefly get to the bottom line of my presentation here as to what we found in this re-evaluation. I think that is the most important aspect of the whole analysis.

MR. KERR: I am with you, especially the "briefly" part.

MR. GREGOR: The bottom line is that we concluded that the plant is indeed capable of shutting down, cooling down. In the initial quick assessment, we did find some structural components that were somewhat over code allowables. We have gone back and done some pencil-sharpening calculations and were indeed able to s... that, with the exception, I believe, of one cable tray hanger that is still slightly overstressed, that we are within the yield for all structures.

In addition to the structure assessment, a lot of the mechanical equipment and components, including piping systems, supports, pumps, valves, heat exchangers, et cetera, we evaluated. Out of that whole list of components, which constituted roughly 200 different kind of components, representing some 500 to 600 individual components, items or components, out of that many we found 25 specific items that are still open items as far as we are concerned, that require some kind of further evaluation by more sophisticated analyses.

In some cases we were unable to show that the test response spectra that the equipment was actually tested to is indeed enveloping the new floor response spectra, and we may have to do some retesting or in situ testing. In some cases it is just a paper trace, records that have to be accumulated. 25 items only are left.

We also, in addition to that, had an independent assessment made of the overall structural behavior of the building and the equipment within, and that independent assessment concluded that if one were to assume or allow immobilization of ductility of less than the ratio of two for the mechanical structures and components within the building, that -- and one would apply then this ductility to modify the existing response spectra for that ductility or account for that, then the revised spectrum would fall below the original DBE spectrum that was the basis for our plant.

The detailed comparison and methods of reducing these spectra and accounting for ductility ratios have been provided in our special report, section 7. Our work, in addition to documenting with the NRC, was also audited in two particular audits during this week, namely in the structural design area and the mechanical component design area. And I would not want to steal the staff's thunder. They are here to report on those results.

And that basically concludes my presentation.

MR. KERR: Thank you, sir.

Mr. Zudans.

MR. ZUDANS: This stress analysis report that was prepared by you I did review, not in complete detail, line by line. And I was happy all the way except in one location, and I want to bring that up.

There is a place in the report that seems to me completely wrong. It may have no effect on your results, but in this report, which says A4.1 -- it comes in a separate volume -- on page 29 when the vertical acceleration is used to compute the loads to the reactor pedestal, only the pedestal mass is included, not the reactor vessel density, not the sacrificial shield, or anything else that rests upon it vertically.

Was the analysis done for the sacrificial shield? Only the weight of the sacrificial shield is taken into consideration. Now, unless there is someplace else where everything that sits on the pedestal is included as initial loading, then this is incorrect. It may not have any effect because it may not add much to the stress.

But what it does to me is it raises a question how good is the rest. Although I went through the results, I am not that convinced any more.

MR. GREGOR: I am not sure whether our statements in the report are indeed correct. I am going to try to find out what the effect is, and I have Dr. Sing from Sargent Lundy, who did a lot of the structural evaluation, re-evaluation in generation of the shared moments and so on in the structures and have him address that.

Dr. Sing, would you come to the microphone?

MR. SING: A.K. Sing, Sargent Lundy.

As, Professor Zudans, you have said, the stress levels for the reactor pedestal and the shield walls for the reassessment are particularly low, and the reason they are low is because the LOCA loads essentially are part of the design. In calculations of any structure which supports a component, in this case the reactor, what you have is -- or for that matter piping -- the only reason that in this analysis the piping -- the reaction of piping mass is not included, is that reaction comes as part of the piping reaction.

MR. ZUDANS: It is not the piping reaction we are talking about. The reactor pedestal -- on top of reactor pedestal is sacrificial shield and other components, too.

MR. SING: Correct.

MR. ZUDANS: On top of reactor pedestal is reactor vessel set.

MR. SING: Yes.

MR. ZUDANS: When you calculate the vertical force, all you use is reactor pedestal weight. A page later, when you come to the sacrificial shield, you only use sacrificial shield. And yet the sacrificial shield sits on top of the vessel and they are about the same weight.

I mean, I am not saying that the stress, the vertical stress because of this weight will significantly affect the result. I am only saying that this is so incorrect that it tends to cast doubt on the rest of it.

MR. SING: We will have to review that particular comment. But this is one area where we went into great detail with the staff. And it was my feeling that these calculations are correct to the extent presented over there. I will have to review it, the exact question in detail, and then would be able to provide it to you

MR. KERR: I would suggest you make certain you understand the question Dr. Zudans has.

MR. SING: I believe I do. The question refers to that in the design of the pedestal whether we included the correct reaction from the reactor and the shield wall.

MR. ZUDANS: Whether you included it at all. It seems like you did not. Maybe I am missing something that is not shown there.

MR. SING: Fine.

MR. ZUDANS: Okay.

MR. KERR: Are there other questions.

MR. ZUDANS: I also have another comment on this independent or alternate analysis that you did, by calling upon ductility. I think in terms of acceleration reduction, it is probably proper, but in terms of relative displacement it is not that simple a decision. Even if you bring your response spectra to that of the original spectra, you are making a fairly good assumption with the spectra observations.

But when you do that ductility method, there is no simple relationship. It is not a linear case. That means you may have a large relative displacement on piping systems, for example. One end point may move a lot more than the other. So if you are requiring a ductility of two to survive, then you are forced to go back and examine relative motion of separate points, which is not done any place in the calculation.

You use some average spectra for a whole piping system, but the end points you do not look at. Fortunately, it seems you do not need to invoke much of the ductility.

MR. GREGOR: It was purely done for the purpose of providing another overview as to, you know, what the real earthquake, almost twice the magnitude for which a design event -- what it would would really do. Cable systems are flexible. Pumps, heat exchangers, their support legs are usually quite flexible so to absorb significant deformations.

And you know, given a ductility ratio of two, which is quite significant --

MR. ZUDANS: In spite of these comments, I say that my personal feeling is this is okay.

MR. KERR: Do you have any further comments, Dr. Sing?

MR. SING: I agee with the last statement that Dr. Zudans made.

MR. KERR: You mean the statement that the calculations are correct, that statement?

MR. SING: I was meaning the ductility one. I guess it was pointed out that we have gone through the whole structural evaluation. We do not need the ductility. The ductility was a parallel study which was done in c̃ se, if we had designed the structure to the limit in the first place, which was not the case.

MR. KERR: If I understood Mr. Zudans' comments, it was that the ductility consideration does not have much to do with anything. In his view, the way it was done -- since you do not need it, I guess it is okay.

MR. SING: That is right.

MR. KERR: Are you complete with your presentation?

MR. GREGOR: I am done.

MR. KERR: Does the staff have any comments on that?

MR. KNIGHT: Just again, very briefly. As pointed out, we did have an audit this week. Both the structural and mechanical people went to their Detroit offices. Although we do have two or three items that will require further discussion, the level of damping to be used, the level of structural damping to be used in terms of generating floor response spectra for equipment qualification, it is an item that needs to be discussed between us.

In actuality, what appears to be the situation -- and again, I approached this fact last night. We have not had a chance really to put all our heads together. It appears the stress levels are low, so low that a little lower damping would be appropriate. And there is a question about the application of response spectra in some of the GE-supplied equipment, the methodology.

Our overall assessment, however, is that the type of changes that would come from what might be said the nitty-gritty of getting the open areas straight are not significant in terms of the plant being ready. The criteria which were used, the acceptance criteria which were used in the design of mechanical systems were quite conservative, and our reviewer generally feels, well, should there be some increases they would still be within safe limits.

So overall, we come back with a very positive prognosis. Some details can be cleaned up. And I must say, the equipment qualification people are going out next week.

MR. KERR: Help me a little bit. Just out of curiosity, why was Fermi required to re-evaluate and Susquehanna was not? Was it because of a re-evaluation of the Indiana earthquake or what?

MR. KIMBALL: The Susquehanna, to the best of my recollection, had used a Reg Guide 1.60 type spectrum or something close enough to that *.at the difference between their smooth design spectrum and Reg Guide 1.60 was minimal, in this case a Housner spectrum which was used basically back in the early 70's and late 60's for this plant.

There is quite a difference between the Reg Guide 1.60 spectrum and the Housner spectrum. All the operating plants that are coming in for an OL review that have used the Housner spectrum ought to be looked at in the same light as Fermi was looked at.

MR. KERR: Thank you.

Any other questions or comments?

(No response.)

We will stop for lunch and reconvene at ten after 2:00.

(Whereupon, at 1:40 p.m., the Subcommittee was recessed, to reconvene at 2:10 p.m. the same day.)

AFTERNOON SESSION

(2:15 p.m.)

MR. KERR: Let us resume.

I show decay heat removal, W.F. Colbert.

MR. COLBERT: That is what it says. Mr. Mohan Deora will present this part of the agenda.

MR. KERR: All right.

MR. DEORA: My name is Mohan Deora. I am system engineer, Detroit Edison Company, Fermi 2 project.

The topic for my presentation is decay heat removal capability under normal and degraded modes.

(Slide.)

In this presentation I am going to go over our normal mode of removing decay heat to main condenser and RHR shutdown cooling mode, and then we get into the degraded mode where, if the normal mode of decay heat removal is not available, we have the alternative modes to remove decay heat.

The principal modes for removing decay heat at Fermi 2 is the shutdown cooling and the containment cooling mode of the residual heat removal system. And the primary objective of decay heat removal function is to transfer the decay heat from the core to the ultimate heat sink, where it is dissipated to the atmosphere.

If we are in normal mode of operation and normal shutdown while the reactor is at high pressure, the pressure being higher than 100 psi typically, the decay heat is transferred to the main steam line, to the main condenser. While the reactor at low pressure reaches typically less than 100 psi, the primary source for removing decay heat is the shutdown cooling mode of the residual heat removal system.

(Slide.)

As depicted in this schematic here, in the shutdown cooling mode of the RHR, the reactor coolant is pumped from the recirculation mode to the RHR pump.

MR. CARBON: Could you use the pointer, please, and step back?

MR. DEORA: Okay.

In the shutdown cooling mode of RHR, the primary coolant is pumped from the recirculation loop with RHR pumps through the RHR heat exchanger and is returned back through the circulation system back to the reactor vessel. The decay heat is removed through the RHR heat exchanger, where it is transferred to the RHR service water, which is finally dissipated through the RHR cooling towers to the atmosphere.

In the normal mode of operation, operator has to open these two valves, the one inside isolation valve next to the primary containment boundary and the outboard isolation valve, which are fed from the opposite division, inside valve being fed from the region one and the outside valve fed from the region two.

Now, in a degraded mode --

MR. KERR: "Fed" means supplied with electrical energy?

MR. DEORA: Electrical energy.

MR. KERR: Thank you.

MR. DEORA: In the degraded mode if, due to loss of offsite power or the main condenser being not available and if the reactor is still at high pressure, the decay heat can be removed through the safety relief valves, where it is discharged to the suppression pool and from where it can be removed through the RHR heat exchanger.

Once the reactor is depressurized below 100 psi, the operator has to open these two valves to allow the -- to establish a cooling path. In the degraded mode, if besides the loss of offsite power we have to use and feed to one of these valves, that means losing a divisional power feed now, if he has a loss of offsite power with a loss of division one. In Fermi 2 design we have provided two parallel valves inside the primary containment, as shown here. And these valves are fed from the opposite divisions, this from division one and division two.

So loss of offsite power with a loss of divisional power would still provide the capability by which we can still open the inboard isolation valve when our primary preferred path is not available and we can establish the cooldown path by opening the division two valve, and outboard valve is also in division two.

Now, in case we have a loss of divisional two power, then a handwheel is provided for this outboard isolation valve for a manual operation, and operator by manually opening this valve can establish this preferred cooling path.

MR. KERR: Physically where is that manual valve located?

MR. DEORA: This valve is located in the reactor building.

MR. COLBERT: It is at the operating floor level in the reactor building.

MR. KERR: Would you identify yourself, please, sir.

MR. COLBERT: Bill Colbert, technical director.

The valve in question is located at the operating floor level in the reactor building.

MR. KERR: Thank you.

MR. DEORA: Now, in the unlikely event where we may lose this preferred path for removing decay heat, there is an alternate path available to the operator.

(Slide.)

This is shown and depicted in this schematic here. Now, in this mode of operation for removing the decay heat, the reactor pressure vessel is flooded with water up to the main steam lines and the primary coolant carrying decay heat is forced through the safety relief valves down to the suppression pool. Thus the decay heat from the core is transferred to the suppression pool and the containment cooling mode of RHR, which uses RHR heat exchangers, will then transfer the decay heat from the suppression pool through the RHR heat exchanger and through the RHR service water ultimately to the atmosphere.

Now, the other degraded mode could be if a reactor shutdown is initiated from an accident, typically like a loss of coolant accident event --

MR. ZUDANS: Could you answer one question?

MR. DEORA: Yes.

MR. ZUDANS: Normal shutdown or this ultimate shutdown cooling mode, how many RHR cooling pumps do you need for this function?

MR. DEORA: Four -- now, a detailed Fermi 2-specific analysis has not yet been done. In terms of my guess, maybe we need only one pump.

MR. ZUDANS: You have a total of four?

MR. DEORA: Four pumps, right.

MR. ZUDANS: Two are now used for core spray and two are used for pumping.

MR. DEORA: No, we have four RHR pumps, two in each division, and we have four core spray pumps, two in each division. So there are a total of eight pumps.

MR. ZUDANS: I thought the same pumps were used for both functions.

MR. DEORA: You mean the RHR pumps? Yes.

MR. ZUDANS: The dedicated core spray pumps?

MR. LUSIS: The core spray pumps and the RHR pumps are separate sets of pumps.

MR. ZUDANS: Separate set of pumps.

MR. KERR: Thank you. Please continue.

MR. DEORA: Now, as I was saying, that if a plant shutdown is initiated from a loss of coolant accident, that type event, then the primary mode of removing decay heat is depicted in this diagram here.

(Slide.)

Now, for a loss of coolant accident type event where integrity of the primary coolant system is breached, the primary coolant is released to the primary containment and the heat is absorbed in the suppression pool. So the decay heat is transferred to the suppression pool direction, and from there the decay heat can be then removed using the RHR decay heat removal mode, where RHR pumps can take suction from the suppression pool to the RHR heat exchanger and ultimately it is dissipated to the atmosphere.

Now, in this mode of operation the capability of the system to perform its safety function it is one of the criteria that the suppression pool temperature has to be limited below 200 degrees Fahrenheit, and the system that we have at Fermi 2 has that capability even under degraded mode, by which I mean we have four RHR pumps. Only one of the RHR pumps available and with one of the two RHR heat exchangers and two of the four RHR service water available, the suppression pool temperature is maintained below 200 degrees, even for a design basis accident, which is a double-ended break in the recirc line on the suction side.

MR. ZUDANS: Within the same scheme, you also have suppression pool spray.

MR. DEORA: Yes, we have suppression pool spray.

MR. ZUDANS: Which pumps feed that? The same RHR pumps?

MR. DEORA: The same RHR pump.

MR. KERR: Please continue.

MR. DEORA: That concludes my presentation.

MR. KERR: Are there other questions?

MR. ZUDANS: All these heat sinks that you describe, are they all in RHR complex, right?

MR. DEORA: Okay. I -- yes.

(Slide.)

This is one divisicı of RHR complex.

MR. KERR: Other questions?

(No response.)

Thank you, sir.

Mr. McKelvey.

MR. McKELVEY: My name is Terry McKelvey. I am the principal electrical engineer involved in the design of Fermi 2. And I am going to present to you the reliability of our station power, including the effects of the loss of various stages of AC and DC, and get into the discussion on station blackout.

In order to do that, I would like to give you a feeling for the diversity of our offsite power source.

(Slide.)

Also for its independence and how we derive it from our electrical transmission system. First of all, the offsite power source at Fermi 2 is made up of five separate transmission lines, two from our 345-kV system and three separate lines on the 120-kV.

As you can see here, the 120-kV lines are from three separate stations on our system.

MR. CARBON: Could you step back and use the pointer, please.

MR. McKELVEY: Okay.

As you can see, the three 120-kV lines originate at three separate locations in our 120-kV transmission system. The 345-kV lines, transmission lines coming to Fermi 2, are both from our Brownstown station, which actually is a split station. It is treated as if it was two separate stations. They come into two separate switch areas located on the Fermi site.

The existing 120-kV switchyard was for the Fermi 1 plant. It has been in existence for approximately 14 years. The 345-kV switchyard is for the transmission of power offsite from Unit 2.

As you can see here, we have a tie to Consumers Power in the area. It has the Whiting power plant. It is approximately 20 miles from the Fermi site. In addition, we are tied to Consumers Power in nine different locations. We are also tied to Ontario Hydro and Toledo Edison. The combined total of our ties to neighboring facilities is 7300 megawatts capacity.

MR. RAY: Those other locations are from other points on the system?

MR. McKELVEY: Yes, there are nine separate points there where we tie to Consumers. For example, two separate Ontario Hydro and three separate transmission lines to Toledo Edison.

MR. RAY: Are any of these ties to other pools at

345-kV?

MR. McKELVEY: Yes, they are. The ties to Toledo Edison are at 345, and also to Ontario Hydro.

MR. CARBON: How widely separated are the five lines coming into the plant?

MR. McKELVEY: Okay. There is a common transmission corridor into Fermi 2 for approximately five miles. That is as shown here (Indicating). The line -- the corridor is 500 feet wide. It has been designed to take the worst failure of any structure falling over that would leave enough lines -- all we need out of these five lines is one separate line existing to power adequate ESF at Fermi 2.

So we have designed the spacing of the transmission lines on this common corridor to accommodate the falling over of a 345-kV tower.

MR. CARBON: Each of the lines has its own --

MR. McKELVEY: Support of tower, right.

MR. KAY: When you say they are designed -- the falling tower will not impact an adjacent line?

MR. McKELVEY: There are five separate lines. The 345-kV towers are on the outside of the 500-foot right of way. In the center are the three 120kV lines. Obviously, the largest structure is the 345-kV tower and if one of those lines fell over across the other transmission lines we would still have adequate transmission lines available to shut the plant down.

MR. RAY: You have not answered the question, but you have given me an implied answer. I gather from what you say that if the EHF line went down it would take down one of the 120-kV lines, is that what you are saying?

MR. McKELVEY: Yes, it would.

MR. RAY: So you would have three left.

MR. McKELVEY: It might take a couple lines down. We would have at least one transmission line available to the site.

MR. RAY: All five lines are on the same right of way?

MR. McKELVEY: that is right.

MR. RAY: The two large ones are on the outside. So between the two large ones there are three others.

MR. McKELVEY: Right.

MR. RAY: Are you saying, then, that if one of the 345-kV lines failed it would take out three 12-kV lines with it? It would take out three 120's? So a falling tower on the high line could leave you with one line.

MR. McKELVEY: That is right, in the worst scenario.

MR. RAY: Is the system stable under these conditions?

MR. McKELVEY: Yes. The system has been analyzed with load flow and system stability programs for that kind of contingency.

MR. RAY: Using four lines you maintain stability of the system, it does not break up on you and thereby deny AC power to the system?

MR. McKELVEY: That is right.

MR. RAY: What kind of faults were assumed for that stability study? Can you tell me how comprehensive it was?

MR. McKELVEY: I did not take part in that stability study. I am afraid I cannot. I can get the information for you.

MR. RAY: The implications here are if you take out -- if there are four lines, one failure, they are all poly-phase, three-phase faults.

MR. McKELVEY: They are all three-phase faults, yes.

MR. COLBERT: Characteristically, all of our studies take into account all kinds of faults, that is three-phased faults, double phase to ground and single-phase faults. And we would take into account the various loss of generators under certain conditions with a reduced transmission system, and then see if the system remains stable and if the generators go out of step.

MR. KERR: Thank you.

Mr. Carbon?

MR. CARBON: Will a typical Midwestern tornado take down one of those power lines or all of them? Will it knock down some transmission lines like that?

MR. McKELVEY: In the common corridor?

MR. CARBON: There or anyplace, is the power line susceptible to being knocked down by a typical Midwest tornado.

MR. McKELVEY: Yes.

MR. RAY: It has happened.

MR. McKELVEY: However, I have one statistic. In the last ten years on the Detroit Edison system we have only lost two towers, two 345-kV towers to tornadoes.

MR. RAY: I notice in some of the background on your station -- I cannot say now where I picked it up, but on the Fermi site you have a 165 oil-fired unit and four small peaking units. Are these connected into the 120-kV system at that substation?

MR. McKELVEY: Yes. We will be getting into that if I can continue.

MR. CARBON: One more question. The supports for those transmission lines, are they designed to resist an SSE earthquake?

MR. McKELVEY: No, they are not.

MR. KERR: Please continue.

MR. McKELVEY: The two switchyards at Fermi 2 -- and this is a very brief description of them. The 120-kV switchyard is located at the Fermi 1 plant.

(Slide.)

It has three separate lines coming from the three separate stations. Two of the lines feed bus 102, with a tie to our bus 101. The Fermi 2 auxiliary electrical system offsite power feed is developed from that bus 102.

In the 345-kV switchyard we have two separate lines, as I said before, from our Brownstown station in a breaker and a half scheme. We also have the Fermi 2 generator, which generator voltage is transformed to the 345 transmission network. And also we have a tap off of bus 301 which feeds another system service transformer for the second source of offsite power.

I wanted to make a note that we do not use the unit auxiliary transformers to eliminate the switching.

MR. RAY: Is that bottom feed to the right that goes downward -- I cannot read it. The 120-kV, is that the point of connection of these other units that are on the Fermi site?

MR. McKELVEY: Right. The next slide will show that. Note that this is bus 102.

(Slide.)

The units in question, bus 102 and feed onto the onsite power system, goes through a transformation down to an intermediate voltage where the peakers -- this symbolizes two peakers -- are tied into the system, two other peakers on a second wing. Those ar. for peaking duty. That voltage is again transformed through a system service transformer into our plant, into our ESF buses.

The two switchyards, I did not mention before, are located approximately a quarter mile apart, so they are very independent. The control power, the battery power, is also derived from separate battery stations.

The auxiliary electrical system is divided into two divisions. Again, one division is completely normal. Offsite power is fed from -- ultimately from the 345 and transformed down to two 4160-volt buses. Inside each division, each bus is backed by an emergency diesel generator at 4160 volts.

There are also associated 480-volt buses inside each division. They are divided up into load groups associated with each emergency diesel. We have intra-divisional ties between the 480-volt buses. Should either a bus or emergency diesel, for example, fail in an emergency situation, we could power one of a kind loads from either load group from the remaining diesel inside a division.

Similarly, for the division one power the ultimate power source is the 120-kV system. Again there are two buses inside the division, backed by emergency transmissions for transmission at lower voltages. Again, any one of a kind type load could be powered from either remaining of the load groups.

MR. RAY: Where is the 165 megawatt oil-fired unit connected?

MR. McKELVEY: It was tied to the other bus, bus 101 in my previous slide.

MR. RAY: I see it. I am sorry, I missed it. Yes, thank you.

(Slide.)

MR. McKELVEY: Getting into --

MR. COLBERT: I think that should be pointed out that that is an oil-fired peaker. It is an economy reserve. It is not normally running. It has to be started up, and it can be started up in a relatively short time, half-start, meaning on the order of hours. That is, it would be on the order of two to three hours to start it up.

MR. RAY: If you lost all other AC power off the site, could you live while you were starting this unit up?

MR. COLBERT: Could you repeat that?

MR. RAY: If you lost all other transmission into the site and you had -- and your diesels did not respond and there was an AC blackout, could you start up this -- could you survive while you started up this fossil unit?

MR. COLBERT: The term "survive," yes, we can black start that power plant. It is equipped with diesel engines to start with its own battery. By the way, that is not a small unit. That is 60 megawatts power.

MR. McKELVEY: You are sort of stealing my thunder.

MR. RAY: I was talking about the one big unit first.

MR. COLBERT: At the present moment that is an economy reserve and I was mistaken when I said it could be started in two to three hours. That would be if we were operational. But right now that plant is not manned. We have to make a conscious decision to send the people down there to man that plant.

MR. RAY: So you have your eye on --

MR. COLBERT: That is correct.

MR. RAY: How fast can they start?

MR. McKELVEY: Ten minutes.

MR. RAY: Are they gas turbines?

MR. McKELVEY: Yes.

MR. RAY: Good.

Excuse me. One last question. Somebody said 60 megawatts. Are there four 60's?

MR. McKELVEY: 480 megawatt units.

MR. RAY: What do you need to start the plant -- to shut down the plant safely? Do you have any idea?

MR. McKELVEY: Yes. All we need is one division power.

MR. RAY: How many megawatts is that?

MR. McKELVEY: That is approximately five megawatts. So we are well within the rating of one.

MR. ZUDANS: Only one of those four is a black start.

MR. LUSIS: A small addition. If we start one unit we can pick up the other three.

MR. McKELVEY: But there is really no need to do that.

MR. KERR: Please continue, Mr. McKelvey.

MR. McKELVEY: Getting into a failure analysis, for long-term cooling we need one of our two divisions to remain intact power supply-wise, and the loss of either division, as you can see, will be backed up by the second division. Loss of either division will result in the loss of that division's emergency core cooling.

(Slide.)

Loss of division one will also result in the loss of all power to the inboard isolation valves. However, in both cases we have divisional -- division two redundant equipment which can act to mitigate the accident.

Incidentally, the ECCS is shared among the four buses. For example, one RHR pump and one core spray pump are fed from each of the four buses and its associated valves are fed from the 480 volt bus associated with that load group.

The emergency diesel generator auxiliaries are on their own load group. They are also fed from that power train. Okay.

(Slide.)

Now, moving into the DC system.

MR. KERR: Mr. McKelvey, I do not want to rush you very much, but your time allocation shows 20 minutes and my watch shows you have already spent most of that. So we want the information, but talk a little faster.

MR. McKELVEY: Will do.

The Fermi DC system consists of five batteries. They are of similar design. Three are at -- graded at 260 volts, two at 48 volts. All five batteries are center-tapped for control at the half-voltage. Three -- one 260-volt and the two 48-volt batteries run non-ESF equipment.

The overhead I have up here right now shows you a typical for either of the two divisions. The batteries are sized for four hours based on a worst case design basis event without battery chargers. The battery chargers were sized for being able to carry the continuous load while simultaneously recharging the battery within 16 hours.

In addition --

MR. ZUDANS: Can I ask a quick -- if you would not have batteries, just a charger by itself, could you use it?

MR. McKELVEY: Definitely.

MR. ZUDANS: How do you maintain the voltages?

MR. McKELVEY: The chargers have their own automatic voltage regulation.

MR. RAY: I presume you could carry such loads on the chargers indefinitely.

MR. McKELVEY: Yes, as long as the AC system remains intact.

MR. RAY: Yes, yes.

MR. McKELVEY: In addition, we have an installed spare battery charger which is able to be connected to either side of the battery to replace either of the two normally installed chargers, normally operating chargers.

Now, getting into a loss of DC, if we lost the division one 260-volt battery we would lose the function of the RCIC system, we would lose the ability to depressurize using the auto safety relief valves. We would also lose the control to division one ECCS, the division one switch gear and also the emergency diesels. However, the loss of this battery can be covered by the existence of the division two battery and any AC that is available in the division two side.

Similarly, for a loss of the division two battery, we would lose the function of the HPCI system, the outboard isolation valves that are DC-powered, the control of division two ECCS, the switch gear and the diesels. Similarly, again, it can be covered by the remaining division one battery and AC sources for the redundant loads.

MR. RAY: A quickie. You said should you lose one division DC supply -- maybe I am interpreting this too broadly -- you could back up those loads from the other division.

MR. McKELVEY: Right.

MR. RAY: This says you have ties between the two divisions.

MR. McKELVEY: No, not the same loads. Redundant loads.

MR. RAY: Okay. So therefore you do not have communication between the two DC systems.

MR. McKELVEY: No, they are fully independent.

MR. RAY: Thank you.

MR. ZUDANS: If you lose division number one, you lose RCIC and that is not recoverable by any redundant means.

MR. McKELVEY: The HPSI system along with the ADS --

MR. ZUDANS: But not the RCIC. If you lose the other division, you lose division two and you have RCIC.

MR. McKELVEY: Right, right.

MR. ZUDANS: Okay. One is ECCS and the other is not.

MR. KERR: Any time Mr. Zudans hesitates, you start talking.

(Laughter.)

MR. McKELVEY: I guess, moving to the last of the station blackout, we feel that there is no single equipment failure on our site that can get us into a loss of both offsite power sources. Beyond that we do not believe it is credible to lose both offsite power sources and also simultaneously lose four emergency diesel generators.

However, if such a highly unlikely event were posed, we could through the use of the black start peaker which was previously mentioned power one entire division. It is easily sized, as we mentioned before, to do so.

That was the conclusion of my presentation, unless you have some other questions.

MR. KERR: Are there other questions?

(No response.)

Mr. McKelvey, can you tell me if you have calculated what is the probability that you will be without offsite power for two hours?

MR. McKELVEY: Our system planning department has done such a calculation. However, it does not take into account all potential offsite power losses, such as a tornado. It does take into account storm-related ones, ice, wind.

MR. CARBON: Sabotage?

MR. McKELVEY: Sabotage was not included, either. But we came up with a 10^{-5}.

MR. CARBON: This is 10^{-5} for the life of the plant or per year?

MR. McKELVEY: Per year.

MR. KERR: You did not take into account tornadoes?

MR. McKELVEY: We did not.

MR. KERR: Is that because you did not know how or you did not think a tornado was likely to occur?

MR. McKELVEY: We did not exactly know how. We are working on that, by the way, and plan to revise it.

MR. KERR: If you cannot do anything else, you will have to gin up a tornado.

MR. McKELVEY: Right. There are contingencies, by the way, for replacing poles in a rapid as possible situation to restore one line to the site should a tornado go through our corridor.

MR. KERR: The 10^{-5} does take into account ice, I assume?

MR. McKELVEY: Yes.

MR. KERR: Any other questions? Mr. Carbon?

MR. CARBON: Is it impractical to design tower power lines against tornadoes or does it become just --

MR. McKELVEY: To the best of my knowledge, it is darn near impossible.

MR. CARBON: Okay.

MR. KERR: Max, there is a slight variation of that which no utility in its right mind would mention, but I will mention, and that is you put them on the ground. But --

(Laughter.)

MR. ZUDANS: And saying everything is possible

I have a question, a basic question. Why are the RCIC and the HPSI systems classified differently?

MR. McKELVEY: Classified differently?

MR. ZUDANS: In terms of safety grade.

MR. KERR: Mr. Colbert.

MR. COLBERT: Bill Colbert, technical director.

The HPSI is classified as class 1. It is class 1 backup, and the other division is actually the automatic ADS. The R-C-I-C is -- let's see, it is the reactor core isolation cooling, and it was not designed originally to recover from an emergency, an accident, but an operating emergency. Why, way back 20 years ago, 10 years ago, that decision was made, I do not know. But the class 1-ness of the other division is the ADS valves. So you operate the ADS valves and you go back to your --

MR. ZUDANS: You lose one of your electric systems, you lose a HPS with it, and you have no way of having high pressure injection. You have to depressurize. That is the normal procedure.

MR. COLBERT: You said if you lose HPSI. That is not true. You do have RCIC.

MR. ZUDANS: No, I think RCIC is not in the same category. It does not have to be there. It does not have to survive. In other words, it is a lower quality type of device and therefore in the case of some postulated situation it does not have to be there. It cannot be counted as a backup in your ECCS calculation.

MR. COLBERT: That is correct.

MR. ZUDANS: Therefore it is not there.

MR. KERR: But in real life it is there.

MR. ZUDANS: That is right.

Now -- and it is a very nice thing to have. I am wondering, what is the philosophy? Actually, not the utility, the NRC should tell us why they do not put this in the same --

MR. KERR: Tomorrow morning at 8:00 o'clock.

(Laughter.)

MR. KERR: Okay. Any other questions of Mr. McKelvey?

(No response.)

Thank you, sir.

MR. LEHNERT: My name is Dan Lehnert, system engineer, Detroit Edison. And the topic is the Mark I containment issue. And what I would like to do is briefly describe the nature of the containment, primary containment design at Fermi 2, the plant-unique program for resolving the Mark I containment issue, and identify for you the results of our implementation efforts.

(Slide.)

The primary containment at Fermi 2 is a steel shaped structure designated by GE to be a Mark I containment. This slide shows the primary containment in an elevated sectional diagram. As you see here, it being the drywell, it is an inverted lightbulb shape connected to a torus-shaped wetwell or suppression pool, and it is interconnected with the drywell in the suppression pool between the vent line, vent header, and the downcomers.

The original design -- the primary containment was designed, erected for the ASME section 3 during the early

1970's. Original design load inputted dead loads, design loads, pressure due to accidents, earthquakes and thermal. Of course, additional loads associated with SRV discharges and loss of coolant accidents have been identified.

The identification of these new loads led to the formation of the Mark I owners group, whose efforts led to the proper load definitions, structural acceptance criteria and application techniques. The Fermi 2 containment, Fermi 2 plant-unique program for the Mark I containment system provided an early and prompt reassessment of the Fermi 2 containment design for these suppression pool hydrodynamic conditions.

The results identified that extensive modifications would be required to restore the originally intended margin of safety in the containment design. Most of these modifications identified in the evaluations have been installed. As you can see by this slide here, it provides a scorecard, if you will, of the modifications that have been completed.

(Slide.)

And you will note by the slide the nature of modifications addresses all areas of the torus, being the torus support system and that system division four vent header deflectors, downcomer line, vent header intersection stiffening, modifications to other internal structures, such as the monorail and the catwalk, rerouting of the SRV piping, piping inside the torus, and providing the additional support capability for the safety relief valve discharge piping inside the torus, and also providing supports for piping attached to the torus routed inside the torus.

In addition, we have modifications currently identified as listed on -- in this table.

(Slide.)

And they include the addition of -- the completion of the torus support system, if you will, by adding mitered joint saddles and providing the hold-down system in the area of the mitered joint saddles, and also the columns and completing our safety relief valve discharge arrangement by adding the quenchers and the quencher supports and the -- completing the modification to the catwalk by installing -- installing grating.

As you notice by the dates here, the estimated completions are all prior to fuel load for all currently identified modifications.

To return to the previous slide briefly, you will notice that Edison in its containment modification program has proceeded at risk in that all of these modifications have been installed prior to the finalization of the generic program criteria.

(Slide.)

Ongoing evaluations of these designs as the generic loads and criteria were finalized, and also comparisons with modifications being installed at other Mark I facilities, have identified that the capacity of these modifications will very likely be sufficient to meet the long-term program acceptance criteria.

As such, we, Edison, anticipate that the long-term program plant analysis will be only a confirmatory type analysis. Edison believes that the Fermi 2 containment program has addressed NUREG-0661 and will lead to a timely resolution of the Mark I containment issue, and with the current -- with the currently installed modifications and identified modifications yet to be installed, we believe that the plant will function safely in the event of all possible safety relief valve transients and loss of coolant accidents.

Any questions?

MR. KERR: Thank you, Mr. Lehnert.

Mr. Zudans?

MR. ZUDANS: Yes. I -- when I was on site visit, I tried to see the braces for the sacrificial shield against the shell, and the reason for wanting to see that was I wanted to see how the truss is connected to the containment shell, because it is a free-standing steel shell and the question of loading reduction was in my mind.

Is there anybody who can tell me how it is done? Because I do not see that in the analysis.

MR. KERR: Do you understand the question, Mr. Lehnert?

MR. LEHNERT: I believe I understand the question. I believe the individual, Mr. Wally Street, could assist in responding to that question.

MR. KERR: Mr. Street, can you help us?

MR. STREET: Well, my name is Wally Street. I am a supervising structural engineer for Detroit Edison.

Okay. The star truss at the top is connected to the shell. There is is a large thick plate in that area and it is attached to that and almost right to the shell, to a connection in the biological shield. The attachment point through that interface into the biological shield has a tongue and groove joint, so that the shell is allowed to deflect axially away from the centerline of the reactor and also particularly with no restraint. And it only takes loads tangentially to the shell at that point.

MR. ZUDANS: Axially with respect to anchor, is that what you mean?

MR. STREET: It can move radially and it can move vertically with no restraint. It will take the load tangentially.

MR. ZUDANS: There is a slot there?

MR. STREET: Yes, there is a slot, a large plate embedded in the concrete.

MR. ZUDANS: That is what I wondered. I thought you might have a girder around there.

MR. STREET: We use the biological shield as a girder there.

MR. ZUDANS: Okay. Thank you.

MR. KERR: Other questions? Mr. Catton?

MR. CARBON: On the downcomers, do you tie the tips together?

MR. LEHNERT: The original design had the downcomers tied together. They were identified as really just shipping braces, and our modification called for the removal of those ties and a more -- the critical location as far as the downcomers and vent headers was the downcomer-vent header intersection.

We provided a crotch plate between the downcomers and shoulder plates, between the downcomer header and the downcomer arms, to provide the additional stiffening.

MR. CARBON: Okay.

MR. KERR: Any further questions?

(No response.)

Does the NRC staff have any comments?

MR. KINTNER: We have some brief comments prepared. Mr. Byron Siegel can tell you how the plant-unique analysis will be reviewed and about the piping.

MR. KERR: It has not yet been reviewed?

MR. KINTNER: No, it has not yet been submitted, the final analysis.

MR. KERR: Okay. I guess -- do you two structural types -- if I could get your attention for just a minute, do you two gentlemen want to know how the staff plans to review the structural -- the plant unique facets of this, or would you rather wait and see the review when it exists?

MR. ZUDANS: I did not -- staff had not reviewed it.

MR. KERR: They have not reviewed it. I am not sure if they have received it. Have you received it?

MR. KINTNER: We have received the interim analysis and we have heard presentations by Detroit Edison recently in the recent review, the last month or two, on the modifications. So we are familiar with the modifications, but we have not -- do not plan to complete our review until the plant-unique analysis is in, and then we will review it.

MR. ZUDANS: We are talking about Mark I.

MR. KERR: Plant-unique aspects of the Mark I I believe is all we are talking about.

MR. ZUDANS: I saw the way it is modified and I just cannot imagine anything that could pull this apart now.

(Laughter.)

MR. KERR: Then I do not think we need to listen to how they are going to review it.

MR. ZUDANS: No.

MR. KERR: I am sorry if we brought you here under false pretenses, but we thank you. Thank you, Mr. Lehnert.

Mark I containment environmental qualifications of equipment. We I think are more interested in whether you foresee any serious problems in your program, rather than a part by part description of how you are going to do it.

MR. DUONG: My name is Kwang Duong. I am a senior electrical engineer. I am here today to present to you a brief description of our environmental qualification program and its present status in response to the NUREG-0588 requirement.

The Detroit Edison Company has organized a task force to review the environmental qualification of safety-related electrical equipment. The equipment over in that review includes only safety-related electrical equipment that are required to mitigate a postulated accident, the installed lessons learned equipment, display instrumentation mentioned in the emergency operating procedures associated with the safety-related equipment with both accident sampling and monitoring and radiation monitoring.

(Slide.)

The equipment that is associated with one preferred path to bring the plant to and maintain it at a cold shutdown condition, the class 1E equipment located outside the Fermi containment which is affected by high radiation during the post-LOCA recirculation or containment fluids.

The review concentrates on the equipment exposed to harsh environments and harsh environment is defined as LOCA, main steam line break inside a primary containment, and high energy line break for outside of containment. Since Fermi 2 CP SER was issued prior to July 1 of 1974, the review was based on the guideline of the NUREG-0588 category 2.

(Slide.)

Three organizations were involved in the task force. Nutech Incorporated as the prime contractor was responsible for the NSSS portion. Wyle Lab as a subcontractor was responsible for the non-NSSS portion. And the Detroit Edison Company, with the overall administration, plant operation and emergency procedures.

The program consist of four parts: First is the development of the master equipment list. This is done by reviewing the FSAR technical specification, emergency operating procedures, electrical single line diagrams and schematic diagrams, and then the retrieval of the qualification records.

Test reports were retrieved from General Electric, Detroit Edison government control center, from Wyle Lab, from BWR Owners Group, EPRI equipment qualification data bank, and vendor files. And test data were evaluated against NUREG-0588, category 2 requirements and Fermi 2 environmental conditions.

And lastly, the preparation of the qualification summary sheet. The sheet contains the information on the equipment, for example the system or subsystem that the equipment belongs to, the location of the equipment, safety function of the equipment, et cetera. It also contains the environmental conditions normal and accident that the equipment would be exposed to, the demonstrated condition.

The program was under an improved QA program from Wyle and Nutech. Summary sheets will be updated as required, and environmental qualifications central file will be periodically audited.

MR. CATTON: As part of your environment, do you attempt to to ac_ unt for steam flow-induced vibrations?

MR. DUONG: I guess that is a separate issue with the seismic requirement.

MR. CATTON: No, I am talking about steam flow-induced vibrations.

MR. DUONG: I do not think so.

MR. CATTON: Do you think it is important?

MR. DUONG: I do not know how to answer that question.

MR. KERR: Mr. Colbert, do you want to respond? Do you understand the question?

MR. COLBERT: I think I do.

MR. CATTON: I can repeat it.

MR. COLBERT: The question as I understand it is, is the environment including vibration caused by steam flow from leakage? Is that what you are looking for?

MR. CATTON: Your environment supposedly, is that due to a steam line break? If you have a steam line break you have loss of flow, and I am not talking about jet impingement. I am talking about just flow of steam, and that is going to cause a lot of vibration, as well as heating.

MR. COLBERT: The answer that I have to give you is I do not know. Not to my knowledge. It would include that.

MR. LEHNERT: Dan Lehnert, system engineer.

In the pipe break evaluation outside of

containment, the relative volumes of the containment outside the break result in very little mass fluxes for the mass flow of the steam to the compartments. So I guess, to answer your question specifically, no, we did not evaluate for mass flow over the instrumentation as far as potential vibration. I would be surprised if it resulted in any appreciable effect.

MR. CATTON: Your steam line is at pretty high pressure, so you have a pretty good flow out of that line and you cannot redistribute the flow across the total volume. I think you have to do some sort of a calculation.

What does the staff think about that?

MR. KINTNER: The seismic qualification review team includes vibration from operation as one of the parameters it is looking at. You are talking about a steam line break. I do not know if they include accidents, but they do include vibration from operation.

MR. CATTON: But you do not include vibration induced by steam flow.

MR. KINTNER: From the accident, a steam line break, I am not sure about that. But from normal equipment operation, abnormal operation, valves closing, that sort of thing, that sort of vibration is included.

MR. CATTON: There are examples of what steam flow vibrations can do. One of them is the HGR containment in

Germany. I think it would be quite devastating. It tears insulation off walls, it throws things around.

I am just kind of curious as to why this has never been a consideration.

MR. KERR: That is the next round.

MR. COLBERT: Mr. Catton, did you just come back to jet impingement now?

MR. CATTON: No, no, no. Jet impingement is fairly straightforward. As soon as you decide what the break is going to be, you can estimate where the flow out of the break is going to hit and you can protect it. Wherever that flow hits, that flow turns. You might have something that is several feet along the wall. Now you protect the thing where the jet impinges, but you forget about the thing that is around the corner, that is also impinged on by the flow at still relatively high velocity.

MR. COLBERT: One other clarification. What steam line are you breaking?

MR. CATTON: Whichever one you broke.

MR. COLBERT: All right, whichever one we broke. The answer Dan Lehnert gave you was rational. When you talk about steam lines of larger size, you are talking about the main steam line, which in our plant is included inside a separate containment. And the flow from there is directed to the turbine building.

In the turbine building we do not have instrumentation which -- or equipment that we call upon in the emergency core cooling systems. So therefore it was not reviewed on that basis.

Steam lines inside the containment, the largest of which is the HPSI steam line, have been specially designed where the line is only pressurized through a one-inch line. So I do not think, at least as I understand your question, that it is applicable.

MR. CATTON: If it is only one-inch lines, you are probably right. Does somebody walk through to make sure when the one-inch line breaks that the flow out of that one-inch line cannot be deflected into something else, or is it just the line of sight? Do they actually do a walk-through?

MR. LEHNERT: I am sorry, would you repeat the question?

MR. CATTON: Sort of like a garden hose on a screen. Do you go through and check to make sure that the deflection does not happen?

MR. LEHNERT: As far as overall equipment, as far as the steam tunnel, there is no equipment inside the steam tunnel that we need to assure safe shutdown of the plant. In terms of outside the steam tunnel, as Mr. Colbert annunciated, there is the HPSI steam line and that is only a warmup line through a one-inch line, so that the mass energy release is rather small.

The other lines that we considered as high energy are the reactor water cleanup line and also the RCIC steam line. And the reactor water cleanup line as I recall is a six-inch line. The RCIC steam line is, I believe, a six-inch line also. Those lines for the postulated break regions discharge into rather large volume areas, and one -- again I reiterate that the mass flux due to -- in those large volumes is kind of like rather small.

MR. CATTON: You cannot spread it out so quickly. Nature does not do that.

MR. LEHNERT: We did evaluate all equipment in the proximity of the area of the break, and regardless of whether it was directly impinged or not our first line of approach in the analysis was to assume it was unavailable and then look for other equipment to achieve the same function.

MR. CATTON: That is fair enough.

MR. KERR: Any other questions?

(No response.)

Does that conclude your presentation?

MR. DUONG: I have a couple more slides.

MR. KERR: Okay.

(Slide.)

MR. DUONG: As a result of the evaluation, the deficiencies are identified on the summary sheet and corrective actions will be taken. Action could be one of the following: analysis to assure that the listed device need not function for mitigation of the accident, to perform an analysis to assure that the device takes this action prior to failure, to relocate or shield the device to provide an acceptable environment, specify a changeout program if the qualified life of the item is shorter than 40 years, perform some additional tests or analysis, some test program or replacement program.

Some of the current activities of Detroit Edison in resolving the deficiencies. We are with the BWR Owners Group in joint qualification of common items. These are items that are being used by three or more utilities. We have contacted several AE's as well as some test labs to discuss the resolution of the plant-specific items. Our goal is to qualify all the 1E items by July of 1982, which is a very tough job.

The response to NUREG-0588 was sent in to the NRC staff on June 26 of this year, and in the week of 7-13 through 7-17 the NRC staff performed an audit. Some of the findings include that the procedures to review the documentation are in place and the requirements of the equipment qualification are understood.

However, there are some problem areas that could be improved. And immediately after that audit finding we called a meeting to establish our action plan, and with this we will review the equipment qualification program for completeness, we will review the submittals and supporting documents for consistency. We have an action plan to achieve full qualification. We will conduct audit on a central file, and then we will resubmit to the NRC staff our response by December 1, 1981.

MR. KERR: Thank you, sir.

MR. DUONG: This concludes my presentation.

MR. KERR: Are there questions?

(No response.)

MR. KERR: I declare a ten-minute break.

(Recess.)

MR. KERR: I am going to interchange two of the items on the agenda if I may, and go to hydrogen control at this point with the comments on ATWS coming directly after that. Is that okay? Is Mr. Green -- .

MR. COLBERT: Mr. Green?

MR. GREEN: Gentlemen, I am John Green, Systems Engineer, Fermi 2 Project and I will be presenting the hydrogen control at the Fermi 2 plant. It is going to be a brief description of four systems we use for the control of hydrogen.

(Slide.)

These systems include the nitrogen inerting system, a hydrogen/oxygen monitoring system, thermal recombiners and a purge system. The inerting system basically avoids a combustible gas mixture by maintaining the primary containment at less than 4% oxygen level. The hydrogen/oxygen monitoring and thermal recombiners are used to detect and control post-LOCA hydrogen, and the purge system is a backup to hydrogen control.

(Slide.)

This drawing shows you a basic drawing of our hydrogen inerting system. The system provides nitrogen to enter the primary containment, nitrogen makeup for the primary containment, pneumatic supply to loads in the containment, and plant nitrogen needs that we have elsewhere.

The system is a BOP system; it is exclusive of penetrations of isolation valves which are Class 2. The system isolates on high dry well pressure, low reactor water level, or high radiation or reactor building exhaust. Basically, what is on this drawing is the two systems; a large inerting system which is 20 and 24-inch valves, and a small makeup system which is an inch and a half.

Your inerting system supplies you through the torus toward the dry well and then to the other side of the other torus. This system is 20 and 24 inches, and it is a high flow, low pressure system that is fed through a steam vaporizer. The vent makeup system is a low blow, high pressure system that is fed through an electric heat exchanger and it feed the dry well and one side of our torus. This system is to accommodate minor fluctuations in pressure control in the dry well.

MR. ZUDANS: What is the pressure increase in this process?

MR. GREEN: Decrease?

MR. ZUDANS: Increase. You pump nitrogen in; what is the delta p?

MR. GREEN: As we inert?

MR. ZUDANS: Yes.

MR. GREEN: This drawing only shows hydrogen inerting. Tied in with it is a purge system, and I tried to separate the two so you can see them. But as the flow supplies the nitrogen, you have an exhaust to the purge system so you are always pumping in and drawing out at a similar time and there is not a very -- a significant change of pressure in the system.

MR. KERR: You are inerted while you are operating, aren't you?

MR. GREEN: When you start up you are not inerted and you have to inert.

MR. KERR: Yes?

MR. GREEN: During operation we will be inerted, yes.

MR. KERR: You said something about something being triggered by high dry well pressure or some -- .

MR. GREEN: These are containment isolation valves.

MR. KERR: All right, okay. I missed that. You were confusing me because I thought you were indicating a system which would inert only if you had an accident, and I did not think that's what you did.

MR. GREEN: No. It is general operation. We run the plant inerted.

MR. KERR: All right.

(Slide.)

MR. GREEN: This is a slide of a hydrogen/oxygen monitoring system. Mr. Wooden already touched on some of the characteristics of this system. Physically, what is shown here is one of the two divisions which we have. The system is engineering safety feature seismically and environmentally qualified, and it takes samples from multiple ports in the dry well or the containment selected at choice; you can sample one or all or any combination of the samples.

MR. MOELLER: How many hydrogen analyzers do you have on that system?

MR. GREEN: Well, each draw point is a suction to our analyzer, so each point goes into the analyzer; in that analyzer is a hydrogen analyzer and oxygen analyzer, so effectively it is eight points.

MR. MOELLER: What happens if the analyzer breaks down?

MR. GREEN: Can I direct that to Larry Wooden?

MR. MOELLER: Is the analyzer in duplicate?

MR. GREEN: Yes. Not within the system. Within this system you have -- this is Division 1. Not shown in this drawing is a Division 2 system. They are redundant and independent.

MR. MOELLER: So if one went out -- .

MR. GREEN: You have the other system, yes.

MR. MOELLER: And what procedure do you use to make sure no one ever enters an inerted containment?

MR. GREEN: Do we have a procedure? Our response at this time was that we do not enter the containment when it is inerted.

(Laughter.)

MR. MOELLER: Right.

MR. GREEN: That is our plant philosophy. I mean -- .

MR. MOELLER: Say I walk up and I want to enter?

MR. KERR: I think Mr. Griffing wants to answer.

MR. GRIFFING: Mr. Giffing, Plant Superintendent. We will have a written procedure which covers entry into the containment following shutdown. We will include in that sampling of the atmosphere to assure that people that enter will have proper oxygen.

MR. KERR: What about people who cannot read?

MR. GRIFFING: They will -- in the first place, it is a locked space and we have administrative controls and operators are trained and operate under orders, too.

MR. KERR: So the man who enters, or woman, has to have a key or somebody has to have the key, and there are procedures for using the key.

MR. GRIFFING: Yes.

MR. KERR: Okay. Thank you, sir. Other questions?

(No response.)

Please continue, sir.

MR. MOELLER: One quick item. Another plant we reviewed one time, they did n^t have to breathe removal or catching units inside the purged lines to protect the isolation valves.

MR. GREEN: We are installing a debris screen. That won't show in the drawing but we do have them.

(Slide.)

The next system is a thermal recombiner system. It provides capability for post-LOCA control of combustible gas mixtures. Once again, the systems are independent and redundant.

What you can see is we have two divisions of the systems; either one is capable of taking a suction for the dry well or the torus, having the reaction in the chamber and returning the gases or water to the torus. These systems are installed on our reactor building and they use dedicated penetrations. They function to limit the combustible gas in a post-LOCA condition, to less than 5% per Reg Guide-1.7. And the equipment to start these systems is located in our main control room. Dr. Kerr?

MR. KERR: There are questions? Thank you, Mr. Green.

MR. GREEN: I have a purge system. I'm going as fast as I can.

(Laughter.)

(Slide.)

This is our purge system. This is used to remove containment atmosphere and provide backup hydrogen control. It, once again, is a BOP system, exclusive of the penetrations and isolation valves. And they have diverse isolation on high radiation reactor building exhaust, high dry well pressure or low reactor water level.

The purge system is used to de-inert containment primarily, and it is a large system of 24 and 28-inch valves. You have provision for flow into -- out of the dry well, and you take air in from the building, and out of the dry well or out of the -- here is the exhaust, excuse me. This is exhaust out of the torus or exhaust out of the dry well to the standby gas treatment system.

We also have a vent system for pressure control, and this is a one and a half inch system that is used on line. The purge system itself is limited to 90 hours operation for conditions other than cold shutdown and refueling per the branch technical paper.

So in summary, we have hydrogen control systems that basically include inerting for avoidance and recombiners for post-LOCA operation.

MR. KERR: Mr. Green, if you needed to get into the containment system in a hurry, how rapidly can you purge to the extent that you can go into the containment?

MR. GREEN: The purging through the standy gas treatment -- are you speaking of --?

MR. KERR: You may remember, it seems to me, there was a leakage inside containment within the last several months. I remember it took 12 hours or so before people could enter this particular containment.

MR. GREEN: We have not performed it yet, but the procedure right now, and timed to our calculations, is less than six hours.

MR. KERR: Thank you. Any other questions? We turn, then, to ATWS, and contrary to what the agenda says, what I was interested in was the training that is being used by Edison operators to deal with a potential ATWS event. I think Mr. Giffing is going to talk about that.

MR. GIFFING: Mr. Giffing, Plant Superintendent, Fermi Unit 2. We have an ATWS procedure which I have an example of right here, covering reactivity control.

(Slide.)

It has been evaluated by the Commission and comments have been fed back to the utility, and we have reacted to that and are training on this procedure at this time.

First, I would like to say that there is an alternative means to shut down the reactor in the event of an anticipated transient without scram, and that is, to use our standby liquid control system. Operator action, however, is required.

I mentioned that we have an emergency procedure which is based on symptoms which provides a direction to control reactivity in the event the scram is called for but does not occur.

And thirdly, here, we have a highly trained core group of licensed operators and line supervision, who have been trained to use these emergency procedures. Our emegency procedure evaluation by the NRC is going to commence on the midnight shift this evening at Chattanooga.

Fourthly, we have directive, a nuclear operations directive, by our Vice President for Nuclear Operations to the nuclear shift supervisor telling him to operate with safety foremost, as opposed to operating with economic considerations or some other considerations ahead of that safety consideration. Safety is foremost.

And finally, we will use these procedures, including standby control if necessary, to control an ATWS if it should occur. Questions?

MR. KERR: Are there questions? Is your simulator to be equipped to deal with the ATWS training in some fashion, or have you decided?

MR. GIFFING: We will have the ability to test the ATWS procedures on our simulator, and we can also test it on the Browns Ferry simulator.

MR. KERR: Thank you. Are there other questions?

MR. RAY: I assume your standby liquid control is not automated actuation.

MR. GIFFING: No, it is not. It does require operator action.

MR. KERR: How long does it take the reactor to get to hot shutdown after you start injecting the standby liquid control system? Do you know?

MR. GIFFING: It would be a fairly short period of time; something on the order of ten minutes; six to ten minutes. The immediate result from injection is going to turn your power level, even if you were at 100% power. So the reactor would effectively be shut down.

But until you drive the power all the way down to the source, you are not going to be at hot shutdown.

MR. KERR: Sure. Other questions, comments?

(No response.)

Thank you, Mr. Griffing. We now have a presentation on the SDV modifications. Are you going to make that, Mr. Page?

MR. PAGE: Yes. I am Earl Page, Nuclear Safety Engineer, Detroit Edison. My comments will deal -- they will be brief and will deal primarily with Edison's esponse thus far to the recent concerns raised by Michaelson in regard to the scram discharge pipebreak. The NRC staff has issued their report on safety concerns associated with pipebreaks in the BWR scram system in the form of a draft NUREG-0785. And attached to that NUREG was a request for a generic and a plant-specific response in 45 and 120 days, respectively.

Edison has already filed the generic response, and in so doing, referenced the General Electric report, NEDO 24342. We expect the generic safety evaluation report from the staff on the scram discharge volume pipebreak to be issued sometime in August.

Now, I just learned that accompanying the issuance of that NUREG, we will get a site reprieve in the length of time we have to respond with our plant-specific application. So we do have 120 days, but it will be counted from the date of issuance of that SER. And during that time, Detroit Edison will review the Fermi-2 design for compliance with the criteria contained in the SER.

That is all I have on the status of our response thus far. This is a fairly new issue.

MR. KERR: Are there questions or comments? How does your design differ from, or is it at all like, Browns Ferry?

MR. PAGE: It is different -- allow me to salvage a portion of my earlier presentation here. In the scram discharge volume area, this is not designed to describe it with regard to the subject at hand, but it will give you the general difference in the configuration.

(Slide.)

Browns Ferry has one instrumentation volume down here and, of course, two scram discharge volumes and the passage from the scram discharge volume to the instrument volume is a very s.all two-inch line.

The Fermi 2 design, which is also similar to most of the newer BWR-4's and 5's and 6's, is characterized by this really simple drawing.

(Slide.)

There are two separate instrument tanks to go one with each scram discharge volume. So they are more closely coupled, first of all, in space and then there is no small two-inch line. This does not show too well. This is part of the scram discharge volume in the real sense. This is an 8-inch pipe here welded directly to the instrument tank, so the main difference is the close hydraulic coupling between both scrame discharge volumes, and two respective or associated instrument volumes.

John, is there anything else?

MR. ZUDANS: And the instrument volumes individually discharge in the drain tank?

MR. KERR: Did you understand the question?

MR. PAGE: No.

MR. ZUDANS: You show that those instrument volumes that connect to the two-inch pipe, connects to each of them, and each of these pipes inlet to a drain line separately; they do not join together.

MR. PAGE: I think they are joined. John?

MR. ZUDANS: At least that's what we were told.

MR. GREEN: The drain lines of each instrument volume will go to redundant drain valves, so you'll have two valves for isolation, and from there they will drain to an equipment drain tank, and it will be one equipment drain tank, not separate equipment drain tanks.

MR. ZUDANS: They do not join before they get into the tank, which is the case in -- .

MR. GREEN: Can I draw you a real quick -- ?

MR. KERR: Why don't you do it on the blackboard so we can see it, too. Is there chalk over there?

MR. GREEN: This is your instrument volumes, and you have a drain line off each one, and they will go through drain valves and then to your equipment drain tank.

MR. ZUDANS: They do join.

MR. GREEN: They do join, yes.

MR. ZUDANS: That is all right. I asked in the plant and they said no, they don't; each one goes to a separate tank.

MR. KERR: Mr. Colbert?

MR. COLBERT: The point, as I remember, was -- there was under discussion at the plant the fact that each of those scram discharge volumes reside one on each side of the reactor building in that the reactor building drain system was divisionalized, and the water from one side we expect to find in one series of drains; sumps on the one side and for the other on the opposite side.

MR. ZUDANS: How could you? They all connect at the same point.

MR. COLBERT: The question was a break in the scram discharge volume. That was the point we were discussing then.

MR. ZUDANS: Oh, that is okay. No need to spend anymore time on it.

MR. KERR: Thank you. Does that conclude your presentation?

MR. PAGE: Yes.

MR. KERR: Thank you, Mr. Page. Mr. Colbert, the agenda shows any TMI issues which have not been discussed which should be discussed by you at this time.

MR. COLBERT: Mr. Kerr, the items that we remain are either those which we totally understand or -- everything left has either been discussed or we understand the staff's position and are reacting to it.

I would like to take this opportunity to get a couple of items in to the record where we either lacked information or a question was asked earlier.

MR. KERR: Please do.

MR. COLBERT: I have two people I want to have address; one is Larry Sherman, our licensing engineer on a couple of points made earlier today.

MR. SHERMAN: I am Larry Sherman. I would like to respond to a question earlier this morning raised by Mr. Ray regarding what de-rating of Fermi 1 would occur if only one cooling tower was in service in the worst weather conditions.

The worst weather conditions in this case would occur during the summertime. With only one tower in service and a 74^{o} F wet bulb temperature, the main turbine condenser backpressure would increase from 1.5 inches of mercury absolute to 4 inches of mercury absolute; resulting in about a 33% reduction in the unit capability, or approximately 650 to 700 megawatts electrical.

Also, I would like to correct a statement that I made earlier on the core thermal power. I will correct that statement. The 3292 megawatt thermal is the 100% rated output of the reactor for which Detroit Edison is requesting a license.

MR. KERR: 32?

MR. SHERMAN: 3292.

MR. KERR: Thank you.

MR. SHERMAN: Reference was made to a number, 3430 megawatt thermal. This corresponds to a stretch capacity, if you will, or a capacity which we believe that the reactor could produce. And it is also equal to the capacity of our emergency core cooling system.

MR. KERR: But the request for license is for licensing at the 3292.

MR. COLBERT: That is correct.

MR. KERR: Thank you, sir.

MR. COLBERT: In response to another question, I would like to have Wally Street address the structural question.

MR. STREET: Yes, I am Wally Street, Supervising Structural Engineer. The question was on the actual loads applied to the pedestal in the Sargent Lundy analysis. We checked back and found that for that particular element of reactor pedestal, that the loads included on it were the load of the reactor and walls and everything above it.

However, the vertical component of those loads above was not included in that analysis. It was felt that the predominant loads in that one were the horizontal loads in the actual vertical component for the reactor and the wall load was left off.

For the sacrificial shield wall itself, all loads were considered.

MR. KERR: I am just trying to picture a situation in which a reactor vessel produces a horizontal load, but I guess that is when it is going li.e this.

MR. STREET: It is seismic analysis, that's what it is.

MR. ZUDANS: I think they could not very well make a mistake on that, because they used previously analyses. I would suggest, though, that you go back and check. There may be other loopholes in it, you know.

MR. STREET: Yes, we have already started doing that check right now.

MR. ZUDANS: All right.

MR. COLBERT: That completes the information we have.

MR. KERR: Thank you, Mr. Colbert. Which brings us to emergency planning. Mr. Mattson? You could have fooled me.

(Laughter.)

MS. MADSEN: I am Ellen Madsen and I'm Environmental Licensing Engineer for the Fermi 2 Project. And I would like to give you just a brief rundown this afternoon on where we stand on emergency planning.

I think to start out with I would like to explain a little further a couple of questions that were raised this morning concerning our emergency planning zone for Fermi 2.

(Slide.)

This is a ten-mile radius arou... the Fermi 2 plant -- this is a ten-mile planning zone around the Fermi 2 plant. It is includes a portion of Monroe County here and a portion of Wayne County which is north of the Huron River. This is approximately 20% of the ten miles, as far as population is concerned, and this is approximately 80%.

MR. ZUDANS: What is in the blank circle?

MR. KERR: The blank half-circle is what he meant, I think.

MR. ZUDANS: The blank half circle.

MS. MADSEN: That is Lake Erie.

MR. ZUDANS: What is that? You talk about those. There is nothing?

MS. MADSEN: Lake Erie.

MR. CARBON: There is a chunk of Canada there, isn't there?

MS. MADSEN: Let's go on to the next slide, which is the 50-mile planning zone for Fermi. This is the five-mile ring here, and this is the ten-mile ring. Now, the border between Michigan and Ohio lies here (indicating). The border between Michigan and Canada lies here. It barely touches Canadian -- at this point, that is a very low, low population density area, basically farm area.

Right now, the Michigan state plan is set up such that the state police emergency services director will contact the people in Ontario should there be an emergency at Fermi, and the United States Coast Guard has agreements with the Canadian Coast Guard should there have to be an evacuation in the Canadian waters. So that is taken care of.

MR. ZUDANS: Where is Detroit on this map?

MS. MADSEN: Detroit is located right here (indicating). Detroit is here, and Windsor is here.

MR. ZUDANS: Thank you.

(Slide.)

MR. CARBON: Question. Do you have any interaction with the Detroit officials on emergency planning?

MS. MADSEN: The present status is the state police, state of Michigan, who head up the Disaster Services Division, who carry out all of the emergency preparedness for the state will be working with people in Wayne County to draw up the plans for the Wayne County area that is within our ten-mile jurisdiction.

MR. CARBON: Just within ten miles?

MS. MADSEN: Just within ten miles, that is right. And that does not include any part of the city of Detroit.

MR. CARBON: Thank you.

MS. MADSEN: Right now, the schedule for the Fermi

2 emergency plan is a state of Michigan plan and was submitted to FEMA Region 5 in February 1981. We submitted our plan to the NRC on March 31, 1981. We have comments from the NRC; we have discussed those, and our responses will be in to them on August 30. FEMA Region 5 will be submitting their first findings on the state plan to FEMA headquarters this month, and the Michigan state police intend to submit to FEMA for approval the state plan with the two county jurisdictions in November of this year.

All this will lead up to a tentative full-scale exercise for the Fermi plant and the NRC appraisal program run concurrently in February of 1982. At that time, we will be submitting our procedures and a revised plan.

(Slide.)

I will give you a very, very brief description of our emergency response facilities. This is the basic outline of the Fermi site. The Fermi-2 security fence, the Fermi-2 buildings enclosed therein. The control room is located in the corner right up here; this is the operational support center; this is the technical support center, and over here, housed in the nuclear operations center which is the building which will house the support personnel that Dr. Jens will be bringing down to the plant to work with everyone onsite, and in the basement of that will be located the emergency operations facility.

That is approximately three-quarters of a mile from the technical support center and the control room.

(Slide.)

The technical support center inside the perimeter of the security fence is located on the first floor of an office services building located in the corner here. The OSC is here, the control room is located in the auxiliary building. The distance between the TSC and the control room is approximately a four-minute walk, entirely inside through this pathway through the turbine housing and into the control room.

MR. KERR: One can go into the turbine house when the reactor is in operation, or does it have to be shut down?

MS. MADSEN: No, one can go into the turbine building during operation.

MR. KERR: Thank you.

(Slide.)

MS. MADSEN: The technical support center itself is designed to handle approximately 25 people. The Detroit Edison people, the support personnel and the NRC. It will basically be an open configuration, the heart of which will be a computer-based information system, and we will be able to move the CRT's and the personnel in any configuration that we wish to within this space in order to more effectively handle any emergencies.

Part of that emergency information system, and a permanent part of that system, will be a closed-circuit TV system whereby the cameras will be located in the control room in the position shown here.

(Slide.)

This will also be our interim facility.

MR. MOELLER: The cameras, then, are remotely controlled so you can look at whatever you want to see?

MS. MADSEN: I believe so. Mr. Lusis?

MR. LUSIS: Ed Lusis. Yes, the cameras will be remotely controlled, remotely zoomed. They will be high resolution color cameras.

MR. MOELLER: So you can read anything on the camera that you could if you were in the room?

MR. LUSIS: Yes.

MR. KERR: Please continue.

MS. MADSEN: The emergency operations facility, as noted before, is about three-quarters of a mile away from the other two facilities. It is located on the site and in the basement of the nuclear operations center.

(Slide.)

Here we can handle approximately 40 people, and do all of the necessary offsite communications, offsite dose analysis, dose assessment work and recommendation for protective actions.

(Slide.)

At the alert level, our emergency organization will begin staffing up in the technical support center. The emergency operations facility will then be brought into activation, should we reach a site emergency or a general emergency.

Now, this is just an indication of the types of individuals that we will have a basic structure in our organization when all three facilities are in operation. And, of course, to support this, we have a basic communications systems that will allow us to communicate both onsite and offsite.

(Slide.)

There is communication between all three support centers, or four support centers; all three major support centers, and we will have direct communications with the NRC, the state police and the county sheriff's office. These are individual offsite organizations that we will make contact with further up the line.

Here, we have contacts with our joint publication information center, which is offsite. We will have contact with our corporate headquarters, and our Wayne Division headquarters. And this is the basic initial communications tree and contact.

MR. KERR: Are those lines representative of telephone circuits, or does one have alternate modes of communication if telephone circuits, for some reason, are unavailable?

MS. MADSEN: Right now, they are representative of telephone circuitry. However, we are working on the area of communications, trying now tc determine what we will put in in the future.

MR. KERR: Thank you.

(Slide.)

MR. MOELLER: Have you begun installation of your alerting system?

MS. MADSEN: No, we have not. We have studied under way to determine what will be done as far as our prompt notification system is concerned.

MR. MOELLER: Is Lake Erie used as a drinking water source near the plant?

MS. MADSEN: The Monroe water intake is -- the city of Monroe drinking water intake is located about 2500 feet to the south of Stony Point out here.

MR. MOELLER: Do they have any raw water storage at the treatment plant.

MR. MOELLER: What would be the procedure in case you had, through some unexplained accident, a major release of radioactive material into the lake and it moved toward their drinking water intake?

MR. MADSEN: We do have in operation right now a continuous monitoring system for water, which is sampled on a monthly basis. We have not as yet --

MR. MOELLER: Wait now. That is at their intake?

MR. MADSEN: At their intake, that's correct. There is an eight mile run between their pumphouse here in Frenchtown and the water treatment plant in the city of Monroe.

MR. MOELLER: So you continuously take a sample, you look at it once a month?

MR. MADSEN: Yes, right here at the pumphouse. We have not as yet worked out detailed procedures with them.

MR. MOELLER: If you had an emergency -- well, let me ask the staff this. If a DOE aerial monitoring team came in, would they be free, or an NRC or EPA, some monitoring team came in, are they free to fly over Canada and do monitoring there as well as over the U.S.? How does that work?

MR. KERR: Does the staff know the answer to this?

MR. KINTNER: Yes. We have some people here from the emergency preparedness. Would you repeat your question again, please.

MR. MOELLER: Mainly I am thinking, if you had an airborne release following an accident and you had some aerial monitoring teams under various federal agencies there, in helicopters or whatever they are, can they fly -- and your prevailing wind is northwest -- can they fly -- or northeast, excuse me.

MR. DIEFAYETTE: I am Robert Diefayette from the staff. I have to turn this over to Bob Jaske from FEMA, since this international planning has now been turned over to FEMA.

MR. KERR: Every time I fly from Washington to

Detroit they fly over Canada. Whether they get special permission or not, I do not know.

MR. JASKE: Robert Jaske, acting director, radiological emergency preparedness division, FEMA.

In answer to your question, the provisions for emergency operations for transport vehicles come under the 1967 emergency services agreement between the two countries, and that type of action can be taken immediately under the existing protocols.

MR. MOELLER: Thank you.

MR. KERR: Other questions?

(No response.)

Please continue.

MR. MADSEN: That is the end of my presentation.

MR. KERR: Any additional questions of Ms. Madsen?

(No response.)

Thank you, ma'am.

This brings us to a point at which we are ready for the closed session on security. But I think before we go into closed session I will ask the Licensee Applicant if you have any further comments that you want to make on anything at this point?

MR. COLBERT: Give me one moment.

(Pause.)

MR. KERR: Mr. Moeller?

MR. MOELLER: Earlier this morning I asked the staff if they could give me some data that supports the estimates in the SER's which show that LaSalle will release routinely far more airborne radioactive material than will Fermi. And I wonder if they had some --

MR. KERR: I had planned to ask Mr. Kintner the same question in the intervening time, in which he can respond to that if he has an answer.

MR. MOELLER: I do have a coupld of questions for the Applicant. I hope they are short.

MR. KERR: So do I.

Do you have any additional comments to make, Mr. Colbert?

MR. COLBERT: Not at this time.

MR. KERR: Mr. Moeller, would you ask your questions of the Applicant, then, please.

MR. MOELLER: You do have an augmented offgas system, charcoal absorption system for your steam injector.

MR. COLBERT: Yes.

MR. MOELLER: Have you done anything in your plant to counteract the LER's which come out of some BWR plants similar to yours due to the fact that the ventilation fan through the compartment or for the compartment through which the steam lines to your HPSI steam-driven turbine pump becomes isolated? What it is, there have been a number of LER's where the fan ceases operation, the compartment heats up, the signal in the control room is that there has been a break in the steam line feeding the HPSI turbine-driven pump.

Do you isolate your system? Have you done anything so that this will not occur at your plant?

MR. COLBERT: One moment.

(Pause.)

MR. COLBERT: No. We will look into it.

MR. MOELLER: Are you satisfied, then, to have your HPSI system isolated?

MR. KERR: There is somebody in the back waving. Are you waving at us?

MR. COLBERT: Rich, come to a microphone if you have something.

While waiting for that, Dr. Moeller, we have looked at the one that indicates the high steam flow. That to my recollection is the majority of the LER's in that area, the steam line break thing which trips on an initiation of the HPSI. It says, despite high flow, and you get a trip. That has been where most of the LER's were that I saw in that area, and we have worked with that.

But now, Rich, do you have anything on the other one? The temperature, the high temperature is the question.

MR. ANDERSON: Rich Anderson, Detroit Edison system engineer.

Was the question on the loss of the ventilation causing the leak detection?

MR. MOELLER: Causing the temperature in the compartment to increase and then indicating that the steam line pipe had broken.

MR. ANDERSON: Right, the leak detection.

MR. MOELLER: Right.

MR. ANDERSON: As far as following the LER's, we have. We have looked at a number of them.

MR. MOELLER: Have you found a way to counteract this so you do not suffer the same problem?

MR. ANDERSON: We think so. We have a little bit different design in the ventilation and there is a set point program that we have been working with GE on to try to, you know, get the right set point to avoid that problem.

MR. MOELLER: My point is, instead of having a temperature gauge that isolates that system, you could have moisture when the steam comes out or you could have a density gauge of some sort that saw the cloud of steam in the compartment. There would be other possible ways of doing this, and I just wondered if you had looked at it.

MR. ANDERSON: No. We have been working with the temperature to enhance it, but --

MR. MOELLER: I have a last question for the staff. I guess you are going to go to them next, so I will wait.

MR. KERR: Mr. Kintner, do you have any additional comments that you want to make, including possible response to Mr. Moeller's question?

MR. KINTNER: Yes. I would like to respond to the questions that staff got this morning regarding the NRC interface with Canada and the emergency plans. The Federal Emergency Management Agency handles the interface with Canada on emergency plans.

MR. KERR: Thank you.

MR. MOELLER: And are they doing it? It is fine for them to have that responsibility.

MR. KINTNER: Yes. Let me ask Mr. Jaske if he would comment on that.

MR. KERR: Answer yes or no, Mr. Jaske.

MR. JASKE: We have a great deal of activity going on with Canada on this. But like all international affairs, sometimes it moves somewhat slower than we would like it to move. But I can comment in detail on our discussions with Canada and arrangements that are being made for this specific site, to the extent that you may want.

MR. KERR: What would you like to know, Mr.

Moeller?

MR. MOELLER: At this stage I mainly wanted to know, are such negotiations and discussions under way. So --

MR. JASKE: Yes, they are.

MR. KERR: Would you like to know, for example, whether they are in French or in English?

MR. MOELLER: No. The answer is adequate.

Well, I have one question left with the staff.

MR. KERR: Please.

MR. MOELLER: If they can answer.

MR. KINTNER: Yes. Regarding your question on why routine radioactive releases are so much less in the Fermi 2 safety evaluation report compared to LaSalle, Fermi 2 has two and a half times as much charcoal as LaSalle. That is basically the reason.

MR. MOELLER: And why would the staff approve so much less for LaSalle? I suggest -- what is the reason?

MR. KERR: It is hard to get charcoal in central Illinois.

(Laughter.)

MR. KINTNER: Well, I think the reason is LaSalle also meets the regulations, they meet our requirements.

MR. MOELLER: Fermi just does it a little better.

MR. KINTNER: Fermi does it a little better.

MR. MOELLER: My last question on the SER, on page 9-2 it talks about defective fuel storage racks. And I wondered what a defective fuel storage rack was -- is. This is with regard to spent fuel storage.

MR. KINTNER: Yes. Those are storage racks to store defective spent fuel.

MR. MOELLER: Oh, it is not the storage rack that is defective.

MR. KINTNER: That is right.

MR. MOELLER: Oh, this is then spent fuel that is a leaker.

MR. KINTNER: Yes.

MR. MOELLER: It is not routine spent fuel.

MR. KINTNER: That is right.

MR. MOELLER: It is defective spent fuel stored in these racks.

MR. KINTNER: I understand so, yes. I have a few more questions I have the answers for. The -- well, the windrose was left out of the safety evaluation report. It is not a standard practice to put these into the safety evaluation report.

MR. KERR: Okay. Mr. Moeller just must have picked up those that had, because he said that this was the first one he had seen that did not have one.

MR. KINTNER: You had a question on, Dr. Kerr, on whether NRC has licensed operators on review teams of emergency operating procedures. Sam McKay, formerly senior reactor operator, he is in the procedures and test review branch, and -- most of the men in the branch are reviewing emergency operating procedures and they have nuclear navy operating experience. Many of them do.

MR. KERR: Do you trust those guys?

(Laughter.)

MR. KINTNER: They sound pretty good to me.

MR. KERR: Okay.

MR. KINTNER: Let's see. I think the only other question was Dr. Carbon's question on differing opinions. I mentioned there was a differing opinion, not formal professional difference of opinion, but within the branch on Fermi 2, on the fire test criteria. And it has been resolved within the branch and there is no difference of opinion on the fire test criteria.

We still have an open item on that. We are looking at the fire test results, as I mentioned this morning.

MR. KERR: If I can add some gratuitous comment, it does seem to me that if you could you would want somebody who had had commercial operating experience. I mean, the Navy people do a very good job of operating submarines, but their training, the way they operate, their philosophy, it seems to me are quite different than is the case with commercial plants. And I just do not think there is anything like some experience with operation to give them perspective, that would give the perspective I think you would want.

I realize you are not the one who makes this decision. But that is --

MR. KINTNER: I will look into that.

MR. KERR: I also want to apologize fpr having ignored Mr. Batch and Mr. Beaudry. In trying to prepare my agenda for the next meeting, I inadvertently marked them off of today's presentation. And I did want to get information on the capacity for onsite storage. And I am not sure whether this is a batch of spent fuel or whether there is somebody whose name is Batch. But I an going to assume it is the latter and that he is going to tell me something about the capacity of your onsite storage of spent fuel.

MR. COLBERT: You are correct, Dr. Kerr. This is Mel Batch, systems engineer.

MR. BATCH: My name is Mel Batch and I am a project systems engineer. The high density fuel storage racks have a capacity for just a shade over three full -- 2505 cells, which will give us a storage capacity for approximately nine years of operation at an 80 percent capacity factor and still have room for a total discharge.

MR. KERR: I am sorry, I was looking at a note that Dr. Moeller had made. If you would not mind repeating that, I would appreciate it very much.

MR. BATCH: We have space for 2300 fuel assemblies, which would give us nine years of operation at 80 percent capacity factor and still leave us room for a full discharge.

MR. KERR: About nine years.

MR. BATCH: Yes.

MR. KERR: Thank you.

Any questions about the spent fuel storage other than that? That was the information I wanted. Thank you.

(No response.)

And Mr. Beaudry, unless you are going to use a slide --

MR. BEAUDRY: I was, Dr. Kerr, but I do not have to.

MR. KERR: Why don't you not, then.

(Laughter.)

MR. BEAUDRY: Okay. As far as solidified radwaste drums are concerned, we have right now built into our 13 storage conveyors somewhere between six months worth and one year's worth of storage available. That is about --

MR. KERR: Do you have any contingency plans in case somebody decides you cannot ship?

MR. BEAUDRY: We sure do, yes, we do.

MR. KERR: Could you tell me something about them?

MR. BEAUDRY: We have been look at it, considering it for some time, and we are about to let a contract out for a good detailed design for a contingency storage building that will be attached directly to our present radwaste system. Within the next possibly six months we have to make a decision as to which.

MR. KERR: As I remember, there is a law in the state of Michigan that prohibits storage of radwaste, but it may not prohibit storage of this kind. You must know.

MR. BEAUDRY: I believe our legal people have looked into it and feel that there is no such law that would apply to this sort of operation.

MR. KERR: This is really not considered storage in the sense of that law.

MR. BEAUDRY: We are not going to bury it in the ground or anything like that. This is extension of our present storage, for all intents and purposes.

MR. KERR: Yes, sir?

MR. WIGLEY: Al Wigley, licensing engineer.

That law in Michigan refers to the disposal of radioactive wastes, not the storage. In fact, the law allows storage of waste that you produce yourself.

MR. KERR: I have thought that, but I just was not sure and I wanted to be certain that I remembered correctly.

Thank you, Mr. Beaudry.

Anything else?

(No response.)

I think then that this will be all -- Mr. Carbon, excuse me.

MR. CARBON: I wanted to address a couple of questions to the staff.

MR. KERR: Please.

MR. CARBON: Mr. Kintner, in the SER on page 417 there is a paragraph on hydrodynamics ability and in that paragraph it says: "However, in order to provide additional margins," and so on, "plant natural circulation at Fermi 2 will be prohibited until our review and so on."

Can you briefly and quickly indicate what the problem and concern is there, or if not would you refer me to a report where I can read about it?

MR. KINTNER: Well, no, I cannot -- I do not know the details of that review, so I cannot speak to that. I do not have anyone here to speak to it.

As far as operation under natural circulation, we plan to have a license condition so that there would not be operation under deliberate natural circulation conditions until the analysis is made. But no, I cannot -- I do not know about the technical aspects of the decreased margin under natural circulation conditions.

MR. CARBON: Would you be prepared at the August meeting to say something very briefly about it?

MR. KINTNER: I certainly will.

MR. CARBON: My last question, the last sheet on your handout this morning had the items to be completed after license issuance. Some of these sound like they might be of possible considerable importance. I presume that the ones that would fit that category, you ve concluded that the probability of an accident arising is low enough that you can let them go until some time after licensing. Is that so?

MR. KERR: Do you understand the question?

MR. KINTNER: Yes, I understand the question. The licensing conditions, why are we allowing operation there for the length of time that we are. Okay. There are -- the first three or so on the list -- high fission gas pressure, channel box deflection, and so on -- these are of course not applicable until the second fuel loading. They have already made the analysis for the first fuel cycle.

On the others, they provide additional assurance of safety margins for long-term operations, but they are not required for the first fuel cycle because of the low probability of them occurring -- of having a problem in that area.

MR. CARBON: Just a probability.

MR. KINTNER: Probability, yes.

MR. KERR: Further questions?

(No response.)

Anybody have any further questions or comments?

(No response.)

We will go into closed session very shortly, then, and after the closed session, which I hope will not take more than 15 minutes, we will have a very brief open session, at which time I will confer with my colleagues to see if we recommend that this go to the full Committee in August. May we go into closed session, then. There will be no more recording required after this, and there will be a brief hiatus while those who are not supposed to be in the closed session remove themselves, please.

(Whereupon, at 4:37 p.m., the meeting was recessed, to be followed by a closed executive session.)

* * *

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/Subcommittee on Fermi-2 Operating License Review

Date of Proceeding: July 24, 1981

Docket Number: ________________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Mary C. Simons

Official Reporter (Typed)

Mary C Simons

Official Reporter (Signature)

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David S. Parker

Official Reporter (Typed)

[signature]

(SIGNATURE OF REPORTER)

ACRS SUBCOMMITTEE MEETING
FERMI-2 OPERATING LICENSE REVIEW
JULY 24, 1981
WASHINGTON, D. C.

-TENTATIVE SCHEDULE OF PRESENTATIONS-

	PRESENTATION TIME	ACTUAL TIME
I. INTRODUCTION		
W. KERR, CHAIRMAN	5 MIN	8:30 A.M.
A. NRC INTRODUCTION		
1. OVERVIEW OF OL REVIEW	5 MIN	8:35 A.M.
2. SER OPEN ITEMS	10 MIN	8:45 A.M.
B. DETROIT EDISON INTRODUCTION	10 MIN	9:00 A.M.
1. SITE AND PLANT DESCRIPTION L. E. SCHUERMAN		
2. RESPONSE TO SER OPEN ITEMS W. F. COLBERT		
II. DISCUSSION OF OL REVIEW ISSUES		
A. ORGANIZATION AND MANAGEMENT W. H. JENS		
1. COMPLIANCE WITH NUREG-0731--"MANAGEMENT STRUCTURE AND TECHNICAL RESOURCES"	10 MIN	9:15 A.M.
- NRC STAFF COMMENT ON ITEM 1	5 MIN	9:25 A.M.
2. FEEDBACK TO OPERATORS FROM INTERNAL SAFETY GROUPS	10 MIN	9:45 A.M.
3. FEEDBACK TO OPERATORS AND STAS OF EXPERIENCE FROM OTHER PLANTS	5 MIN	10:00 A.M.
BREAK	10 MIN	10:10 A.M.

-TENTATIVE SCHEDULE OF PRESENTATIONS-

		PRESENTATION TIME	ACTUAL TIME
B.	OPERATOR TRAINING L. E. KANOUS		
	1. OPERATOR SELECTION	10 MIN	10:20 A.M.
	2. USE OF SIMULATORS IN TRAINING PROGRAM	5 MIN	10:35 A.M.
	3. TRAINING FOR SERIOUS ACCIDENTS INCLUDING ACCIDENTS BEYOND DBA'S	5 MIN	10:45 A.M.
	4. SELECTION AND TRAINING OF MAINTENANCE PERSONNEL	10 MIN	10:55 A.M.
C.	CONTROL ROOM E. LUSIS		
	1. DESIGN REVIEW (HUMAN FACTORS, ETC.)	10 MIN	11:15 A.M.
	2 HABITABILITY FOR SERIOUS ACCIDENTS (BEYOND DBA) R. J. BEAUDRY	5 MIN	11:30 A.M.
D.	INSTRUMENTATION TO FOLLOW THE COURSE OF A SERIOUS ACCIDENT	20 MIN	11:40 A.M.
	1. INADEQUATE CORE COOLING INSTRUMENTATION L. F. WOODEN		
	2. REGULATORY GUIDE 1.97 REQUIREMENTS L. F. WOODEN		
	3. NRC STAFF COMMENT ON ICC INSTRUMENTATION REQUIREMENT		
E.	PLANT SEISMIC DESIGN F. E. GREGOR (W. M. STREET)	30 MIN	12:10 P.M.
	1. RECENT REANALYSIS OF STRUCTURES SYSTEMS, AND COMPONENTS REQUIRED FOR SAFETY SHUTDOWN - MOST VULNERABLE PART OF DECAY HEAT REMOVAL SYSTEM		
	2. NRC STAFF COMMENTS ON SEISMIC REANALYSIS		

-TENTATIVE SCHEDULE OF PRESENTATIONS-

		PRESENTATION TIME	ACTUAL TIME
	LUNCH	60 MIN	12:55 P.M.
F.	DECAY HEAT REMOVAL W. F. COLBERT (M. K. DEORA) - NORMAL AND DEGRADED MODES	10 MIN	1:55 P.M.
G.	RELIABILITY OF STATION ELECTRICAL POWER T. M. MCKELVEY 1. LOSS OF AC/DC T. M. MCKELVEY 2. STATION BLACKOUT T. M. MCKELVEY	15 MIN	2:15 P.M.
H.	MARK I CONTAINMENT MODIFICATIONS D. F. LEHNERT 1. STATUS 2. NRC STAFF COMMENT	10 MIN	2:35 P.M.
I.	ENVIRONMENTAL QUALIFICATION OF EQUIPMENT Q. H. DUONG	10 MIN	2:50 P.M.
J.	ATWS E. M. PAGE - DETROIT EDISON POSITION ON ATWS MODIFICATIONS - SDV MODIFICATIONS	10 MIN	3:05 P.M.
	BREAK	10 MIN	3:20 P.M.
K.	HYDROGEN CONTROL J. R. GREEN 1. DETROIT EDISON PRESENTATION 2. NRC REVIEW OF HYDROGEN CONTROL MEASURES	10 MIN	3:30 P.M.

-TENTATIVE SCHEDULE OF PRESENTATIONS-

		PRESENTATION TIME	ACTUAL TIME
L.	TMI-ISSUES-STATUS (BALANCE OF ITEMS NOT DISCUSSED ABOVE) E. P. GRIFFING, W. F. COLBERT	10 MIN	3:45 P.M.
M.	EMERGENCY PLANNING E. F. MADSEN	15 MIN	4:00 P.M.
	1. EMERGENCY SUPPORT FACILITIES		
	2. ROLE OF FEMA, STATE, AND LOCAL AGENCIES		
N.	CAPACITY FOR ONSITE STORAGE OF SPENT FUEL M. L. BATCH		
O.	CAPABITY FOR ONSITE STORAGE OF LOW-LEVEL WORK R. J. BEAUDRY		
P.	SECURITY (CLOSED SESSION) W. W. HODGES	15 MIN	4:15 P.M.
III.	DISCUSSION	15 MIN	4:30 P.M.
IV.	SUBCOMMITTEE CAUCUS AND INSTRUCTIONS TO APPLICANT	5 MIN	4:55 P.M.
V.	ADJOURN		5:00 P.M.

SITE AND PLANT DESCRIPTION

FERMI-2 PLANT IS LOCATED IN A 1120-ACRE SITE, APPROXIMATELY 30 MILES SOUTH OF DETROIT AND 25 MILES NORTHEAST OF TOLEDO, OHIO AS SHOWN IN FIGURE 1. THE SEVERAL CONTINGUOUS BUILDINGS COMPRISING THE FERMI-2 PLANT ARE SITUATED ON THE WESTERN SHORE OF LAKE ERIE AS SHOWN IN FIGURE 2.

THE 1,120-ACRE FERMI 2 SITE IS LOCATED ON THE WESTERN SHORE OF LAKE ERIE IN FRENCHTOWN TOWNSHIP, MONROE COUNTY, MICHIGAN. THE PLANT IS APPROXIMATELY 30 MILES SOUTHWEST OF DOWNTOWN DETROIT, AND 25 MILES NORTHEAST OF TOLEDO, OHIO. APPROXIMATELY 90% OF THE LAND AREA WITHIN TEN MILES OF THE PLANT LIES WITHIN MONROE COUNTY; THE REMAINING 10% IS IN WAYNE COUNTY. OF THE TEN-MILE AREA IN MONROE, APPROXIMATELY 55% CONSISTS OF FARMLAND. WITHIN A 50-MILE RADIUS OF THE SITE ARE ALL, OR PORTIONS OF, ELEVEN COUNTIES IN MICHIGAN, TEN IN OHIO, AND TWO IN ONTARIO, CANADA. THE 1980 CENSUS DATA SHOWED A POPULATION OF 84,600 WITH TEN MILES OF THE PLANT AND 5.5 MILLION WITHIN 50 MILES.

FERMI 2 UTILIZES A BWR-4 BOILING WATER REACTOR DESIGNED AND SUPPLIED BY GENERAL ELECTRIC AND A MARK I CONTAINMENT. THE REACTOR CONTAINS THE CORE, CONTROL

RODS, INSTRUMENTATION, STEAM SEPARATOR AND DRIER ASSEMBLIES, JET PUMPS, AND THE CONTROL ROD DRIVE MECHANISMS, WHICH ARE MOUNTED ON THE BOTTOM OF THE REACTOR PRESSURE VESSEL.

THE REACTOR CORE CONTAINS 764 FUEL ASSEMBLIES AND 185 CONTROL RODS ARRANGED IN AN UPRIGHT CYLINDER CONFIGURATION. EACH FUEL ASSEMBLY CONSISTS OF AN 8 x 8 ARRAY OF RODS, 62 OF WHICH CONTAIN FUEL AND TWO OF WHICH CONTAIN WATER. THE DESIGN POWER LEVEL OF THE REACTOR IS 3430 MEGAWATTS THERMAL.

THE EMERGENCY CORE COOLING SYSTEM IS OF CONVENTIONAL DESIGN, EMPLOYING A HIGH PRESSURE COOLANT INJECTION SYSTEM, A LOW PRESSURE COOLANT INJECTION SYSTEM, AND A CORE SPRAY SYSTEM, AND AN AUTOMATIC DEPRESSURIZATION SYSTEM TO BRIDGE THE CAPABILITIES OF THE HIGH PRESSURE AND LOW PRESSURE SYSTEMS. THE RESULTING SYSTEM PROVIDES REDUNDANCY AND DIVERSITY.

THE IN-HOUSE ELECTRICAL DISTRIBUTION SYSTEM IS DESIGNED TO PROVIDE SUFFICIENT NORMAL AND STANDBY SOURCES OF ELECTRICAL POWER TO PERMIT SAFE SHUTDOWN AND TO MAINTAIN THE PLANT IN A SAFE CONDITION UNDER ALL CREDIBLE CIRCUMSTANCES. IN ADDITION, THE POWER SOURCES ARE ADEQUATE TO ACCOMPLISH ALL ESF FUNCTIONS REQUIRED UNDER POSTULATED DESIGN BASIS ACCIDENT CONDITIONS.

THE FERMI 2 FACILITY EMPLOYS A RADWASTE SYSTEM DESIGNED TO LIMIT THE DOSE TO THE GENERAL PUBLIC DUE TO RADIOACTIVE EFFLUENTS TO LEVELS WHICH ARE AS LOW AS REASONABLY ACHIEVABLE. IN ADDITION, THE FERMI 2 FACILITY DESIGN INCORPORATES FEATURES WHICH MINIMIZE THE OCCUPATIONAL RADIATION EXPOSURE UNDER NORMAL AND POSTULATED ACCIDENT CONDITIONS.

THE DETROIT EDISON COMPANY POSSESSES A LARGE POOL OF QUALIFIED PERSONNEL TO SUPPORT THE FERMI 2 PLANT. MOST OF THE NUCLEAR OPERATIONS PERSONNEL ARE LOCATED ADJACENT TO THE PLANT SITE. OPERATING PERSONNEL CAN DRAW UPON THIS RESOURCE AS NEEDED. IN ADDITION, SIGNIFICANT OVERSIGHT IS PROVIDED BY DETROIT EDISON MANAGEMENT TO ASSURE THAT OPERATIONS WILL BE SAFE AND EFFICIENT. THIS OVERSIGHT PLUS THE INTIMATE, EXPERIENCED SUPPORT STAFF ENSURES THAT THE OPERATING STAFF WILL BENEFIT FROM EXPERIENCE GAINED THROUGH DESIGN AND OPERATION OF FERMI 2 AND FROM OTHER NUCLEAR FACILITIES.

FERMI 2 IS SIMILAR TO A NUMBER OF MODERN BWR POWER REACTORS THAT ARE PRESENTLY OPERATING IN THE UNITED STATES, SUCH AS HATCH AND BROWNS FERRY. THE ENCLOSED TABLE SUMMARIZES KEY DESIGN PARAMETERS OF FERMI 2.

WITH ISSUANCE OF A CONSTRUCTION PERMIT FOR FERMI 2, THE ACRS IDENTIFIED A NUMBER OF OUTSTANDING ITEMS WHICH MUST BE RESOLVED PRIOR TO OPERATION OF FERMI 2. THESE ITEMS,

ADDRESSED IN APPENDIX B OF THE FERMI 2 FSAR, HAVE BEEN RESOLVED. IN ADDITION, SINCE CONSTRUCTION OF FERMI 2 HAS BEEN STARTED, A NUMBER OF GENERIC ISSUES HAVE BEEN IDENTIFIED, REQUIRING DESIGN CHANGES, AS FOLLOWS:

a. IGSCC (1974-78) RESULTED IN MATERIAL, PROCESS, WELD, AND SYSTEM CHANGES: RR, CRD, CORE SPRAY SPARGERS.

b. FIRE AT BROWNS FERRY RESULTED IN COMPLETE FIRE HAZARDS ANALYSIS, INSTALLED FIRE DETECTION, AND PROTECTION SYSTEMS UPGRADED: SAFE SHUTDOWN ANALYSIS AND SOME PLANT CHANGES: FIRE BARRIERS, FIRE STOPS, MORE FIRE PROTECTION APPARTUS.

c. SECURITY REQUIREMENT - EDISON'S PLAN ORIGINATED FOR FERMI 2 CONTROLS ACCESS TO AN ESSENTIALLY LOCKED PLANT WHICH USES NATURAL PLANT STRENGTH AS THE BARRIER TO INTRUSION AND HAS EXTERNAL WARNING PERIMETER WITH MULTIPLE SENSING TO ALERT GUARD FORCES.

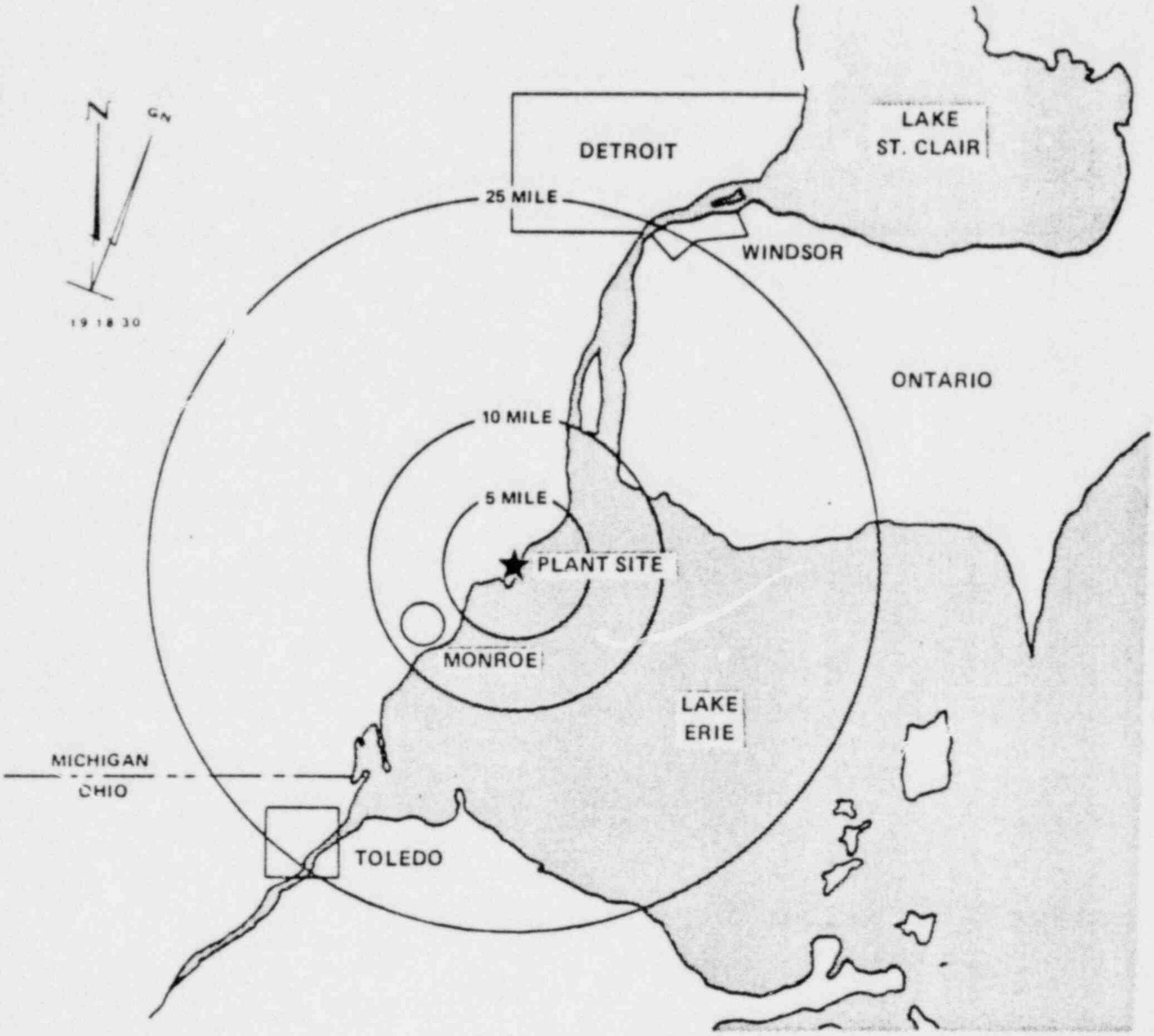
d. CRD SYSTEM REFINEMENT WITH REMOVAL OF CRD RETURN LINES TO VESSEL.

e. FEEDWATER NOZZLE CRACKING PROBLEMS (1975-1978) AT OTHER PLANTS RESULTED IN SPARGERS CHANGED OUT TO LATER DESIGN.

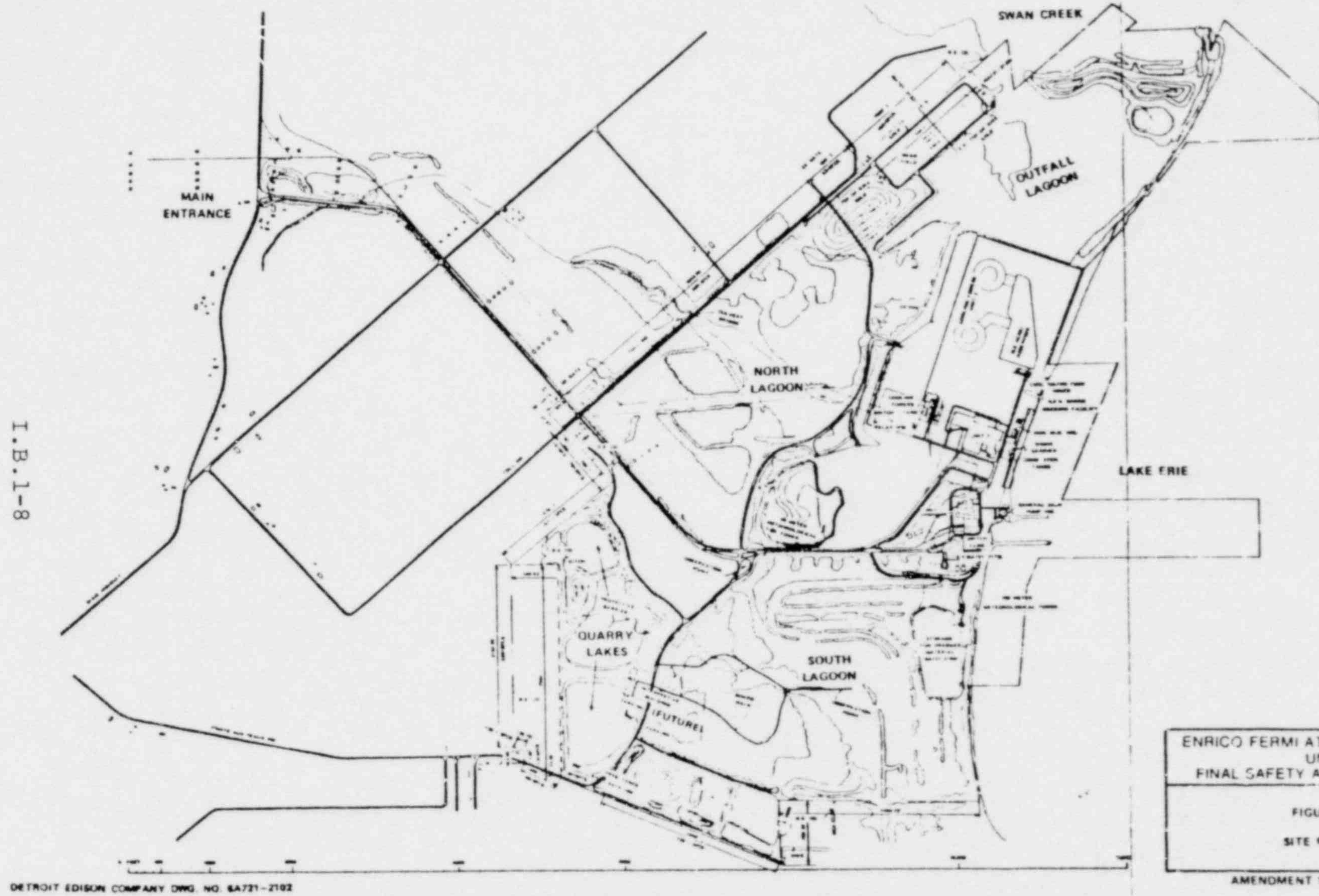
SUMMARY OF PLANT DESIGN

DESIGN FEATURE	FERMI-2
RATED THERMAL POWER (MW_{th})	3293
ECCS DESIGN POWER (Mw_{th})	3430
GROSS ELECTRICAL OUTPUT (Mw_e)	1154
MAINSTREAM FLOW RATE (lb/hr)	$14.156(10^6)$
REACTOR TOTAL FLOW RATE (lb/hr)	$100.0(10^6)$
SYSTEM PRESSURE (psi)	1005
VESSEL SIZE (DIAMETER, IN.)	251
VESSEL DESIGN PRESSURE (psi)	1250
NUMBER OF FUEL ASSEMBLIES	764
FUEL TYPE (8x8)	62 + 2
RECIRC. LOOP INSIDE DIAMETER (IN.)	28
MAXIMUM LINEAR POWER GENERATION (Kw/ft)	13.4
MAXIMUM FUEL TEMPERATURE (^oF)	3435
TOTAL PEAKING FACTOR	2.43
CORE HEIGHT (IN.)	150
NUMBER CONTROL RODS	185
MAIN CONDENSOR CAPACITY (BTU/hr)	7547×10^6

DESIGN FEATURE (CONTINUED)	FERMI-2
CIRCULATING WATER PUMPS	5
LPCS (NUMBER + FLOW RATE)	2 @ 6250
HPCI (NUMBER + FLOW RATE)	1 @ 5000
LPCI (NUMBER + FLOW RATE)	3 @ 10000
ADS	5 VALVES
RHR (NUMBER LOOPS + FLOW RATE)	4 @ 7200
RHR HEAT EXCHANGERS (NUMBER + DUTY)	2 @ 41.6(10^6)
RCIC (gpm)	600
PRESSURE SUPPRESSION TYPE	MK I STEEL
SUPPRESSION DESIGN PRESSURE (PSI)	
EXTERNAL	2
INTERNAL	56
DRYWELL VOLUME (ft^3)	163,780
WETWELL AIRSPACE (ft^3)	130,900
SUPPRESSION POOL VOLUME (ft^3)	117,450
DRYWELL TEMPERATURE (^oF)	281



ENRICO FERMI ATOMIC POWER PLANT UNIT 2 FINAL SAFETY ANALYSIS REPORT
FIGURE 1 GENERAL REGION OF THE FERMI SITE



ENRICO FERMI ATOMIC POWER PLANT
UNIT 2
FINAL SAFETY ANALYSIS REPORT

FIGURE 2

SITE PLOT PLAN

AMENDMENT 1 - NOVEMBER 1975

DETROIT EDISON RESPONSE TO SER OPEN ITEMS

	PRELICENSING ISSUES	STATUS AT DETROIT EDISON
AUG.	1. RULES AND REGULATIONS	WORK IN PROGRESS, COMPLETION EXPECTED AUG.1
AUG.	2. GEOTECHNICAL ENGINEERING	
	a. BURIED PIPING	CONSIDERED CLOSED AT AUDIT JULY 22
	b. SHORE BARRIER	CONSIDERED CLOSED AT AUDIT JULY 22
	3. SEISMIC REASSESSMENT	REV.1 OF SUPPLEMENTARY EVALUATION DOCKETED JULY 15, FOLLOWUP ON DEFICIENT COMPONENTS IN PROGRESS.
	4. MARK I CONTAINMENT	PUA TO BE COMPLETED MAY, 1982, AS COMMITTED
	5. PRESERVICE TESTING	ASME SECTION XI TEST PROGRAM BEING REWRITTEN
	6. SEISMIC & DYAMIC QUALIFICATION-	SQRT AUDIT SCHEDULE WEEK OF JULY 27
	7. ENVIRONMENTAL QUALIFICATION	NRC AUDIT TOOK PLACE JULY 13-17, FOLLOWUP IN PROGRESS
	8. SCRAM DISCHARGE VOLUME BREAK	NRC REVIEWING BWR GENERIC RESPONSE
AUG.	9. 10CFR50 APPENDICES G & H	AS COMMITTED, ADDITIONAL INFORMATION JUSTIFYING EXEMPTIONS WILL BE PROVIDED BY AUGUST 1
	10. CONTAINMENT LEAKAGE TESTS	JULY 17 CALL REVEALED NEW NRC CONCERNS ABOUT DETROIT EDISON APPENDIX J TESTING
AUG.	11. IE BULLETIN 79-27	WE BELIEVE THIS IS CLOSED BY OUR JULY 22 LETTER
AUG.	12. FIRE PROTECTION	AS COMMITTED, ALL MODIFICATIONS WILL BE INSTALLED BY FUEL LOAD. DOCUMENTATION OF PANEL TEST WILL BE PROVIDED

	13.	DIESEL GENERATOR DRY STARTS	AS COMMITTED, DESIGN AND TEST DATA WILL BE SUBMITTED BY SEPTEMBER 1. WORK NOW IN PROGRESS AT STONE & WEBSTER
AUG.	14.	SECURITY PLAN	REVISIONS ANSWERING NRC'S CONCERN BY JULY 31
AUG.	15.	ATWS	SIMULATOR WALK-THROUGH SCHEDULED FOR JULY 25-26
	16.	TMI ISSUES	
		a. I.C.1 PROCEDURE GUIDELINES	NRC REVIEWING BWR OWNERS GROUP GUIDELINES
AUG.		I.C.5 EXPERIENCE	TO BE COMPLETED AUG. 1
		I.C.8 MONITORING OF PROCEDURES	SIMULATOR WALK-THROUGH SCHEDULED FOR JULY 25-26
AUG.		b. I.D.1 CONTROL ROOM REVIEW	AS COMMITTED, WILL MEET THE AUGUST 1 DEADLINE FOR HVAC, PROCEDURE FOR PANEL MODIFICATIONS, AND FERMI 2 COLOR CODE STANDARD.
		c. I.G.1 LOW-POWER TESTING	AS COMMITTED, PROCEDURE AND SAFETY ANALYSIS FOR SIMULATED BLACKOUT TEST REQUIRED BY NRC WILL BE PROVIDED AT LEAST 90 DAYS BEFORE FUEL LOAD.
AUG.		d. II.B.4 DEGRADED CORE TRAINING	OUTLINE OF REVISED TRAINING PROGRAM WILL BE PROVIDED AUG. 1

	e. II.D.1 SRV TESTS	NRC TO REVIEW GENERIC TEST DATA
	f. II.E.4.2 CONTAINMENT ISOL.	NRC TO CONDUCT CONFIRMATORY AUDIT OF PURGE VALVE OPERABILITY.
	g. III.A.1.1 EMER. PREPAREDNESS	REVISED EMERGENCY PLAN WILL BE SUBMITTED BY SEPTEMBER 1
	III.A.1.2 EMER. FACILITIES	UNDER NRC REVIEW
AUG.	III.A.2 EMER. PREP.	AS COMMITTED, REVISED MET TOWER INSTRUMENTATION WILL BE PROVIDED AUGUST 1, AND MET MODEL BY NOVEMBER 1.

LICENSE CONDITIONS

1.	ANALYSIS OF FISSION GAS RELEASE	NEW ANALYSIS REQUIRED PRIOR TO SECOND CYCLE
2.	CHANNEL BOX DEFLECTION	REQUIRED BEFORE SECOND CYCLE
3.	HYDRODYNAMIC STABILITY ANALYSIS	REQUIRED BEFORE SECOND CYCLE
4.	MULTIPLE CONTROL FAILURES	REQUIRED BEFORE SECOND CYCLE
5.	HELB EFFECTS ON CONTROL SYSTEMS	WITHDRAWN BY NRC STAFF
6.	LP TURBINE DISC INSPECTION	WILL BE DONE AS COMMITTED
7.	RETAIN PERSON WITH BWR EXPERIENCE	DETROIT EDISON DISAGREES WITH THE WORDING OF THE NRC REQUIREMENT TO RETAIN BWR EXPERIENCED PERSONNEL UP TO 100% POWER PREFERRING TO RETAIN THEM UNTIL 9 MONTHS OPERATION.
8.	POST-ACCIDENT SAMPLING	DISAGREE WITH NRC ON REQUIREMENT TO SAMPLE BORON, CHLORIDES, pH AND DISSOLVED OXYGEN.

9.	INSTR. TO DETECT INADEQUATE CORE COOLING	DISAGREE WITH NRC POSITION ON INCORE THERMOCOUPLES

ORGANIZATION AND MANAGEMENT

THE DETROIT EDISON NUCLEAR OPERATIONS ORGANIZATION HAS BEEN STRUCTURED TO BE AN ALMOST INDEPENDENT OPERATING UNIT OF THE COMPANY, CONSISTING OF ABOUT 500 PERSONS. TO UNDERSTAND THIS ORGANIZATIONAL STRUCTURE AND ITS STRENGTHS, IT IS DESIRABLE TO BRIEFLY PROVIDE A SHORT HISTORY OF DETROIT EDISON'S NUCLEAR INVOLVEMENT.

THE COMPANY WAS ONE OF THE ORIGINAL UTILITIES THAT PARTICIPATED IN EARLY POWER DEMONSTRATION PROGRAMS UNDER THE "ATOMS FOR PEACE" PROGRAM. AS A RESULT, ITS EXPERIENCE EXTENDS OVER SOME 25 TO 30 YEARS.

THE COMPANY WAS THE LEAD UTILITY IN ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. AND THE POWER REACTOR DEVELOPMENT COMPANY--ORGANIZATIONS THAT DEVELOPED, TESTED AND OPERATED THE FERMI 1 BREEDER REACTOR. MANY OF THE THE PEOPLE IN KEY POSITIONS ON FERMI 1 THEN WENT ON IN THE EARLY 1970's TO DESIGN THE FERMI 2 PLANT, AND MANY WILL BE INVOLVED IN ITS OPERATION. AS A MATTER OF FACT, OUR NEW CHIEF EXECUTIVE OFFICER, MR. WALTER MCCARTHY, WAS THE INITIAL PROJECT MANAGER OF FERMI 2 AND WAS THE GENERAL MANAGER OF THE POWER REACTOR DEVELOPMENT COMPANY.

FURTHERMORE, IN ADDITION TO FERMI 2, EARLY IN THE 1970's, THE COMPANY INITIATED FERMI 3 AND TWO NUCLEAR UNITS AT ITS GREENWOOD ENERGY CENTER--GREENWOOD 2 and 3.

THE COMPANY EXPANDED ITS HUMAN RESOURCES TO HANDLE ALL OF THESE ENGINEERING ACTIVITIES. IN THE CASE OF FERMI 2, THE COMPANY TOOK THE LEAD ENGINEERING RESPONSIBILITY: HOWEVER, IN THE CASE OF FERMI 3 AND GREENWOOD 2 AND 3, THE COMPANY CONTRACTED THE LEAD DESIGN RESPONSIBILITY TO OTHER A/E'S BUT ACTIVELY PARTICIPATED IN THE SPECIFICATIONS OF THE PLANT, IN CRITICAL DECISIONS, IN THE LICENSING AND IN REVIEWS OF THE DESIGN WORK BEING CONDUCTED FOR THE COMPANY.

HOWEVER, IN 1975, WE TERMINATED WORK ON FERMI 3; AND IN 1980, WE TERMINATED ALL WORK ON GREENWOOD 2 AND 3. THIS RELEASED ADDITIONAL RESOURCES TO COMPLETE FERMI 2 AND PERMITTED US TO CONCENTRATE THESE RESOURCES AND TO DEDICATE VERY KNOWLEDGEABLE ENGINEERS INVOLVED IN THE DESIGN AND CONSTRUCTION OF THE PLANT TO THE OPERATION, MODIFICATIONS AND POTENTIAL PROBLEM SOLVING OF FERMI 2 WHEN THE PLANT GOES INTO OPERATION. SINCE WE HAVE NO OTHER NUCLEAR RESPONSIBILITIES OR OBLIGATIONS WITHIN THE COMPANY, IT ALSO PERMITS US TO LOCATE THESE VALUABLE HUMAN RESOURCES ON THE PLANT SITE IN A NEW 100,000 SQUARE FOOT FACILITY THAT WE ARE CONSTRUCTING. THIS PERMITS TECHNICAL RESOURCES TO BE AVAILABLE 100 PERCENT OF THE TIME IN CLOSE CONTACT WITH THE OPERATING PERSONNEL TO AID THEM IN EVERY WAY POSSIBLE TO CARRY OUT THE SAFE AND SUCCESSFUL OPERATION OF FERMI 2.

FIGURE 1 SHOWS THIS CONCENTRATION OF HUMAN RESOURCES IN ONE ORGANIZATION UNIT. THE NUMBER IN THE LOWER RIGHT CORNER OF EACH BOX IS THE PRESENT AUTHORIZED COMPLEMENT FOR THE UNIT. FIGURE 2 SHOWS A BREAKDOWN OF NUCLEAR PRODUCTION UNDER THE PLANT SUPERINTENDENT. FIGURE 3 IS THE ORGANIZATIONAL BREAKDOWN OF NUCLEAR ENGINEERING. FIGURE 4 IS THE ORGANIZATIONAL STRUCTURE OF THE NUCLEAR SAFETY AND PLANT ENGINEERING DIVISION.

FIGURE 5 SHOWS HOW THE NUCLEAR OPERATION AREA REPORTS WITHIN THE COMPANY. ALTHOUGH WE HAVE ORGANIZED NUCLEAR OPERATIONS TO CONTROL ALL SAFETY-RELATED ACTIVITIES, WE INTEND TO UTILIZE OTHER DEPARTMENTS OF THE COMPANY TO SUPPORT PLANT OPERATION AS WELL AS OUTSIDE CONTRACTORS TO SUPPLEMENT OUR STAFF IN AREAS OF DESIGN, MAINTENANCE AND PLANT MODIFICATIONS IF THEY SHOULD BE REQUIRED.

THE NRC STAFF HAS AGREED THAT OUR NUCLEAR OPERATIONS ORGANIZATION MEETS THE GUIDANCE IN NUREG-0731. THE STAFF WAS CONCERNED THAT THE PHASED TRANSITION OF PERSONNEL FROM ENGINEERING AND CONSTRUCTION ACTIVITIES TO OPERATIONAL ACTIVITIES MAY IMPACT SAFETY. IT IS IMPORTANT TO POINT OUT THAT THE MAJORITY OF THOSE PERSONNEL WHO WILL BE TRANSFERRED WILL BE ENGAGED IN BASICALLY THE SAME ACTIVITY IN EACH ORGANIZATION AND WILL CARRY OUT THEIR PREVIOUS OBLIGATIONS ONLY IN THE NEW ORGANIZATION RATHER THAN THE OLD. THE STAFF HAS FOUND OUR

SHIFT MANNING TO BE ACCEPTABLE, AND WE HAVE AGREED TO HAVE PERSONNEL FROM GENERAL ELECTRIC AVAILABLE ON EACH SHIFT AND A SENIOR GENERAL ELECTRIC ENGINEER AVAILABLE TO THE NUCLEAR PRODUCTION STAFF THROUGH STARTUP TO HIGH POWER OPERATION. THESE GENERAL ELECTRIC ENGINEERS WILL HAVE AT LEAST THREE MONTHS OF BWR STARTUP EXPERIENCE AND WILL ALL HAVE BEEN INVOLVED IN PREVIOUS PLANT STARTUPS.

ONE OF THE IMPORTANT LESSONS THAT RESULTED FROM THE ACCIDENT AT THE THREE MILE ISLAND NUCLEAR PLANT IS THE NEED FOR EVALUATING AND DISSEMINATING OPERATING EXPERIENCE, BOTH AT THE SPECIFIC PLANT AND AT OTHER NUCLEAR PLANTS.

OVER THE PAST TEN YEARS, EDISON'S SYSTEM ENGINEERS WHO HAVE BEEN DIRECTLY INVOLVED IN THE DESIGN OF FERMI 2 HAVE REVIEWED LER'S AND OTHER SOURCES OF SIGNIFICANT OPERATING INFORMATION AS EDISON RECEIVED THIS INFORMATION. THEIR REVIEWS HAVE BEEN DOCUMENTED. HOWEVER, AS MORE PLANTS CAME ON LINE AND THE OPERATING INFORMATION BECAME MORE VOLUMINOUS, THE TASK OF REVIEWING ALL INFORMATION BECAME DIFFICULT AND THE REVIEWS HAD TO BE RESTRICTED TO ONLY SIGNIFICANT EXPERIENCE. THIS SIGNIFICANT EXPERIENCE WAS DETERMINED BY THE INFORMATION THAT WAS SUMMARIZED AND ISSUED EITHER BY THE NRC, IN THE FORM OF BULLETINS, CIRCULARS, ETC., OR BY THE INSTITUTE OF NUCLEAR POWER OPERATIONS, THE NUCLEAR SAFETY ANALYSIS CENTER OF GENERAL ELECTRIC.

WE WILL CONTINUE THIS PROCESS OF REVIEWING OPERATING INFORMATION DURING THE OPERATING LIFE OF FERMI 2. OF COURSE, WE WILL ADD THE REVIEW OF OPERATING EXPERIENCE AT FERMI 2. ALL FERMI 2 LER'S WILL BE REVIEWED BY THE ON-SITE REVIEW ORGANIZATION (OSRO).

EDISON MANAGEMENT HAS APPOINTED AN LER COORDINATOR WHO IS INDEPENDENT OF THE PLANT OSRO ORGANIZATION, AND HE WILL ALSO REVIEW EACH LER AND DETERMINE IF AN INDEPENDENT ANALYSIS IS REQUIRED. THE OSRO WILL APPROVE ALL CHANGES IN NUCLEAR PRODUCTION PROCEDURES AND RECOMMEND CHANGES IN PLANT DESIGN. THE OSRO WILL ALSO REQUEST REVIEWS OF SIGNIFICANT LER'S BY THE SAFETY AND PERFORMANCE ENGINEERING GROUP (SPEG). THE SPEG WILL REVIEW AND APPROVE ALL PLANT DESIGN CHANGES AND WILL, AT THE REQUEST OF OSRO, REVIEW LER'S REFERRED TO THEM AND ANY LER'S THAT LEADS TO PROCEDURAL CHANGES THAT INVOLVE AN UNREVIEWED SAFETY QUESTION.

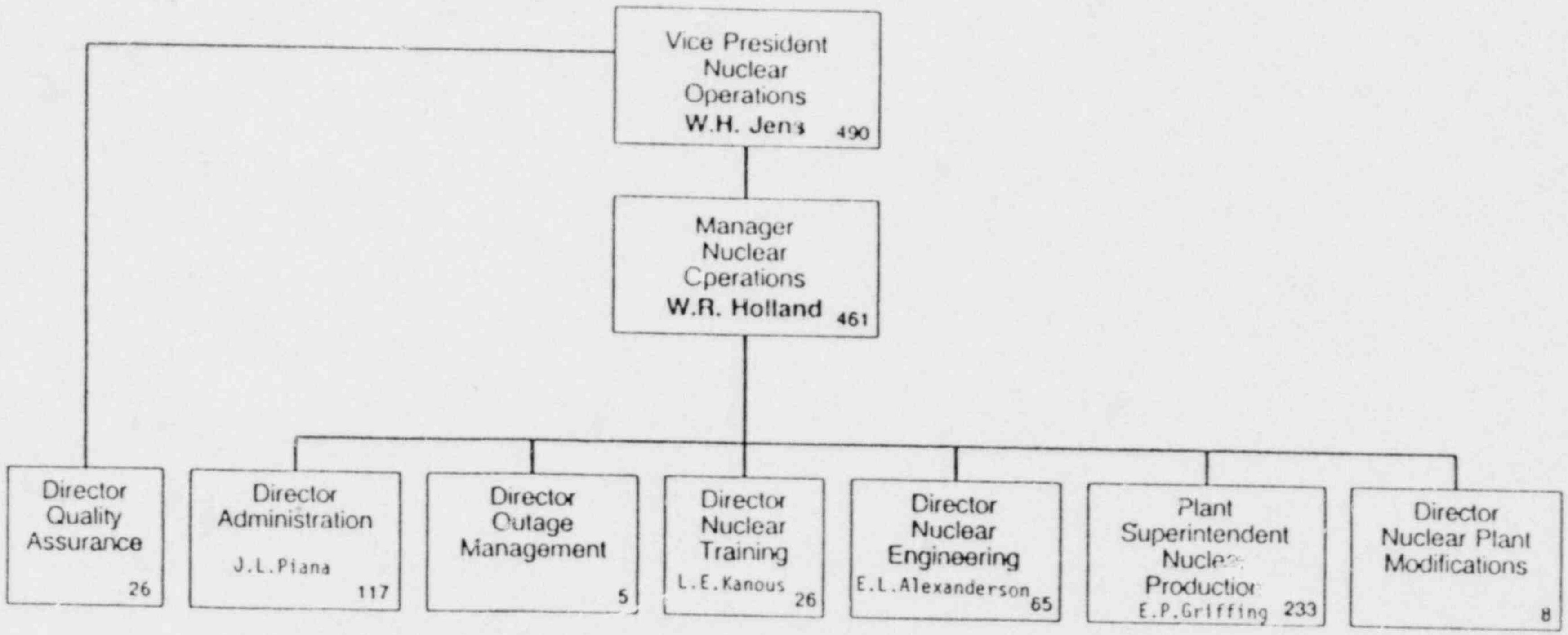
FOLLOWING THESE REVIEWS AND APPROVALS OF FERMI 2 LER'S THE RESULTING REPORTS WILL BE ROUTED TO THE SHIFT SUPERVISORS AND SHIFT TECHNICAL ADVISORS FOR INFORMATION. ALL REVIEWS OF EXTERNAL OPERATING EXPERIENCE THAT LEAD TO RECOMMENDED CHANGES IN DESIGN OR PROCEDURES WILL ALSO BE ROUTED TO THE SHIFT SUPERVISORS AND SHIFT TECHNICAL ADVISORS. THE LER COORDINATOR WILL ALSO BE RESPONSIBLE FOR DIRECTING THE OPERATIONAL EXPERIENCE REPORTS TO THE PROPER ORGANIZATIONAL UNITS FOR REVIEW SUCH AS OPERATIONS, MAINTENANCE, RADIATION PROTECTION, TRAINING, DESIGN, ETC. COPIES OF THE REVIEWS WILL

BE SENT TO THE AFFECTED ORGANIZATIONAL UNITS, AND THEY ARE OBLIGATED TO COMMENT IF THEY DISAGREE WITH THE CONCLUSIONS OR RECOMMENDATIONS. IF THE LER COORDINATOR CANNOT RESOLVE THESE DIFFERENCES, THE ISSUE WILL BE RESOLVED AT HIGHER LEVELS IN THE ORGANIZATION WITH APPEALS POSSIBLE TO THE VICE PRESIDENT-NUCLEAR OPERATIONS. FIGURE 6 INDICATES THE ORGANIZATIONAL ARRANGEMENT FOR SAFETY REVIEWS AND ANALYSIS THAT HAVE BEEN PREVIOUSLY DESCRIBED. FIGURE 6 INDICATES THAT PREVIOUSLY UNREVIEWED SAFETY ISSUES, AFTER ANALYSIS AND APPROVAL BY SPEG, ARE REVIEWED BY THE INDEPENDENT REVIEW AND AUDIT GROUP (IRAG), CONSISTING OF COMMITTEE MEMBERS WHO ARE EXPERTS FROM WITHIN AND OUTSIDE THE COMPANY. THE IRAG IS CHAIRED BY THE DIRECTOR-NUCLEAR ENGINEERING, AND REPORTS DIRECTLY TO THE VICE PRESIDENT-NUCLEAR OPERATIONS.

IT IS OUR PRESENT INTENTION TO HAVE A SUBCOMMITTEE OF THE BOARD OF DIRECTORS OF DETROIT EDISON ESTABLISH THE OVERALL NUCLEAR SAFETY PHILOSOPHY OF THE COMPANY AND TO REVIEW THE SIGNIFICANT EVENTS, TRENDS, AND OPERATING EXPERIENCE OF NUCLEAR OPERATIONS.

Nuclear Operations Organization

FIGURE 1.



*Not including 74 security and medical personnel

FIGURE 2.

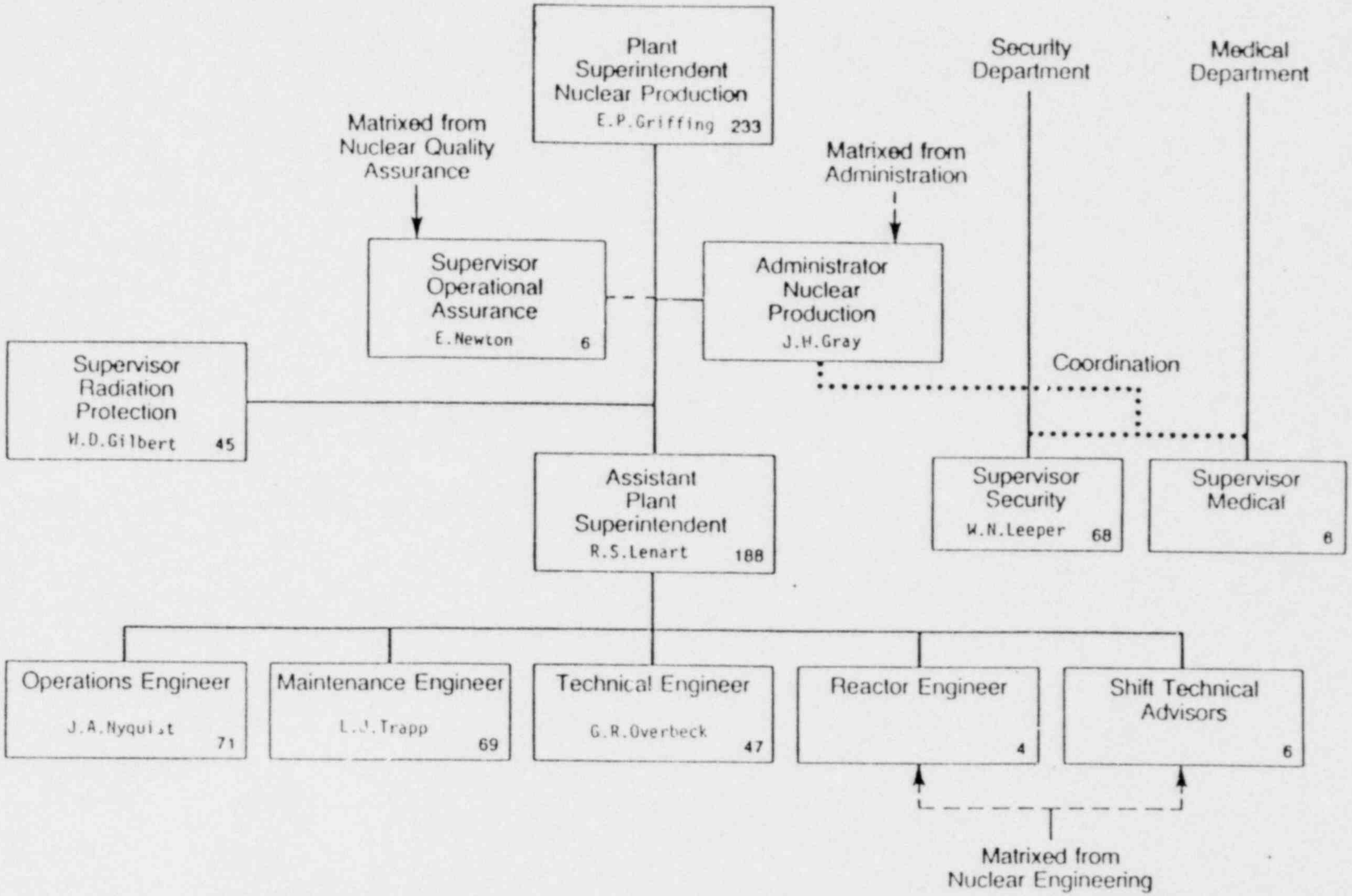

Plant Superintendent Nuclear Production
E.P.Griffing 233
Security Department
Medical Department
Matrixed from Nuclear Quality Assurance
Matrixed from Administration
Supervisor Operational Assurance
E.Newton 6
Administrator Nuclear Production
J.H.Gray
Coordination
Supervisor Radiation Protection
W.D.Gilbert 45
Assistant Plant Superintendent
R.S.Lenart 188
Supervisor Security
W.N.Leeper 68
Supervisor Medical
6
Operations Engineer
J.A.Nyquist 71
Maintenance Engineer
L.J.Trapp 69
Technical Engineer
G.R.Overbeck 47
Reactor Engineer
4
Shift Technical Advisors
6
Matrixed from Nuclear Engineering

FIGURE 3.

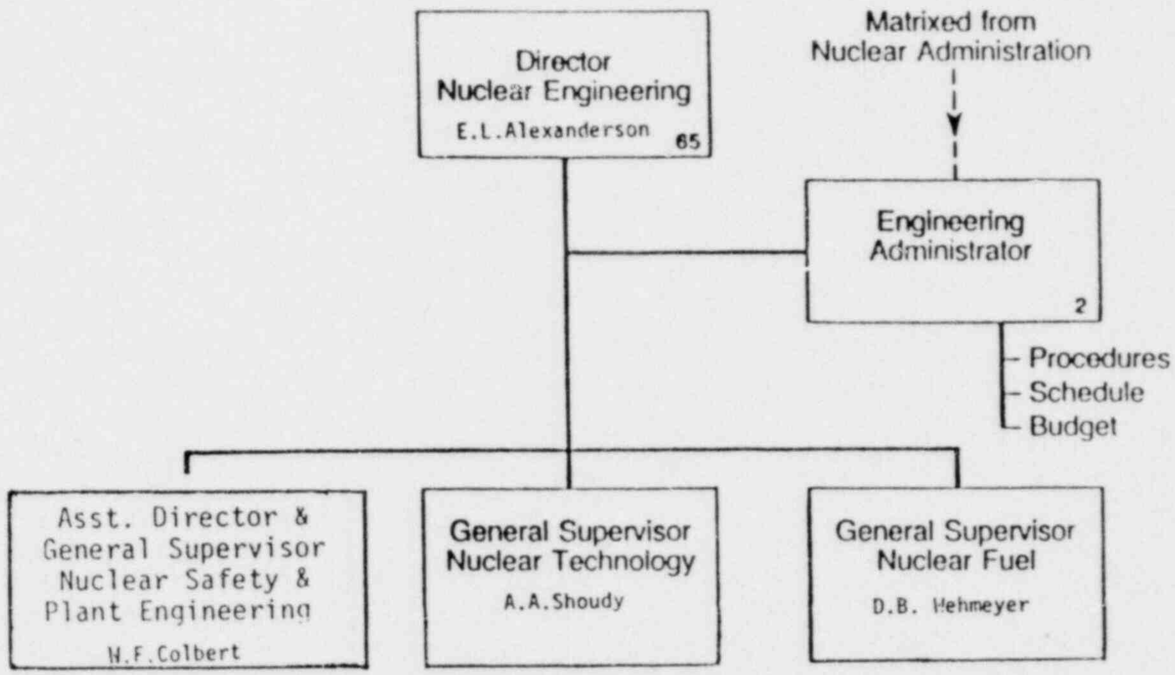
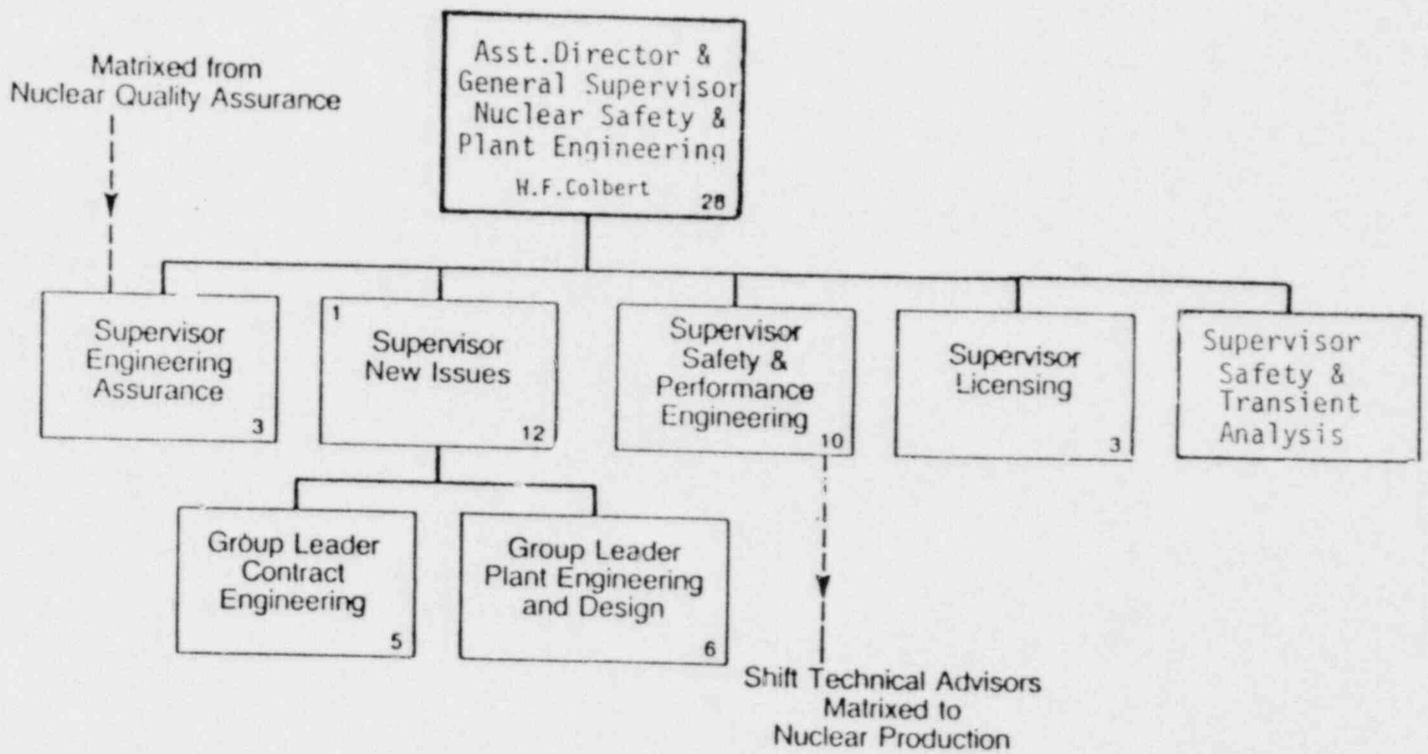

Director
Nuclear Engineering
E.L.Alexanderson
65
Matrixed from
Nuclear Administration
Engineering
Administrator
2
Procedures
Schedule
Budget
Asst. Director &
General Supervisor
Nuclear Safety &
Plant Engineering
W.F.Colbert
General Supervisor
Nuclear Technology
A.A.Shoudy
General Supervisor
Nuclear Fuel
D.B. Wehmeyer

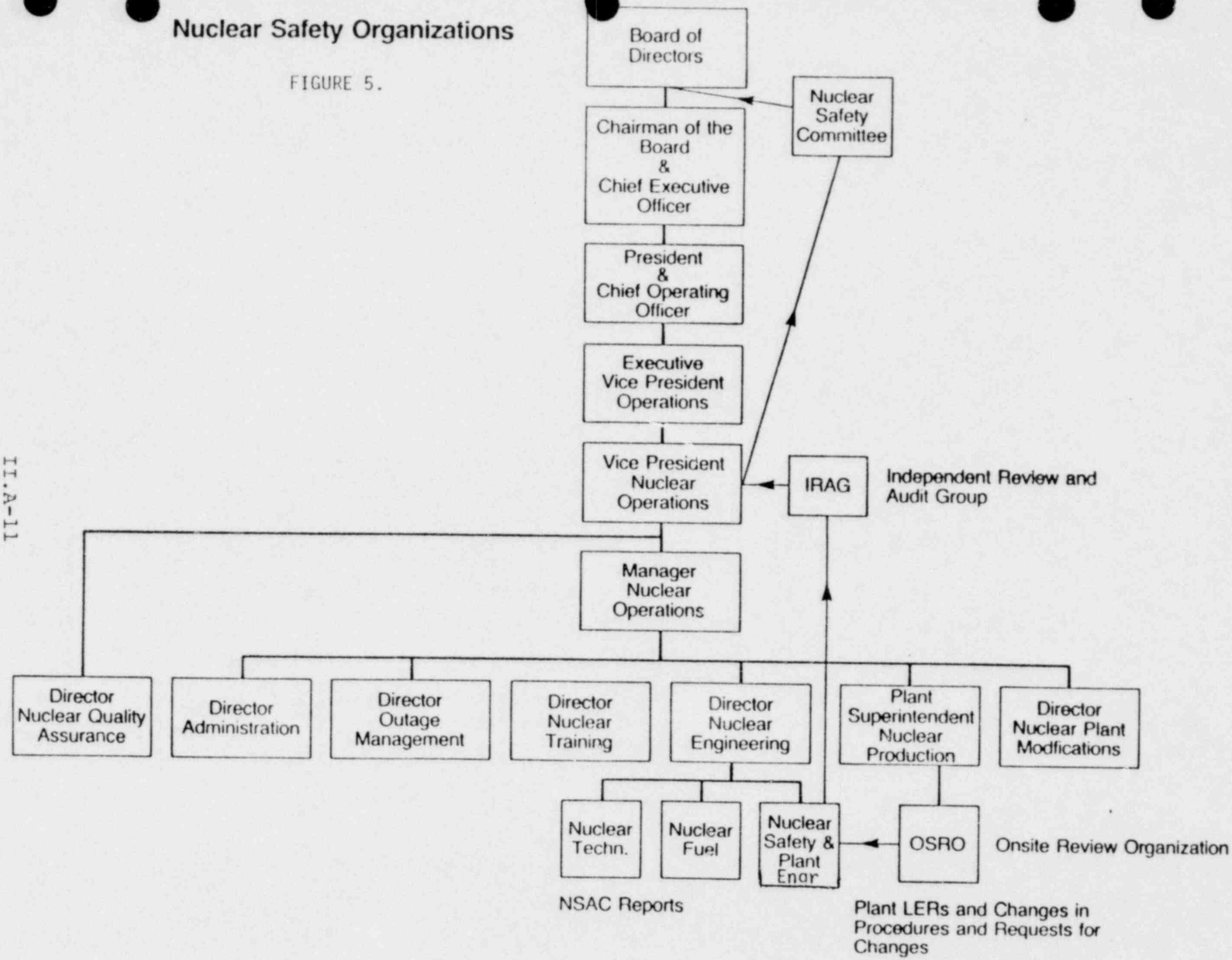
FIGURE 4.



Asst.Director & General Supervisor Nuclear Safety & Plant Engineering
W.F.Colbert
28
Matrixed from Nuclear Quality Assurance
Supervisor Engineering Assurance
3
1
Supervisor New Issues
12
Supervisor Safety & Performance Engineering
10
Supervisor Licensing
3
Supervisor Safety & Transient Analysis
Group Leader Contract Engineering
5
Group Leader Plant Engineering and Design
6
Shift Technical Advisors Matrixed to Nuclear Production

Nuclear Safety Organizations

FIGURE 5.



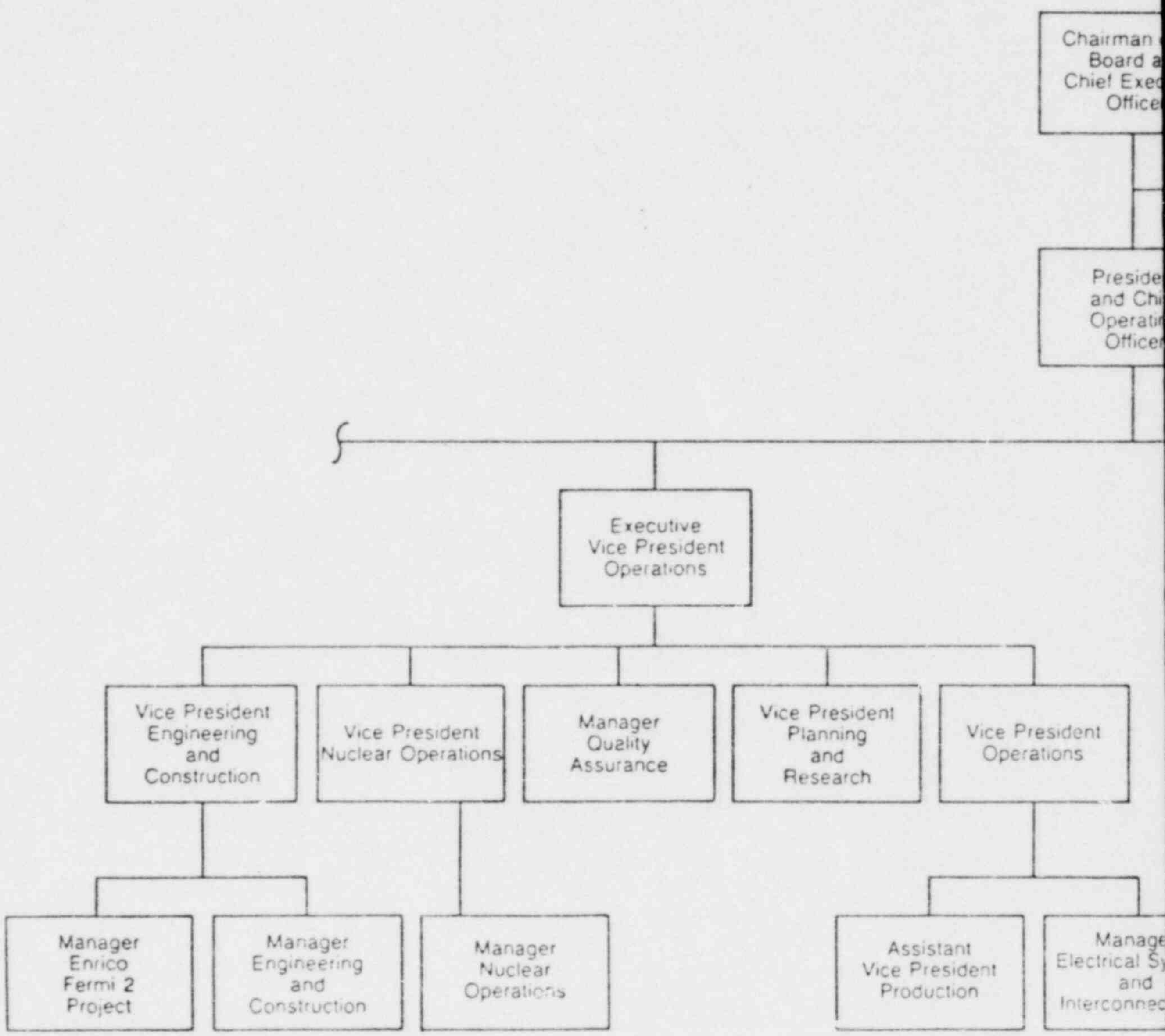
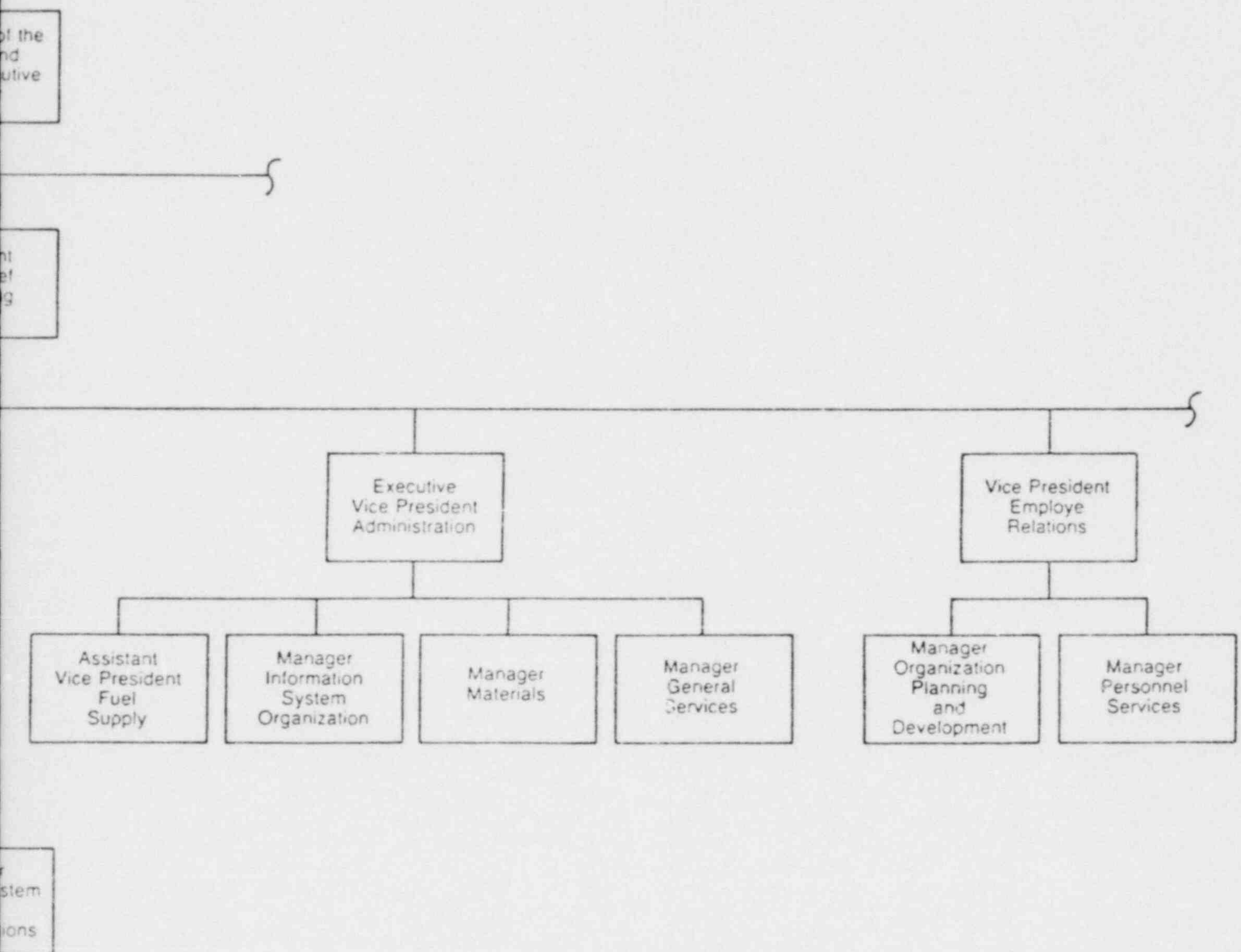

Chairman
Board a
Chief Exec
Office
Preside
and Chi
Operati
Officer
Executive Vice President Operations
Vice President Engineering and Construction
Vice President Nuclear Operations
Manager Quality Assurance
Vice President Planning and Research
Vice President Operations
Manager Enrico Fermi 2 Project
Manager Engineering and Construction
Manager Nuclear Operations
Assistant Vice President Production
Manage
Electrical Sy
and
Interconnec

FIGURE 6.



Executive Vice President Administration
Vice President Employe Relations
Assistant Vice President Fuel Supply
Manager Information System Organization
Manager Materials
Manager General Services
Manager Organization Planning and Development
Manager Personnel Services

USE OF SIMULATORS IN TRAINING PROGRAMS

TRAINING WITHIN DETROIT EDISON (ENRICO FERMI 2) CURRENTLY INCLUDES FOUR (4) MAJOR SIMULATOR PROGRAMS. UP TO THIS POINT, ALL TRAINING HAS BEEN ACCOMPLISHED AT THE BROWN'S FERRY SIMULATOR, UNDER PROGRAMS PRESENTED BY GENERAL PHYSICS, INC.

THE FIRST PROGRAM IS THE STANDARD GENERAL PHYSICS 12-WEEK BWR OPERATOR CERTIFICATION TRAINING WHICH ALL LICENSE CANDIDATES MUST ATTEND PRIOR TO APPLYING FOR A LICENSE. THE STUDENT BASE FOR THIS PROGRAM HAS BEEN EXPANDED TO INCLUDE SELECTED PLANT ENGINEERS. UTILIZATION OF THE SIMULATOR IN THIS MANNER HAS DEVELOPED, WITHIN THE ENGINEERING DIVISION, A KNOWLEDGE OF THE NEEDS OF THE OPERATING STAFF.

THE SECOND PROGRAM IS A TWO-WEEK REFRESHER TRAINING COURSE FOR LICENSE CANDIDATES. THIS COURSE HAS BEEN EXPANDED FROM THE ORIGINALLY SCHEDULED ONE-WEEK TO ALLOW FOR A GREATER REFAMILIARIZATION WITH OPERATIONS TECHNIQUES AND RESPONSES, AND WILL BE PRESENTED PRIOR TO FUEL LOAD AT ENRICO FERMI 2. THE COURSE WILL RELATE SPECIFICALLY TO ENRICO FERMI 2 THROUGH SUPPLEMENTARY PRESENTATIONS TO THE SIMULATOR EXERCISES. THE SECOND WEEK OF INSTRUCTION WILL PROVIDE FOR ACCIDENT SIMULATIONS, SINCE DEVELOPED, FROM TMI LESSONS LEARNED.

THE THIRD PROGRAM IS BEING DEVELOPED TO SUPPORT THE TRAINING OF SHIFT TECHNICAL ADVISORS (STA'S). THE LENGTH OF THIS PROGRAM IS NOT YET SPECIFIED. EMPHASIS WILL BE PLACED UPON ENSURING THAT THE STA HAS RECEIVED TRAINING IN THE EVOLUTIONS AND ABNORMAL OCCURRENCES REFERENCED IN NUREG-0737. TRAINING WILL CONCENTRATE ON THE STA'S ROLES OF ASSESSING THE PLANT CONDITIONS, ACTING AS A TECHNICAL RESOURCE FOR THE OPERATING CREW, AND BEING AN INFORMATION RESOURCE FOR PLANT SUPPORT GROUPS (E.G., INDEPENDENT SAFETY AND EVALUATION GROUP). IT WILL STRESS THE IMPORTANCE OF KEEPING THE "BIG PICTURE" IN MIND, NOT GETTING SIDETRACKED INTO LITTLE PROBLEM AREAS, WHILE USING ALL AVAILABLE INPUTS TO ANALYZE WHAT IS HAPPENING.

THE FOURTH TRAINING PROGRAM IS THAT ASSOCIATED WITH REQUALIFICATION TRAINING FOR OPERATORS AND STA'S. EMPHASIS SHALL BE PLACED UPON THE TEAM CONCEPT OF TRAINING. AS MUCH AS IS PRACTICAL, THIS TRAINING SHALL BE GIVEN TO INTEGRAL SHIFTS TO REFRESH AND REFINE THEIR CAPABILITIES TO RESPOND AS A COHESIVE UNIT TO NORMAL, TRANSIENT, AND EMERGENCY CONDITIONS. DEVELOPMENT IS CURRENTLY ONGOING TO ARRIVE AT THIS GOAL.

FUTURE UTILIZATION OF SIMULATOR TRAINING WILL CENTER AROUND THE SITE SPECIFIC SIMULATOR FOR WHICH DETROIT EDISON HAS CONTRACTED. IT WILL BE LOCATED IN CLOSE PROXIMITY TO THE ENRICO FERMI 2 SITE, AT THE NUCLEAR OPERATIONS CENTER. SCHEDULED DELIVERY DATE IS LATE 1983. ALL CONTROL ROOM PANELS, SWITCHES, AND COMPUTER CAPABILITIES WILL BE AVAILABLE; THE STANDARDS OF ANSI 3.5 WILL BE MET, OR EXCEEDED.

DETROIT EDISON HAS BEEN UTILIZING SIMULATIONS AND DYNAMIC "PART-TASK TRAINERS" AS A TRAINING VEHICLE FOR MANY YEARS AND IS FIRMLY CONVINCED OF THEIR VIABILITY AS A TRAINING TOOL. IN THIS REGARD, FUTURE SIMULATOR UTILIZATION IS ANTICIPATED TO DEVELOP FAR BEYOND THAT PRESENTLY EMPLOYED.

IN THE AREA OF OPERATOR TRAINING, SIMULATOR EXERCISES ARE PLANNED FOR BOTH LICENSED AND NON-LICENSED OPERATORS. THIS WILL ALLOW FOR A GREATER APPRECIATION, WITHIN THE OPERATING SHIFT, OF THE IMPORTANCE OF EACH MEMBER'S ACTIONS. CARRYING THIS PHILOSOPHY A STEP FURTHER, SIMULATOR TRAINING WILL UTILIZE PLANS AND TRAINING MATERIALS TO SUPPORT THE CONCEPT OF "TEAM TRAINING" A COMPLETE SHIFT OF OPERATORS AS AN INTEGRATED UNIT.

ADDITIONAL UTILIZATION OF SIMULATOR TRAINING IS BEING CONTRACTED TO BE DEVELOPED FOR SUPPORT AND MANAGERIAL ACTIVITIES. TASKS FOR THE MAINTENANCE AND TECHNICAL STAFF, I&C PERSONNEL, AND MANAGERS AND SUPERVISORS ARE PLANNED. THE SIMULATOR HAS CABLING TO CLASSROOM ENVIRONMENTS WHICH WILL BE UTILIZED TO DEVELOP AND SUSTAIN AND EXPERTISE IN ALL PERSONNEL WITH AN IDENTIFIED ROLE TO EXPEDITIOUSLY AND SAFELY CONDUCT BOTH NORMAL AND EMERGENCY OPERATIONS. THIS WILL INCLUDE TASKS RANGING FROM THE INTERRELATABILITY OF TYPICAL MAINTENANCE FUNCTIONS AND THE CONTROL ROOM, TO SIMULATION INPUT ASSOCIATED WITH EMERGENCY PLAN EXERCISES AND DRILLS.

DETROIT EDISON WILL CONTINUE TO USE SIMULATORS AS A TRAINING VEHICLE IN MAINTAINING THE HIGHEST QUALITY WORK FORCE.

WE ARE COMMITTED TO PROVIDING THE BEST SIMULATOR TRAINING AVAILABLE.

TRAINING FOR SERIOUS ACCIDENTS BEYOND D.B.A.

IN ACCORDANCE WITH NUREG-0737, NUREG-0660, APPENDIX H OF THE ENRICO FERMI 2 FSAR, AND THE MARCH 28, 1980 NRC STAFF DIRECTIVE (DENTON LETTER), DETROIT EDISON HAS ESTABLISHED A PROGRAM TO TRAIN ITS PERSONNEL IN CONTROLLING AND MITIGATING ACCIDENTS BEYOND THE DESIGN BASED ACCIDENT (DBA). THIS PROGRAM WILL MEET BOTH THE INTENT AND REQUIREMENTS OF THE ABOVE DIRECTIVES, AND THE RECOMMENDATIONS OF THE INSTITUTE OF NUCLEAR POWER OPERATION (INPO) DOCUMENT TITLED "TRAINING GUIDELINES FOR RECOGNIZING AND MITIGATING THE CONSEQUENCES OF SEVERVE CORE DAMAGE", DATED JUNE 30, 1980.

THE TABLE SHOWN PRESENTS A TOPICAL SUMMARY OF THE TRAINING TO BE REPRESENTED IN THE DETROIT EDISON TRAINING PROGRAM. CONTACT HOURS OF INSTRUCTION RECEIVED IS NOT LISTED. IT IS THE DETROIT EDISON VIEWPOINT THAT KNOWLEDGE ATTAINMENT/RETENTION IS THE KEY TO THE VALIDITY OF A TRAINING CURRICULUM. THEREFORE, OUR COURSE DESIGNS WILL BE SUCH TO INSURE THAT THE PERSONNEL HAVE ATTAINED THE LEVEL OF EXPERTISE TO BE ABLE TO APPROPRIATELY RECOGNIZE AND RESPOND, IRRESPECTIVE OF THE TIME NEEDED TO REACH THAT LEVEL.

DETROIT EDISON'S LICENSED OPERATOR TRAINING PROGRAM HAS RECEIVED EXTENSIVE MODIFICATION AND AUGMENTATION SINCE THREE MILE ISLAND TO MEET THE PRESENT GUIDELINES. TRAINING IN INFREQUENT, ABNORMAL, AND EMERGENCY PROCEDURES HAS BEEN MODIFIED TO EMPHASIZE THE BIG PICTURE, AND OPERATOR UTILIZATION OF ALL INPUT DATA FOR MAINTAINING THE CORE COVERED AND THE CONTAINMENT INTACT. TRAINING IN PROCESS COMPUTER OPERATIONS IS BEING EXPANDED TO INCLUDE USE OF THE COMPUTER DURING DEGRADED CORE CONDITIONS FOR ATTAINING AND ANALYZING INFORMATION DESCRIBING THE ACTUAL CORE STATUS.

NEW COURSES HAVE BEEN PLACED INTO THE PROGRAM TO INSURE THAT ALL OPERATORS HAVE THE KNOWLEDGE TO PROPERLY EVALUATE AND RESPOND TO THESE EMERGENCY CONDITIONS. THEORETICAL TRAINING, WITH PRACTICAL APPLICATIONS, IN THERMODYNAMICS AND HEAT TRANSFER HAS BEEN IDENTIFIED AS NECESSARY. LIKEWISE, A COURSE IN MITIGATION OF CORE DAMAGE HAS BEEN INCLUDED. AT THIS TIME, DETROIT EDISON IS EVALUATING THE CAPABILITIES OF VARIOUS TRAINING ORGANIZATIONS TO SUPPORT OUR TRAINING OBJECTIVES IN THIS AREA. ESTABLISHMENT OF A FINAL CURRICULUM HAS NOT BEEN COMPLETED. HOWEVER, ALL APPROPRIATE SUBJECTS TO INCLUDE UTILIZATION OF INCORE AND EXCORE INSTRUMENTATION, CORE COOLING MECHANICS, PRIMARY CHEMISTRY, RADIATION MONITORING, GAS GENERATION AND HAZARDS, AND RECRITICALITY POTENTIAL WILL BE COVERED.

TRAINING IN THIS AREA WILL BE GIVEN, AS APPROPRIATE, TO ALL OPERATING PERSONNEL AT ENRICO FERMI 2, FROM THE PLANT SUPERINTENDENT TO THE LICENSED OPERATOR. SHIFT TECHNICAL ADVISORS (STA'S) HAVE BEEN ESTABLISHED TO SUPPLEMENT THE OPERATING CREWS' CAPABILITIES DURING ABNORMAL CONDITIONS. THEIR TRAINING WILL INCLUDE ALL OF THE ABOVE MENTIONED CURRICULUM, AND BE EXPANDED TO GREATER DEPTH. TRAINING FOR INSTRUMENT MAINTENANCE AND RAD-CHEM PERSONNEL WILL BE CONDUCTED ON BOTH A THEORETICAL AND PRACTICAL LEVEL FOR THE ACCOMPLISHMENT OF HIGH RADIATION SAMPLING.

FINALLY, ALL PLANT PERSONNEL WILL BE TRAINED TO A DEGREE COMMENSURATE WITH THEIR DUTIES ON EMERGENCY PLAN IMPLEMENTATION. THIS WILL INCLUDE SUCH TOPICS AS ACTIVATING THE ONSITE TECHNICAL SUPPORT CENTER AND THE OFFSITE EMERGENCY OPERATIONS FACILITY, ALERT NOTIFICATION PROCEDURES AND RESPONSIBILITIES, EVACUATION PROCEDURES, AND SUPPORT PROCEDURES.

IN SUMMARY, WE BELIEVE WE SATISFY THE CRITERIA FOR TRAINING FOR SERIOUS ACCIDENTS BEYOND D.B.A.

SERIOUS ACCIDENTS BEYOND D.B.A.

TOPICS

1. CORE COOLING MECHANICS
2. POTENTIALLY DAMAGING OPERATING CONDITIONS
3. GAS/STEAM BINDING ON CORE COOLING
4. RECOGNIZING CORE DAMAGE
5. HYDROGEN HAZARDS DURING ACCIDENT CONDITIONS
6. MONITORING CRITICAL PARAMETERS DURING ACCIDENT CONDITIONS

    INCORE INSTRUMENTATION

    EXCORE INSTRUMENTATION

    PROCESS COMPUTER

    HIGH RADIATION SAMPLING

7. RADIATION HAZARDS AND RADIATION MONITOR RESPONSE
8. CRITERIA FOR OPERATION AND COOLING MODE SELECTION
9. INFREQUENT ABNORMAL AND EMERGENCY OPERATING PROCEDURES
10. THERMODYNAMICS AND HEAT TRANSFER
11. RECRITICALITY POTENTIAL
12. EMERGENCY PLAN

SELECTION AND TRAINING OF MAINTENANCE PERSONNEL

THE DETROIT EDISON COMPANY IS FIRMLY COMMITTED TO THE CONCEPT OF MAXIMIZING THE SAFETY OF NUCLEAR POWER PLANTS THROUGH THE USE OF WELL-TRAINED PERSONNEL. ALL CRAFT WORKERS ASSIGNED TO THE MAINTENANCE DEPARTMENT AT THE ENRICO FERMI ATOMIC POWER PLANT UNIT 2 (EF2) WILL BE JOURNEYMEN. OVER A DECADE AGO, THE DETROIT EDISON COMPANY INSTITUTED A PROGRAM IN TRAINING MAINTENANCE CRAFT WORKERS TO BECOME GENERAL MAINTENANCE JOURNEYMEN (GMJ). DEPARTING FROM THE TRADITIONAL ONE-MAN, ONE-CRAFT CONCEPT, DETROIT EDISON REALIZED THAT INCREASING JOB SCOPE TO INCLUDE MORE THAN ONE CRAFT WOULD BENEFIT BOTH THE WORKER AND THE COMPANY. THE INCREASED VARIETY OF SKILLS THAT THE WORKER WAS CALLED UPON TO PERFORM WOULD INCREASE JOB SATISFACTION, WHICH WOULD LEAD TO HIGHER QUALITY WORK. THIS WOULD OBVIOUSLY BENEFIT THE COMPANY, BUT ALSO THE INCREASED DIVERSITY OF SKILLS WOULD BENEFIT THE COMPANY IN MANPOWER UTILIZATION.

EACH GMJ IS TRAINED IN TWO SKILLS, A PRIMARY SKILL AND A SECONDARY SKILL. THE GMJ IS ABLE TO PERFORM AND LEAD OTHER WORKERS IN HIS PRIMARY SKILL, IS ALSO COMPLETELY TRAINED TO PERFORM ANY TASK RELATED TO HIS SECONDARY SKILL AND FURTHERMORE IS ABLE TO ASSIST IN ANY CRAFT. THE TRAINING PROGRAM TO PRODUCE A GMJ IS A STATE-OF-THE-ART PROGRAM. BASED ON TASK ANALYSIS AND CLOSELY CRITERION-REFERENCED, THE PROGRAM HAS

BEEN EXTREMELY SUCCESSFUL SINCE ITS INCEPTION. THE GMJ APPRENTICE TRAINING PROGRAM IS MODULAR IN FORMAT TO ALLOW FOR DIFFERENT ENTRY LEVELS AND IS PERFORMANCE BASED. THE TRAINEE IS ALLOWED TO PROGRESS THROUGH EACH TRAINING MODULE AT HIS/HER OWN PACE WITHIN A DEFINED SCHEDULE WHICH IS TIED TO INCREMENTS IN PAY. WHEN THE TRAINEE ASCERTAINS THAT HE/SHE HAS COMPLETED THE OBJECTIVES OF THE MODULE, THEY ARE GIVEN A FORMAL EVALUATION BY THE TRAINING AND/OR SUPERVISORY STAFF. THIS EVALUATION REQUIRES THAT THE TRAINEE DEMONSTRATE THE SKILLS LEARNED AND IS EVALUATED AGAINST MEASURABLE, OBSERVABLE CRITERIA. THE TRAINEE IS NOT ALLOWED TO CONTINUE TO MORE ADVANCED MODULES UNTIL HE/SHE HAS DEMONSTRATED MASTERY OF THE PREVIOUS MODULE. INTEGRAL WITH THE CLASSROOM PORTION OF THE TRAINING PROGRAM IS EXTENSIVE PLANNED, ORGANIZED AND EVALUATED ON-THE-JOB EXPERIENCE, APPLYING ALL OF THE SKILLS PREVIOUSLY LEARNED.

ONCE THE INDIVIDUAL HAS COMPLETED THE GMJ PROGRAM, HE/SHE IS ALLOWED TO BID ON A POSITION AT EF2. THE BIDDING SYSTEM USED IS A CONSTRAINED SYSTEM; NOT ALL BIDDERS ARE ACCEPTED. IN ORDER TO BE ACCEPTED FOR A MAINTENANCE POSITION AT EF2, THE INDIVIDUAL MUST NOT ONLY SHOW AN INTEREST IN THE POSITION, BUT ALSO MUST HAVE DEMONSTRATED HIGH ACHIEVEMENT IN HIS/HER PREVIOUS TRAINING AND EXPERIENCE.

EACH GMJ WHO IS ACCEPTED AT EF2 THEN BEGINS A SECOND PHASE OF TRAINING TO PROVIDE HIM/HER WITH THE ADDITIONAL SKILLS AND ABILITIES NECESSARY TO WORK AT A NUCLEAR POWER PLANT. AFTER COMPLETING TRAINING PROGRAMS TO SHOW THE GJM HOW HE/SHE FITS INTO THE "BIG PICTURE", TRAINING BEGINS IN EACH OF THE FOLLOWING FOUR AREAS CONCURRENTLY.

MANY OF THE SKILLS THAT THE GMJ ALREADY POSSESS WILL BE USED AT THE NUCLEAR PLANT, BUT APPLIED IN NEW WAYS. GMJ PIPEFITTERS, FOR EXAMPLE, ARE WELL TRAINED IN REPACKING VALVES, BUT AT EF2 THEY WILL HAVE TO BE INSTRUCTED IN HOW TO PERFORM THIS TASK IN A GLOVE BOX.

AS FERMI 2 WILL BE THE ONLY OPERATING NUCLEAR POWER PLANT IN THE EDISON SYSTEM, THE GMJ WILL BE REQUIRED TO WORK ON TYPES OF EQUIPMENT THAT WERE NOT ADDRESSED IN THEIR APPRENTICE TRAINING PROGRAM, PERHAPS WITH TOOLS THAT ARE NOT USED IN OTHER POWER PLANTS. PROGRAMS TO ENSURE PROFICIENCY IN THESE TASKS WILL BE GIVEN TO THESE WORKERS.

WORKING AT A NUCLEAR POWER PLANT IS A DIFFERENT ENVIRONMENT FROM THAT OF A FOSSIL PLANT. NEW SKILLS NEEDED TO ENSURE THE SAFETY (BOTH RADIATION AND INDUSTRIAL) OF THE GMJ AND OTHER PERSONNEL WILL BE REQUIRED. EACH GMJ WILL BE TRAINED TO BE PROFICIENT IN DONNING AND REMOVING PROTECTIVE CLOTHING, SPILL PREVENTION, CLEANUP AND DECONTAMINATION, JUST TO NAME A FEW.

TO ENSURE THAT THE GMJ DEVELOP THE PROPER WORKING HABITS AND AN UNDERSTANDING OF THE NUCLEAR ENVIRONMENT, SPECIFIC PROGRAMS ADDRESSING THE DISCIPLINED NATURE OF NUCLEAR WORK ARE BEING DEVELOPED. NUCLEAR CODES AND STANDARDS, PLANT AND MAINTENANCE ADMINISTRATIVE PROCEDURES, QA/QC INDOCTRINATION, HOUSEKEEPING AND SECURITY PLAN TRAINING ARE EXAMPLES OF SUCH PROGRAMS.

THESE NUCLEAR SPECIFIC TRAINING PROGRAMS ARE BEING DEVELOPED USING THE SAME HIGH STANDARDS USED IN THE GMJ APPRENTICE TRAINING PROGRAM. TASK ANALYSES, CRITERION-REFERENCING AND FORMALIZED PROFICIENCY DEMONSTRATION EVALUATIONS WILL BE EXTENSIVELY USED.

ATTACHMENT 1 DETAILS THE TYPES OF TRAINING PROGRAMS THAT WILL BE USED FOR EACH JOB TITLE IN THE MAINTENANCE DEPARTMENT.

MAINTENANCE WORKER TRAINING

ALL WORKERS WILL BE JOURNEYMEN

GENERAL MAINTENANCE JOURNEYMAN PROGRAM

- INCREASED JOB SCOPE ⟶ INCREASED JOB SATISFACTION
- FACILITATE MANPOWER UTILIZATION
- PRIMARY VS SECONDARY SKILL

GMJ APPRENTICE TRAINING PROGRAM

- BASED ON TASK ANALYSIS
- MODULAR FORMAT
- SELF-PACED, BUT TIED TO PAY INCREASES
- PERFORMANCE BASED EVALUATION

ENRICO FERMI MAINTENANCE SELECTION PROCESS

- CONSTRAINED BIDDING SYSTEM
- GMJ MUST <u>WANT</u> TO WORK AT FERMI
- GMJ MUST BE HIGH PERFORMER

ENRICO FERMI SPECIFIC TRAINING

"BIG PICTURE"

- NEW APPLICATION OF OLD SKILLS
- NEW TOOLS AND EQUIPMENT
- NEW SKILLS DUE TO NUCLEAR ENVIRONMENT
- DEVELOPMENT OF PROPER WORKING HABITS

ATTACHMENT 1
TRAINING PRESCRIPTION

	GMJ	Welding Engineer	Shop Welding Foreman	Field Welding Foreman	Maintenance Foreman-Plant	Asst. General Foreman	General Foreman	Senior Engineering Technician	Engineer	Senior Engineer	Asst. Maintenance Engineer	Maintenance Engineer
General Employe	●	●	●	●	●	●	●	●	●	●	●	●
Emergency Plan	●	●	●	●	●	●	●	●	●	●	●	●
Security Plan	●	●	●	●	●	●	●	●	●	●	●	●
Fire Protection Plan	●	●	●	●	●	●	●	●	●	●	●	●
First Aid	●	●	●	●	●	●	●	●	●	●	●	●
Cardiopulmonary Resucitation	●	●	●	●	●	●	●	●	●	●	●	●
Plant Administrative Procedures	●	●	●	●	●	●	●	●	●	●	●	●
Maintenance Administrative Procedures	●	●	●	●	●	●	●	●	●	●	●	●
QA/QC Indoctrination	●	●	●	●	●	●	●	●	●	●	●	●
Nuclear Mechanical Seals	●				●	●						
Mobile Equipment Operation	●	●	●	●	●	●	●					
Nuclear Precision Tools	●	●	●	●	●	●	●					
Safety Tagging	●	●	●	●	●	●	●	●	●	●	●	●
Cutting & Burning	●	●	●	●	●							
MSIV Maintenance					●	●						
Safety Relief Valve Maintenance					●	●						
Recirc. Pump Seal Maintenance					●	●						
CRD Hydraulic System Maintenance					●	●						
Emergency Diesel Generator Maintenance	●				●	●	●					
Dexter Valve Reseater Operation	●				●	●						
Wacks Cutter Operation	●				●	●						
Hydraulic Torquing Eq. Operation	●				●	●						
Limitorque Valve Actuator Maintenance	●				●	●						
Powell Valve Maintenance	●				●	●						
Rototorque Valve Maintenance	●				●	●						
Generic Systems and Procedures	●	●	●	●	●	●	●	●	●	●	●	●
Respirator Use	●	●	●	●	●	●	●	●	●	●	●	●
Radiation Worker Skills	●	●	●	●	●	●	●	●	●	●	●	●
Decontamination Skills	●	●	●	●	●	●	●	●	●	●	●	●
Housekeeping	●	●	●	●	●	●	●	●	●	●	●	●
Nuclear Codes and Standards	●	●	●	●	●	●	●	●	●	●	●	●
Tool Calibration	●	●	●	●	●	●	●					
Visual NDE Examination	●	●	●	●	●	●	●					
Visible Dye Penetrant Examination	●	●	●	●								

DETROIT EDISON SIMULATOR PROGRAMS

PRESENT PROGRAMS

- O OPERATOR CERTIFICATION
- O OPERATOR REFRESHER
- O SHIFT TECHNICAL ADVISOR
- O REQUALIFICATION

FUTURE PROGRAMS

- O OPERATOR CERTIFICATION
- O SHIFT TECHNICAL ADVISOR
- O REQUALIFICATION
- O NON LICENSED OPERATORS TASKS
- O MAINTENANCE PERSONNEL TASKS
- O I & C PERSONNEL TASKS
- O MANAGEMENT/SUPERVISOR TASKS
- O EMERGENCY PLAN DRILL/EXERCISE INTERFACE

SERIOUS ACCIDENTS BEYOND D. B. A.

TOPICS

1. CORE COOLING MECHANICS
2. POTENTIALLY DAMAGING OPERATING CONDITIONS
3. GAS/STEAM BINDING ON CORE COOLING
4. RECOGNIZING CORE DAMAGE
5. HYDROGEN HAZARDS DURING ACCIDENT CONDITIONS
6. MONITORING CRITICAL PARAMETERS DURING ACCIDENT CONDITIONS
    - INCORE INSTRUMENTATION
    - EXCORE INSTRUMENTATION
    - PROCESS COMPUTER
    - HIGH RADIATION SAMPLING
7. RADIATION HAZARDS AND RADIATION MONITOR RESPONSE
8. CRITERIA FOR OPERATION AND COOLING MODE SELECTION
9. INFREQUENT ABNORMAL AND EMERGENCY OPERATING PROCEDURES
10. THERMODYNAMICS AND HEAT TRANSFER
11. RECRITICALITY POTENTIAL
12. EMERGENCY PLAN

CONTROL ROOM DESIGN REVIEW FOR HUMAN FACTORS

THE DETROIT EDISON COMPANY HAS BEEN AWARE OF THE PROBLEMS THAT A MAN-MACHINE INTERFACE CAN CAUSE IN THE COMPLEX MODERN POWER PLANTS, CONTROLLED FROM A LOCATION REMOTE FROM THE EQUIPMENT. TO COPE WITH THESE PROBLEMS THE COMPANY DECIDED IN 1965 TO CONVENE A TASK FORCE WHENEVER A NEW CONTROL ROOM WAS BEING DEVELOPED.

THE FERMI 2 CONTROL ROOM TASK FORCE (CRTF) COMPRISED TWO OPERATING EMPLOYEES, REPRESENTATIVES FROM THE THREE ENGINEERING DISCIPLINES: (MECHANICAL, ELECTRICAL AND I&C), AND A PSYCHOLOGIST, TO PROVIDE WHAT IS NOW CALLED THE HUMAN ENGINEERING INPUT. SYSTEM ENGINEERS, FAMILIAR WITH THE FUNCTIONAL REQUIREMENTS OF EACH SYSTEM, WERE CALLED IN WHEN THEIR ASSIGNED SYSTEMS WERE DISCUSSED.

TO COMPENSATE FOR THE LIMITED EXPERIENCE IN NUCLEAR OPERATION, THE CRTF VISITED THE MORRIS SIMULATOR TO GAIN SOME "HANDS ON" EXPERIENCE WITH THE NUCLEAR POWER PLANT SYSTEMS. OTHER NUCLEAR PLANTS WERE ALSO VISITED TO SEE THEIR CONTROL ROOMS, AND TO OBTAIN COMMENTS FROM THEIR OPERATORS. SOME OF THE BASIC DECISIONS REACHED BY THE CRTF, AND LATER IMPLEMENTED IN THE FERMI 2 CONTROL ROOM WERE:

- ALL NORMAL AND ABNORMAL OPERATION OF THE PLANT SHOULD BE DONE FROM THE CONTROL ROOM.

- A PRIORITY SYSTEM FOR EASY ACCESS TO CONTROL COMPONENTS: THE EMERGENCY OPERATION FIRST, THEN STARTUP AND SHUTDOWN, AND THEN THE NORMAL OPERATION.

- PANELS ARRANGED IN A DOUBLE HORSESHOE, WRAPPED AROUND AN OPERATOR'S DESK.

- THE CROSS SECTION OF THE PANEL TO BE CONTOURED RATHER THAN FLAT, TO AFFORD A MAXIMUM WORK SPACE ACCESSIBLE TO THE REACH OF THE OPERATOR.

- FUNCTIONAL SYSTEMS CONTROL COMPONENTS ARRANGED TOGETHER WITH THE INFORMATION REQUIRED FOR OPERATING THEM.

- CONSISTENT USE OF SHAPE CODING: ALL VALVES OPERATED BY MASTER SPECIALTIES PUSBUTTONS, ALL OTHER EQUIPMENT BY CMC SWITCHES: FLOW INDICATORS DIFFERENT SHAPE THAN TEMPERATURE OR PRESSURE INDICATORS, ETC.

- THE DETROIT EDISON STANDARD USE OF COLOR FOR INDICATING LIGHTS AND BACKLIGHTED SWITCHES WAS CONSISTENTLY APPLIED: RED FOR ENERGY FLOW, GREEN FOR NO FLOW, WHITE FOR ABNORMAL WARNING, AMBER FOR CAUTION.

- MIMICS USED WHEREVER POSSIBLE TO HELP THE OPERATORS IN LOCATING THE COMPONENTS MORE EASILY.

- THE ANNUNCIATOR WINDOW LETTERING LARGE ENOUGH TO READ FROM THE CENTER OF THE HORSEHOE.

- USE OF COLOR CODED ANNUNCIATOR WINDOWS.

- MARKING INSTRUMENT SCALES WITH COLOR BANDS TO DENOTE NORMAL, ABNORMAL AND EMERGENCY SITUATIONS.

A FULL SIZE MOCKUP OF THE CONTROL ROOM WAS CONSTRUCTED, AND EACH PLANT FUNCTIONAL SYSTEM LAYED OUT ON THE MOCKUP. AFTER THE TASK FORCE HAD AGREED ON THE LAYOUT, THE OPERATORS PERFORMED MOCK OPERATION, STARTUP, SHUTDOWN, AND EMERGENCIES. THE MOCKUP WAS THEN ADJUSTED FOR DEFICIENCIES OBSERVED BY THE TASK FORCE.

THE FINAL PANEL LAYOUT WAS COMPLETELY REVIEWED BY THE PROJECT SYSTEM ENGINEERING GROUP, BEFORE SENDING IT TO THE MANUFACTURER.

AFTER THREE MILE ISLAND, DETROIT EDISON REVIEWED THE CONTROL ROOM COMPARING IT WITH EPRI REPORT NP-1118-54, AND ISSUED GED TECHNICAL REPORT NO. 154 IN MARCH 1980. IT INDICATED CLOSE CONFORMANCE TO THE PRINCIPLES DEFINED BY EPRI.

DETROIT EDISON HAS ALSO ACTIVELY PARTICIPATED IN THE BWR CONTROL ROOM COMMITTEE, DEVELOPING THE CONTROL ROOM REVIEW PROCEDURE AND CHECKLIST. IN JANUARY 1981 THE FERMI 2 CONTROL ROOM WAS REVIEWED BY THE OWNERS' GROUP TEAM COMPRISING REPRESENTATIVES FROM TWO OTHER UTILITIES, GENERAL ELECTRIC PEOPLE AND A HUMAN FACTORS EXPERT FROM MIT. TO ASSURE THE INDEPENDENCE OF THE OBSERVATIONS, DETROIT EDISON LIMITED ITS PARTICIPATION TO SUPPORTING FUNCTIONS ONLY. NO SIGNIFICANT DEVIATIONS FROM THE LATEST HUMAN FACTORS PRINCIPLES WERE FOUND.

THE NRC CONDUCTED A VERY THOROUGH AND DETAILED REVIEW WITH SIMILAR RESULTS. THE NUMEROUS SMALL HUMAN ENGINEERING DEVIATIONS THAT THE STAFF REQUESTED TO CORRECT BEFORE FUEL LOAD WILL BE COMPLETED.

THERE WERE FIVE OPEN ITEMS THAT COULD NOT BE EVALUATED BY THE NRC STAFF.

1. PERMANENT HVAC SYSTEM PERFORMANCE

2. PROCEDURES FOR PERMANENT PANEL MODIFICATIONS

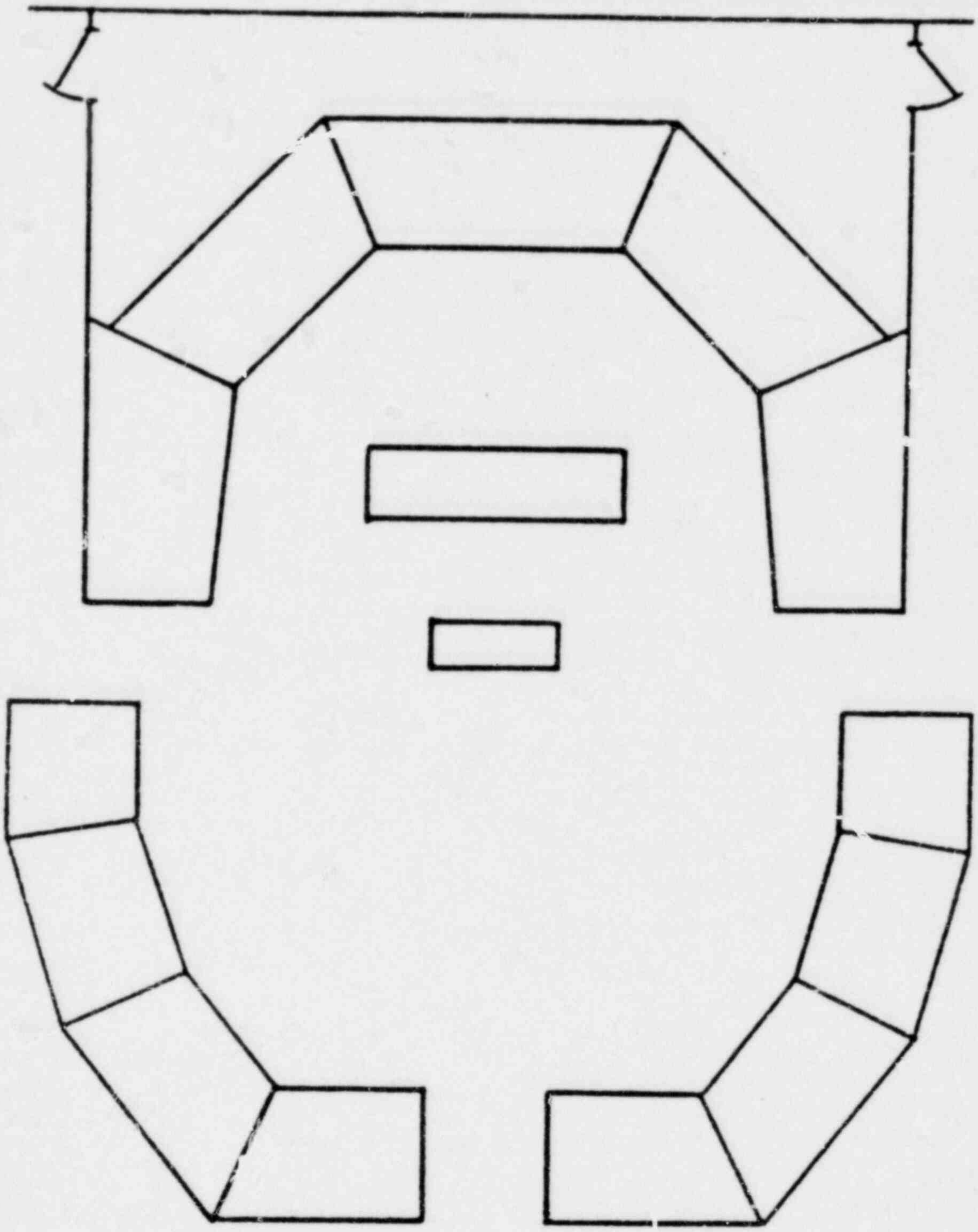
3. CONTROL ROOM SIGNAL EVACUATION

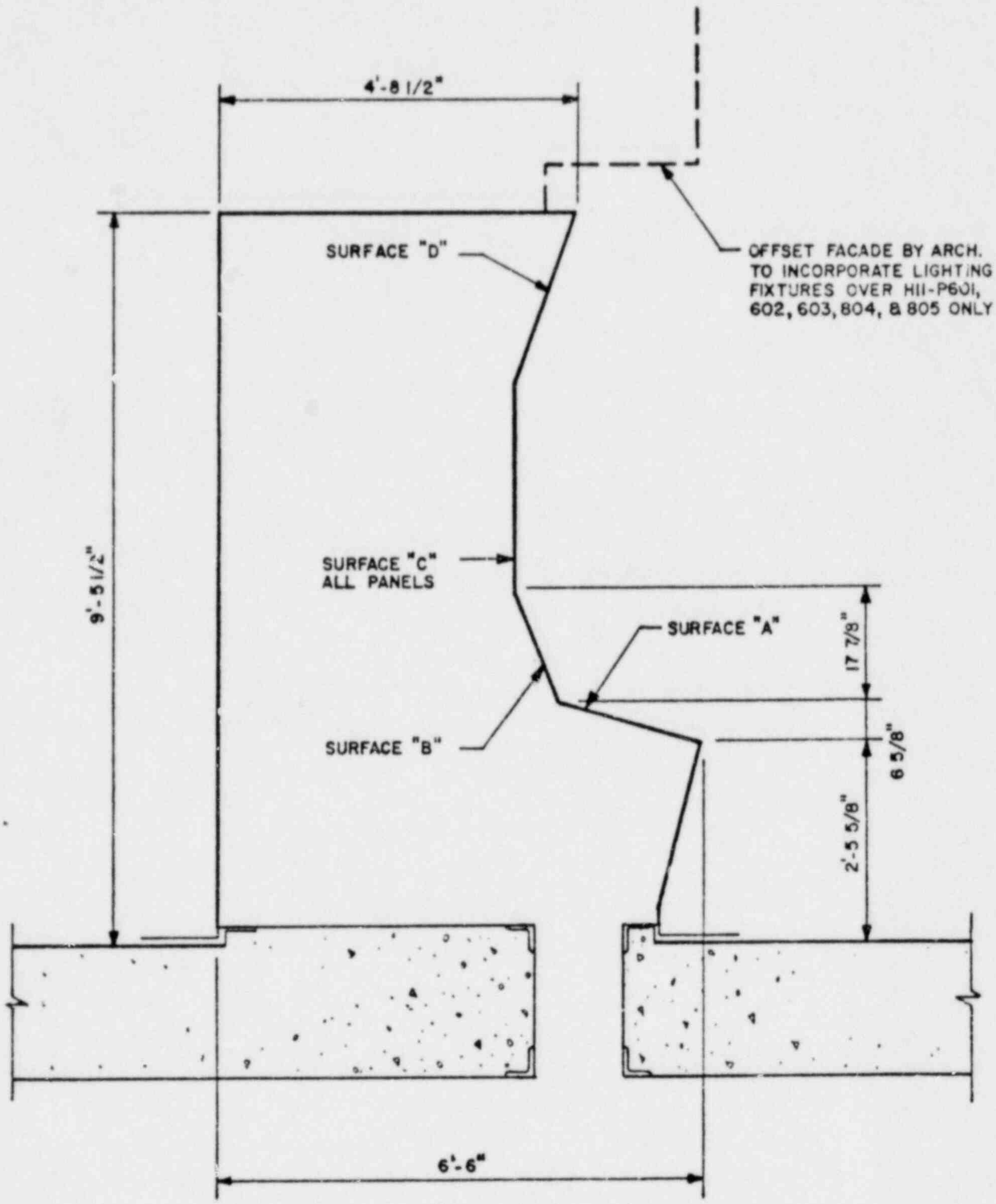
4. PROCESS COMPUTER SOFTWARE

5. COLOR CODING STANDARD

ITEMS 1, 2 AND 5 HAVE BEEN ADDRESSED AND REPORTS ARE BEING SUBMITTED TO THE NRC STAFF BEFORE THE DUE DATE OF AUGUST 1, 1981. ITEM 4 WILL BE SUBMITTED TO THE STAFF BEFORE FUEL LOAD. (IT IS A GE STANDARD PACKAGE, BUT THE PLANT SPECIFICS ARE NOT YET INCORPORATED.)

ITEM 3 IS BEING PROCURED, IT WILL BE TESTED OUT AND RESULTS WITNESSED BY THE NRC RESIDENT I&E INSPECTOR AND SUBMITTED TO THE STAFF BEFORE OCTOBER 15, 1981.





EF2 CONTROL PANEL PROFILE

CONTROL ROOM HABITABILITY

THE FERMI 2 MAIN CONTROL (MCR) HABITABILITY SYSTEM HAS BEEN CAREFULLY DESIGNED TO ENSURE HABITABILITY FOR PERSONNEL AND INTEGRITY OF THE SAFETY-RELATED CONTROL EQUIPMENT AND COMPONENTS INSIDE THE CONTROL ROOM UNDER ALL PLANT OPERATING CONDITIONS, INCLUDING A DESIGN BASIS ACCIDENT. THE HVAC SYSTEM SERVING THIS ROOM IS DESCRIBED IN DETAIL IN FSAR SECTION 6.4. A FEW OF THE MAJOR ITEMS INSTALLED ARE:

a. DUAL INDEPENDENT OUTSIDE AIR INLETS, MISSILE-PROTECTED AND SEPARATED ON OPPOSITE SIDES OF THE REACTOR BUILDING

b. EACH INTAKE IS PROVIDED WITH REDUNDANT RADIATION MONITORS. A HIGH-RADIATION SIGNAL AUTOMATICALLY ROUTES THE OUTSIDE MAKEUP AIR THROUGH THE EMERGENCY FILTER UNIT (WHICH IS CAPABLE OF REMOVING 95% OF ALL RADIOACTIVE AND NONRADIOACTIVE FORMS OF IODINE); AUTOMATICALLY ISOLATES THE OUTSIDE AIR INTAKES; AND PLACES THE HVAC SYSTEM IN 100% RECIRCULATION MODE WHEREBY ALL RETURN AIR IS ROUTED THROUGH A NORMALLY BYPASSED RECIRCULATION CHARCOAL-ABSORBER TRAIN (95% IODINE ABSORPTION)

c. COMPLETE REDUNDANCY OF ACTIVE COMPONENTS

d. AUTOMATIC CONTROL OF INLETS SO THAT CONTROL-ROOM AIR INTAKE IS FROM WHICHEVER INLET HAS THE LOWER RADIATION LEVEL.

BOTH THE CONTROL-ROOM RADIATION SHIELDING AND SAFETY-RELATED HVAC SYSTEM CONTROL THE TOTAL INTEGRATED DOSES FOLLOWING A DESIGN-BASIS EVENT (LOCA). THE ACCEPTED INTERPRETATION OF CRITERION 19 OF 10 CFR 50, APPENDIX A, IS THAT AN OPERATOR IN THE CONTROL ROOM DURING A POST-LOCA RECOVERY PERIOD WILL NOT RECEIVE DOSES IN EXCESS OF 5 REM WHOLE BODY OR 20 REM THYROID, BASED UPON THE FISSION-PRODUCT RELEASE ASSUMPTIONS OF REGULATORY GUIDE 1.3 (FOR A BWR). THE VALUES FOR THE FERMI 2 PLANT ARE GIVEN IN THE ACCOMPANYING TABLE. THE VALUES CALCULATED BY THE NRC STAFF ARE REASONABLY CLOSE TO DETROIT EDISON'S VALUES; BOTH ARE LESS THAN THE APPLICABLE LIMITS. THE OVERWHELMING MAJORITY OF THE OPERATOR DOSE COMES FROM CONTROL-ROOM INHALATION (ONLY ABOUT 2 REM RESULTS FROM DIRECT RADIATION "SHINE" THROUGH THE CONCRETE WALLS, FLOORS, AND CEILINGS). IN FACT, THE MCR WAS DESIGNED FOR DIRECT DOSE TO PERSONNEL BASED UPON THE VERY CONSERVATIVE ASSUMPTION OF 1% CORE RELEASE OF SOLID FISSION PRODUCTS, IN ADDITION TO THE NOBLE GASES AND IODINES.

DETROIT EDISON FEELS THAT THERE IS BASIC CONSERVATISM IN THE ALREADY-ASSUMED FISSION-PRODUCT RELEASE FRACTIONS, IN ADDITION TO THE VERY CONSERVATIVE METHODOLOGY FOR DOSE CALCULATIONS. THEREFORE, EVEN IF A MORE SEVERE ACCIDENT WERE ASSUMED BEYOND THE FERMI 2 DBA, THE RESULTANT CONTROL-ROOM DOSES SHOULD NOT BE SIGNIFICANTLY HIGHER THAN THE PRESENT VALUES (SEE THE ACCOMPANYING TABLE 2):

a. THE WHOLE-BODY, WHICH DEPENDS ALMOST ENTIRELY ON THE NOBLE GAS, ALREADY IS BASED UPON 100% RELEASE. THEREFORE, THIS DOSE SHOULD NOT INCREASE.

b. THE THYROID DOSE DEPENDS ENTIRELY UPON THE RELEASED IODINE, FOR WHICH 50% WAS ALREADY ASSUMED. THEREFORE, THIS DOSE SHOULD PROBABLY INCREASE NO MORE THAN DOUBLE.

c. THE DIRECT DOSE DEPENDS UPON ALL FISSION-PRODUCTS. BUT THE PRESENT DESIGN WAS ALREADY BASED UPON 100% NOBLE GAS, 50% IODINES, AND 1% SOLIDS RELEASED, WHICH WOULD THEREBY GIVE AN INTEGRATED MCR DOSE OF 0.6 REM. DOUBLING THE IODINE RELEASED WOULD NOT AFFECT THIS VALUE. IT IS DIFFICULT TO ENVISION EVEN 1% OF THE SOLID FISSION-PRODUCT INVENTORY REMAINING SUSPENDED IN THE ATMOSPHERE.

POST-LOCA CONTROL ROOM HABITABILITY DOSES

	DETROIT EDISON ANALYSIS			
	REALISTIC BASIS	DESIGN BASIS	NRC ANALYSIS	NRC LIMITS
WHOLE BODY	2×10^{-5} REM	2 REM	1.5 REM	5 REM
THYROID	3×10^{-5} REM	4 REM	15.7 REM	30 REM
BASIS	NO FUEL MELT	REGULATORY GUIDE 1.3 100% NOBLE GAS 50% IODINE 0.005/DAY LEAKAGE TO ATMOSPHERE		

DIRECT DOSE THROUGH MCR CONCRETE WALLS 0.001 REM

CONTROL-ROOM HABITABILITY FOR ACCIDENTS BEYOND DBA

A. WHOLE BODY DOSE: WILL NOT CHANGE.
(100% NG RELEASE ALREADY ASSUMED)

B. THYROID DOSE: MIGHT DOUBLE.
(50% RELEASE ALREADY ASSUMED)

C. DIRECT DOSE THROUGH CONCRETE

1. PRESENT MCR DESIGN ALREADY BASED UPON RELEASE MORE SEVERE THAN DBA: 100% NG
50% I
1% SOLIDS

DIRECT INTEGRATED DOSE IN MCR (FOR ABOVE RELEASE): 0.6 REM

2. CONCRETE SHIELDING AVAILABLE AROUND CORE:

A.	NORMAL OPERATION: (NEAREST ACCESS)	11' NEEDED
B.	NORMAL OPERATION (TO CONTROL ROOM)	16' - 19'
C.	FUEL MELTED INTO DRYWELL	15' - 23'

INADEQUATE CORE COOLING INSTRUMENTATION

AS A DIRECT RESULT OF THE TMI EVENTS, THE NRC STAFF DEFINED A BROAD OBJECTIVE IN ITEM II.F.2 OF NUREG-0737. THIS INADEQUATE CORE COOLING (ICC) INSTRUMENTATION REQUIREMENT WAS DIRECTED PRIMARILY TOWARD THE PRESSURIZED WATER REACTOR DESIGNS AS A RESULTS OF THE PROBLEMS FOUND IN THE MEASUREMENT OF CORE COOLING ADEQUACY UTILIZING PRESSURIZER LEVEL SYSTEM AND/OR THE CORE EXIT THERMOCOUPLES.

THE NEW REQUIREMENT SPECIFIES THAT AN UNAMBIGUOUS INDICATION OF ICC BE DEVELOPED. IN THE CASE OF A BWR DESIGN, THE WATER LEVEL INSTRUMENTATION CAN NEVER BE AMBIGUOUS. THE BWR WATER LEVEL INSTRUMENTATION PROVIDES AN ADVANCED INDICATION OF THE APPROACH TO ICC DUE TO THE USE OF INSTRUMENTS OF MULTIPLE OVERLAPPING RANGES. A VERY COMPLETE ANALYSIS OF THE LEVEL MEASUREMENT SYSTEM OF A BWR SUCH AS FERMI 2 CAN BE FOUND IN NEDO-2470A. EDISON WAS REQUESTED BY THE STAFF TO SUPPLEMENT THIS INFORMATION BY LETTER AND ESTABLISH FERMI 2 SPECIFIC RANGES, INSTRUMENT TYPES AND OTHER PHYSICAL DESIGN INFORMATION. THE STAFF REVIEWED AND REPORTED ON THE APPLICABILITY OF THIS NEDO TO THE FERMI 2 LEVEL MEASUREMENT SYSTEM IN THE SER SECTION H.II.K.1.23.

IN ADDITION TO THE INSTRUMENTATION THE SYMPTOMATIC EMERGENCY PROCEDURES WHICH DEPEND PRIMARILY ON THE LEVEL SYSTEM ARE ESSENTIALLY COMPLETE AND HAVE BEEN APPROVED BY THE NRC.

INADEQUATE CORE COOLING INSTRUMENTATION

IMMEDIATELY FOLLOWING THE TMI EVENT, EDISON INDEPENDENTLY REVIEWED THE FERMI 2 LEVEL MEASUREMENT SYSTEM BECAUSE OF THE AMBIGUITY ASSOCIATED WITH THE PRESSURIZER LEVEL WHICH WAS OBSERVED AT TMI DURING THE ACCIDENT.

BASED ON THE RESULT OF OUR INTERNAL STUDY AND ANALYSIS, EDISON ENDORSED THE BWR OWNERS' GROUP POSITION THAT FERMI 2 DOES NOT NEED ANY ADDITIONAL INSTRUMENTATION TO PROVIDE UNAMBIGUOUS INDICATION OF ADEQUATE CORE COOLING.

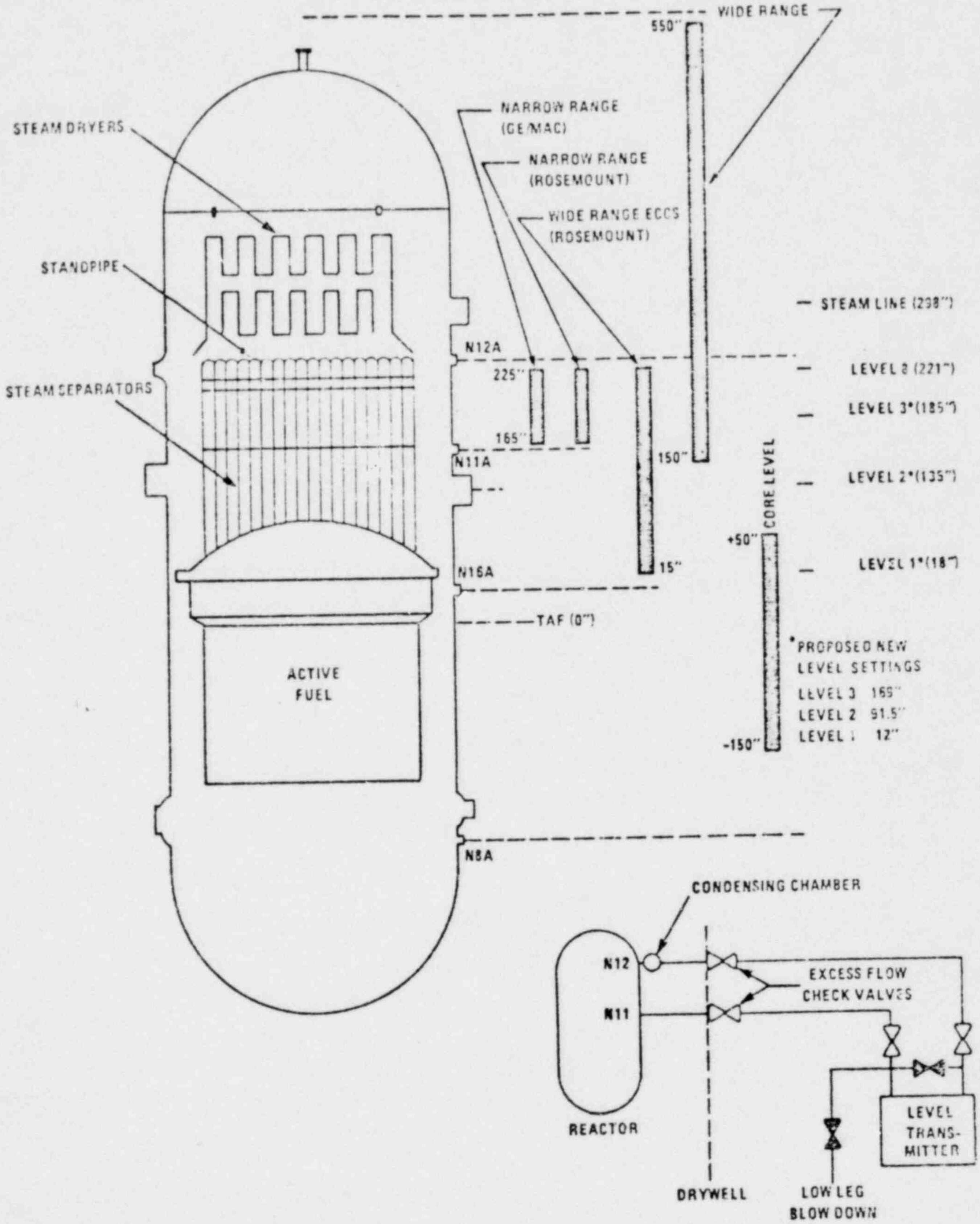


FIGURE 2-7
FERMI 2 VESSEL LEVEL RANGES (REFERENCED TO TOP OF ACTIVE FUEL) AND DESIGN OF TYPICAL INDICATOR LINE

REGULATORY GUIDE 1.97 REQUIREMENTS

FOLLOWING THE TMI EVENT, EDISON INTERNALLY INITIATED A REVIEW OF ACCIDENT PREVENTION AND DETECTION INSTRUMENTATION ALONG WITH THE POST ACCIDENT INSTRUMENTATION.

THE EDISON REVIEW INDICATED THE NEED FOR SEVERAL CHANGES WHICH WERE MADE TO THE PLANT SYSTEMS. A SHORT TIME LATER, REGULATORY GUIDE 1.97, REVISION 2, WAS ISSUED BY THE NRC AND CONCURRENTLY THE MORE SIGNIFICANT INSTRUMENT REQUIREMENTS WERE INCLUDED IN NUREG-0737.

EDISON BELIEVES THAT THE FERMI 2 PLANT, WITH FEW EXCEPTIONS, (NOTABLY THE INCORE THERMOCOUPLES), IS IN CLOSE CONFORMANCE WITH THE INTENT OF REGULATORY GUIDE 1.97, REVISION 2.

IN ORDER TO MORE COMPLETELY DESCRIBE THE NUMBER AND EXTENT OF CHANGES MADE TO THE FERMI 2 INSTRUMENTATION TO MEET THE REQUIREMENTS OF NUREG-0737 AND REGULATORY GUIDE 1.97, THE INSTRUMENTS AFFECTED ARE SHOWN ON THE ATTACHED SKETCHES AND LISTED BELOW:

- NOBLE GAS EFFLUENT MONITORS
- IODINE AND PARTICULATE MONITORS

- HIGH RANGE CONTAINMENT RADIATION MONITORS

- SUPPRESSION POOL LEVEL MONITORS

- DRYWELL PRESSURE MONITORS

- CONTAINMENT HYDROGEN AND OXYGENMONITORS

- SAFETY/RELIEF VALVE POSITION MONITORS

- POST ACCIDENT SAMPLING SYSTEM

- DRYWELL SUMP LEVEL MONITORS

EDISON PLANS TO WORK WITH THE BWR OWNERS' GROUP IN ADDRESSING THE ADDITIONAL REQUIREMENTS OF REGULATORY GUIDE 1.97 BEYOND NUREG-0737.

ACRS SUBCOMMITTEE MEETING

EVALUATION OF
BWR CORE THERMOCOUPLES

JULY 24, 1981
WASHINGTON, D. C.

II.D.2-3

OVERVIEW

- BWR CORE THERMOCOUPLE ISSUE
- POSTULATED CONDITIONS OF USEFULLNESS
- BWR DESIGN EVALUATION
- BWR OPERATION EVALUATION
- BWR RISK EVALUATION
- BWR IMPACT EVALUATION
- SUMMARY

BWR CORE THERMOCOUPLES

NRC PURPOSES

- MONITOR CORE COOLING
- DIVERSE INDICATION OF WATER LEVEL

GE CONCERN

- CORE EXIT TEMPERATURE MEASURED BY THERMOCOUPLES WILL NOT SATISFY EITHER NRC PURPOSE EXCEPT ONE NARROW CONDITION
- THERMOCOUPLES NOT COST EFFECTIVE AND CAUSE HIGH MAN-REM DOSAGES

SOLUTION

- PRESENT BWR WATER LEVEL INSTRUMENTATION IS ADEQUATE TO RELIABLY MONITOR WATER LEVEL AND CORE COOLING
- OTHER INSTRUMENTATION PROVIDE DIVERSE INDICATION OF WATER LEVEL FOR THE SAME NARROW CASE AS THERMOCOUPLES

RAH:HMC:RM/1691
6/30/81

THERMOCOUPLES USEFULLNESS

SYSTEMS UNAVAILABLE		THERMOCOUPLES WILL PROVIDE USEFUL INFORMATION	THERMOCOUPLES WILL WORK
1.	RCIC/CRD	NO	NO
2.	HPCI	NO	NO
3.	RCIC/CRD/HPCI	NO	NO
4.	LPCI/CS	NO	NO
5.	RCIC/CRD/LPCI/CS	NO	NO
6.	HPCI/LPCI/CS	NO	NO
7.	ADS	NO	NO
8.	RCIC/CRD/ADS	NO	NO
9.	HPCI/ADS	NO	NO
10.	ALL WATER LEVEL MONITORING	NO	NO
11.	ALL WATER LEVEL MONITORING PLUS ANY OF 1-9	NO	NO
12.	HPCI/RCIC/CRD PLUS LPCI/CS/OR ADS	NO	YES
13.	HPCI/RCIC/CRD PLUS LPCI/CS/OR ADS PLUS ALL WATER LEVEL MONITORING	NO	YES

BWR DESIGN EVALUATION

- BWR MULTIPLE SOURCES OF WATER - 13 PUMPS
  - CORE SPRAY
  - CORE FLOODING
  - HIGH PRESSURE SYSTEMS
  - ADS
  - LOW PRESSURE SYSTEMS

- CORE SPRAY PREVENTS THERMOCOUPLES FROM WORKING

- BWR MEASURES LEVEL DIRECTLY IN REACTOR VESSEL

- BWR REDUNCANCY OF LEVEL INDICATIONS
  - 18 LEVEL TRANSMITTERS
  - 7 LEVEL METERS/RECORDERS
  - 3 LEVEL RANGES TO NOTIFY OPERATOR WHEN WATER LEVEL BELOW NORMAL
  - 2 LEVEL METERS/RECORDERS IN FUEL ZONE

- RADIATION AND HYDROGEN MONITORS, PLUS REACTOR COOLANT SAMPLING PROVIDE DIVERSE MONITORING

BWR OPERATION EVALUATION

- OPERATOR WILL RESTORE WATER LEVEL LONG BEFORE FUEL UNCOVERY
- SYMPTOMATIC EMERGENCY PROCEDURE GUIDELINES PROVIDE SAFE OPERATOR ACTIONS
  - WITHOUT ANY ECCS
  - WITHOUT ANY WATER LEVEL MONITORING
  - WITHOUT ECCS AND WATER LEVEL MONITORING
- THERMOCOUPLE INDICATIONS WOULD NOT PROVIDE OPERATOR ACTION INFORMATION
- FALSE THERMOCOUPLE INDICATION MAY MISLEAD THE OPERATOR

BWR RISK EVALUATION

- PROBABILISTIC RISK EVALUATION IN PROGRESS
  - ASSUMES THERMOCOUPLES ARE INSTALLED
  - ASSUMES THERMOCOUPLE TIME LAG OF MINUTES
  - ASSESSES IMPROVEMENTS IN OPERATOR ACTIONS
  - ASSESSES CHANGE IN TOTAL CORE MELT FREQUENCY
  - ASSESSES APPROXIMATE EFFECT ON EARLY FATALITY RISK
  - TARGETED COMPLETION - END OF AUGUST
- RISK REDUCTION EXPECTED TO BE SMALL
- RISK INCREASE MAY BE SHOWN BY FALSE THERMOCOUPLE INDICATION

BWR IMPACT

- RADIATION

  - RADIATION DOSE TO MAINTAIN AND INSTALL: APPROXIMATELY 100 MANREM PER PLANT

- COST

  - COST FOR INSTALLATION: APPROXIMATELY $600,000 PER PLANT

- SAFETY GOAL COMPARISON

  - ACRS SAFETY GOAL PROPOSED COST BENEFIT
    $\$ 1 \times 10^6$ PER LIFE

  - BEIR III LINEAR DOSE FATALITY COEFFICIENT
    $\$ 1 \times 10^6$ PER LIFE = $100/MAN-REM

  - BWR CORE THERMOCOUPLES
    $ 6.000/MAN-REM (OCCUPATIONAL)

- INCREASED POTENTIAL FOR FORCED OUTAGE

SUMMARY

- THERMOCOUPLES WILL WORK FOR ONLY ONE NARROW CONDITION
  - WATER LEVEL IN THE CORE REGION

- BWR HAS MULTIPLE INJECTION AND FLOODING SOURCES

- BWR HAS REDUNDANT AND SUFFICIENT WATER LEVEL MONITORING

- OTHER INSTRUMENTS PROVIDE DIVERSITY

- THERMOCOUPLES NOT BENEFICIAL FOR OPERATIONS
  - OPERATOR WILL RESTORE WATER LEVEL WITHOUT THERMOCOUPLES
  - FALSE INDICATIONS MAY MISLEAD THE OPERATOR
  - OPERATOR CAN ACT SAFELY EVEN WITHOUT WATER LEVEL INSTRUMENTATION

- DOSE AND COST ARE SIGNIFICANT

- NO REDUCTION IN RISK: MAYBE COUNTER-PRODUCTIVE

- POTENTIAL INCREASE IN FORCED OUTAGES

PLANT SEISMIC DESIGN

ON MARCH 12, 1981 DETROIT EDISON WAS REQUESTED BY THE NRC STAFF TO REEVALUATE THE SEISMIC DESIGN OF THE FERMI-2 PLANT BASED ON:

a. USE OF A REGULATORY GUIDE 1.60 SPECTRUM SHAPE AND ANCHORED AT 19% G OR

b. DEVELOPMENT OF A SITE SPECIFIC GROUND RESPONSE SPECTRA REPRESENTATIVE OF EARTHQUAKE HISTORIES OF MAGNITUDE 5.3.10.5 APPLICABLE TO ROCK SITES.

IN A MEETING WITH NRC STAFF ON MARCH 27, 1981, DETROIT EDISON COMMITTED TO CONDUCT A REEVALUATION OF THE PLANT'S CAPABILITY TO SAFELY SHUTDOWN, UTILIZING A QUASI SITE SPECIFIC SPECTRUM, DEVELOPED BY WESTON GEOPHYSICAL BASED ON A MAGNITUDE 5.3. $\pm$ 0.5 EARTHQUAKE. THE SHAPE OF THE SPECTRUM EVELOPES THE LAWRENCE LIVERMORE SPECTRUM AND FOLLOWS IN GENERAL TREND THE REGULATORY GUIDE 1.60 SHAPE. IN ADDITION, FAR FIELD STRONG MOTION EFFECTS HAVE BEEN INCLUDED. EVEN THOUGH THE ANCHOR POINT (ZPA) DID NOT CHANGE FROM THE SSE (0.15 g), THE GROUND ACCELERATION INCREASED BY AS MUCH AS 60% OVER THE PREDOMINANT BUILDING FREQUENCY RANGE. A COMPARISON OF THE SSE AND MODIFIED GROUND RESPONSE SPECTRA ARE SHOWN IN FIGURE 2.1-2.

ADEQUATE EARTHQUAKE HISTORIES ARE NOT AVAILABLE FOR DEVELOPMENT OF A SITE SPECIFIC VERTICAL GROUND RESPONSE SPECTRUM. THE SSE VERTICAL SPECTRUM IS BASED ON FOUR INDIVIDUAL TIME HISTORIES AND WAS COMPARED TO THE REGULATORY GUIDE 1.60 VERTICAL SPECTRUM SHAPE. THE COMPARISON (SHOWN ON FIGURE 2.1-3) DETERMINED THAT THE LATTER EXCEEDS THE SSE BY A FACTOR OF 1.6 IN THE DOMINANT STRUCTURAL FREQUENCIES.

DUE TO INHERENT VALLEYS IN THE FOUR TIME HISTORIES USED FOR THE VERTICAL SSE, WHICH TEND TO DRIVE DOWN THE AVERAGE SPECTRA, A FACTOR OF 2.0 WAS DETERMINED TO BE THE APPROPRIATE BASIS TO BE USED FOR THE REEVALUATION. THE MULTIPLIER WAS APPLIED DIRECTLY TO THE SSE FLOW RESPONSE SPECTRA.

HISTORICAL RECORDS OF SEISMIC ACTIVITY IN THE CENTRAL REGION OF THE U.S. WERE USED TO ESTIMATE THE FREQUENCY OF OCCURRENCE OF A 0.08 g OBE GROUND ACCELERATION. ON THE BASIS OF THIS HISTORICAL ANALYSIS, IT WAS CONCLUDED THAT THE OBE RECURRENCE FREQUENCY AT THE FERMI 2 SITE IS, AS A MINIMUM, IN THE ORDER OF 100 TO 300 YEARS. THEREFORE, NO CHANGE IN THE OBE DEFINITION HAS OCCURRED.

BASED ON THE ABOVE CRITERIA, NEW SYNTHETIC N-S AND E-W TIME HISTORIES, MATCHING THE 7% DAMPED SITE SPECTRUM WERE DEVELOPED AND APPLIED SIMULTANEOUSLY TO OBTAIN ACCELERATION RESPONSE

TIME HISTORIES FOR THE STRUCTURAL MODES. THESE TIME HISTORIES IN TURN ARE USED AS INPUT MOTIONS TO THE GENERATION OF FLOOR RESPONSE SPECTRA. THE REACTOR VESSEL AND ITS INTERNALS ARE PART OF THE STRUCTURAL MODEL, GENERATING SHEAR LOADS AND MOMENTS USED IN THE COMPONENT STRESS EVALUATION.

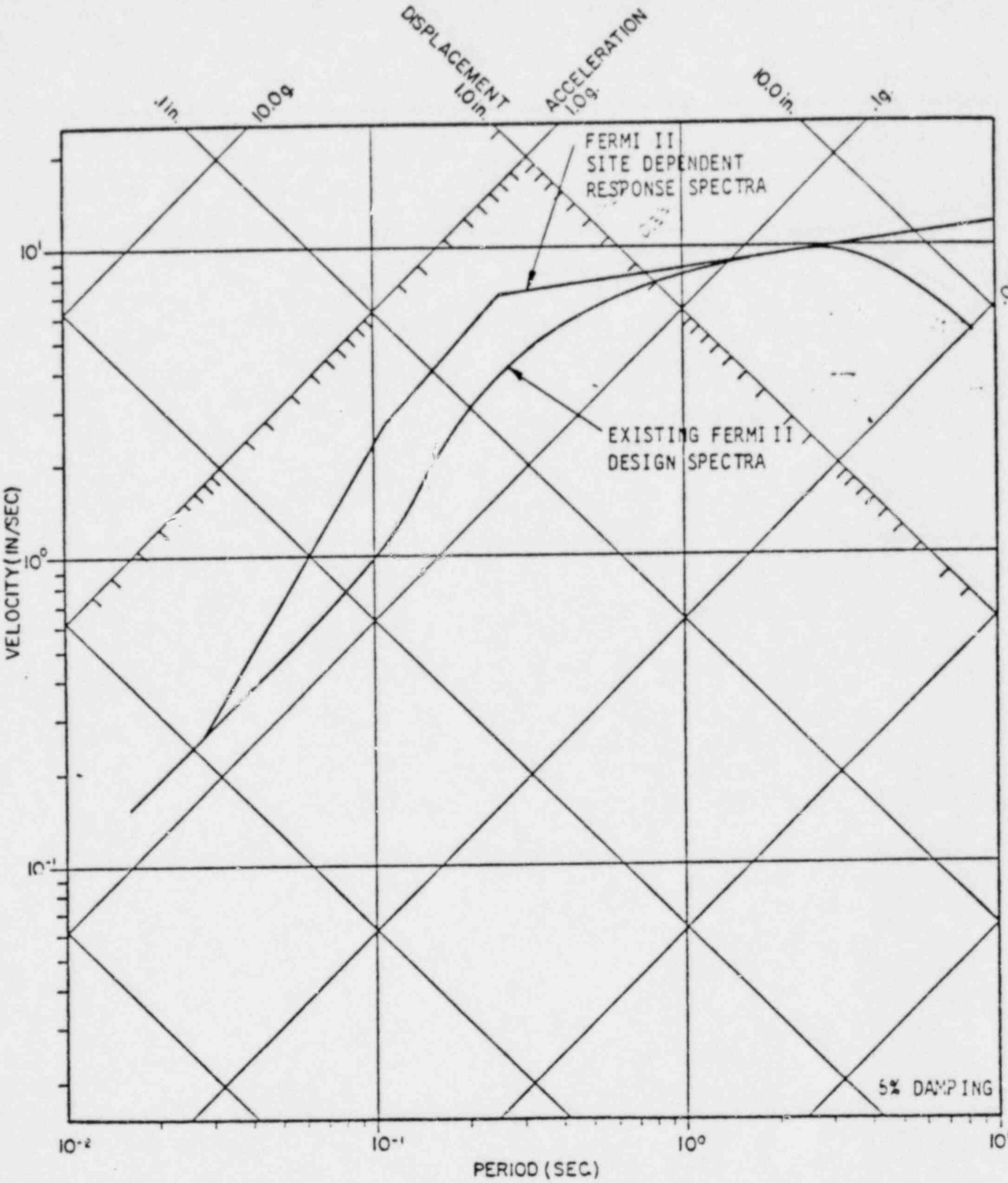
REEVALUATION PROGRAM

THE PRINCIPAL STRUCTURES, SYSTEMS AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN AND COOLDOWN WERE REEVALUATED BASED ON THE NEW EARTHQUAKE CRITERIA AND EMPLOYING DAMPING RATIOS AND LOAD COMBINATIONS CONSISTENT WITH CURRENT PRACTICE (I.E. REGULATORY GUIDE 1.60 AND SRP).

THE EVENT SCENARIO (TRANSIENT) WAS DEVELOPED TO IDENTIFY SHUTDOWN AND COOLDOWN EQUIPMENT REQUIRED TO FUNCTION, BUT NOT INCLUDING A SIMULTANEOUS LOCA. LOSS OF OFFSITE POWER (LOPA) WAS ASSURED TO OCCUR; TABLE 5.2-1 LISTS THE PRINCIPAL AND AUXILIARY SYSTEMS REQUIRED FOR THIS EVENT. PROCESS AND FUNCTIONAL CONTROL DIAGRAMS WERE THEN USED TO IDENTIFY PARTICULAR EQUIPMENT AND COMPONENTS. OVER 200 INDIVIDUAL PIECES OF EQUIPMENT WERE SELECTED FOR REASSESSMENT, REPRESENTING MORE THAN 500 ITEMS. DETAILED RESULTS OF THE EVALUATION ARE PRESENTED IN A PRELIMINARY REPORT EF2-53,332 AND A FINAL REPORT REVISION 1 OF EF2-53,332 WHICH WAS DOCKETED ON JULY 15, 1981.

EVEN THOUGH THE REEVALUATION IDENTIFIED ABOUT 24 ITEMS THAT REQUIRE FURTHER ANALYTICAL REFINEMENT OR OTHER CORRECTIVE ACTIONS, THE REASSESSMENT CONCLUDED THAT THE PLANT IS CAPABLE OF SAFELY SHUTTING DOWN AND COOLING DOWN AND DEMONSTRATING THE MARGINS OF SAFETY AVAILABLE TO WITHSTAND A HIGHER MAGNITUDE EARTHQUAKE.

IN PARTICULAR THE PRIMARY SHUTDOWN AND COOLDOWN EQUIPMENT, INCLUDING THE PIPING SYSTEMS AND VALVES INHIBIT AMPLE MARGINS, EVEN WHEN COMPARED TO CODE ALLOWABLES.



COMPARISON OF FERMI II SITE DEPENDENT RESPONSE SPECTRA AND EXISTING FERMI II DESIGN SPECTRA

FIGURE 2.1-2

SARGENT & LUNDY
ENGINEERS

30 APR 81
A943YC

CALC NO. CALC NO 8 11 0-3
PROJECT FERMI REV 0
PROJECT NO. 6139-38
DAMPING 0.050
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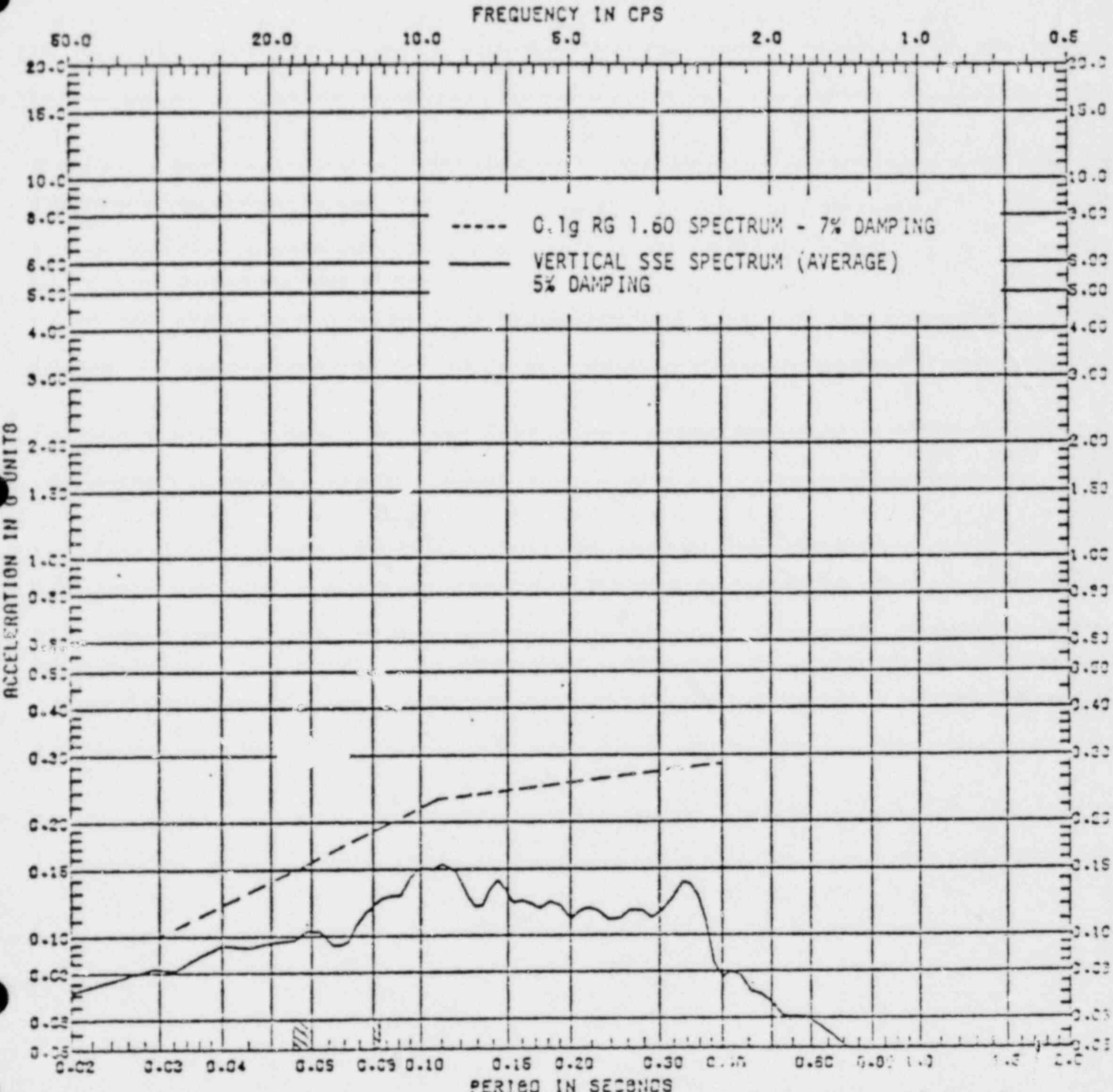


FIGURE 2.1-3

Comparison Of Regulatory Guide 1.60 and Average Time History Spectra

RESPONSE SPECTRUM SPECTRA NO
II.E-6
MODE AVERAG ELEVATION
1 DIRECTION ANGLE LOCATION ELCEN40.34.TAFT.OLYMPIA

TABLE 5.2-1

PRINCIPAL AND AUXILIARY SYSTEMS REQUIRED FOR SAFE SHUTDOWN AND COOLDOWN

PRINCIPAL SYSTEMS

1. RCIC
2. NUCLEAR BOILER
3. RHR-DIVISION II
4. CRD

AUXILIARY (SUPPORT) SYSTEMS

1. RHR SW - DIVISION I
2. DIESEL GENS - DIVISION I
3. DIESEL FUEL OIL & LUBE OIL - DIVISION I
4. EECW DIVISION I
5. EESW DIVISION I
6. EDGSW DIVISION I
7. CONTROL AIR - DIVISION I
8. CONTROL CENTER HVAC - AIR SIDE*
9. CONTROL CENTER HVAC - WATER SIDE*
10. DRYWELL COOLING - 4 TWO-SPEED FANS ONLY
11. DIESEL GENERATOR VENTILATION - DIVISION I
12. TORUS AND ATTACHED PIPING

*ESSENTIAL HVAC

SUPPLEMENTARY SEISMIC EVALUATION

- NRC Staff Request of March 12, 1981
  - Use R.G. 1.60 Spectra Shape Anchored at 19%G or
  - Site Specific for Rock site at Magnitude 5.3 ± .5
- Met with NRC on March 27, 1981 and Committed to:
  - Use Site Specific Spectrum
  - Modified for Far-field Strong Motions
  - Envelopes LLL-spectrum (84th % for M = 5.3)
  - Has Basic R.G. 1.60 Shape
  - Anchored at 15%G
  - Use Vertical Acceleration to be 2 x DBE (Floor Response)
  - No Change in OBE (100 - 300 Year Return Period)

E.E. Gregor
7/20/81

REEVALUATION

- Conducted Reevaluation:
  - Determine Shutdown scenario with LOCA
  - Identify Systems, Structures and Components
  - Not in Combination with LOCA
  - All Other Ground Rules per SRP
  - Completed Reassessment May 29, 1981, Report EF2-53,332
  - Final Report Docketed July 15, 1981 (Revision 1)

- Conclusions:
  - Plant is Capable of Safely Shutting Down
  - One Cable Tray Hanger Slightly Over Yield
  - 25 Items Require Further Evaluation, Requalification, Retesting or Replacement
  - Independent Assessment Using Ductility
  - Ductility Ratio of 2.0 will Reduce SSE Below DBE
  - NRR Audits on Structural and Mechanical Components

F.E.Gregor
7/20/81

DECAY HEAT REMOVAL CAPABILITY - NORMAL AND DEGRADED MODES

PRINCIPAL MEANS FOR REMOVING DECAY HEAT AT FERMI 2 IS BY THE SHUTDOWN COOLING AND CONTAINMENT COOLING MODES OF THE RESIDUAL HEAT REMOVAL (RHR SYSTEM).

THE PRIMARY OBJECTIVE OF THE DECAY HEAT REMOVAL FUNCTION IS TO TRANSFER THE DECAY HEAT FROM CORE TO THE ALTERNATE HEAT SINK WHICH IS DISSIPATED TO THE ATMOSPHERE.

THE NORMAL MODE OF REMOVING DECAY HEAT WHILE AT HIGH PRESSURE ($>$100 PSIG) IS VIA THE MAIN STEAM LINES TO THE TURBINE CONDENSER. WHEN THE SYSTEM IS AT LOW PRESSURE ($<$ 100 PSIG) THE DECAY HEAT IS REMOVED BY THE SHUTDOWN COOLING MODE OF THE RHR SYSTEM.

AS DEPICTED IN THIS FIGURE, IN THE SHUTDOWN COOLING MODE, REACTOR COOLANT IS PUMPED FROM RECIRCULATION LOOP, THROUGH THE RHR HEAT EXCHANGERS AND RETURNED TO THE REACTOR VESSEL VIA THE RECIRCULATION SYSTEM. HEAT EXCHANGERS ARE COOLED BY THE RHR SERVICE WATER SYSTEM. DECAY HEAT FROM THE RHR SERVICE WATER IS DISSIPATED TO THE ATMOSPHERE THROUGH RHR COOLING TOWERS.

THIS LINE PROVIDES PATH FOR THE REACTOR COOLANT TO EITHER OR BOTH DIVISIONS OF THE RHR SYSTEM. THE LINE INCLUDES TWO VALVES, AN INBOARD ISOLATION VALVE FED FROM DIVISION I AND AN OUTBOARD ISOLATION VALVE FED FROM DIVISION II.

TO ESTABLISH THE COOL DOWN PATH, THE OPERATOR HAS TO OPEN THE INBOARD AND OUTBOARD ISOLATION VALVES.

IF THE CONDENSER IS NOT AVAILABLE (LOSS OF OFFSITE POWER OR ISOLATION), DECAY HEAT CAN BE REMOVED VIA THE SAFETY RELIEF VALVES TO THE SUPPRESSION POOL. ONCE THE REACTOR IS DEPRESSURIZED BELOW 100 PSIG, THE DECAY HEAT CAN BE REMOVED UTILIZING THE SHUTDOWN COOLING MODE OF THE RHR SYSTEM.

DEGRADED MODE LOSS OF OFFSITE POWER, WITH POSTULATED FAILURE OF DIVISIONAL EMERGENCY POWER, RESULTS IN LOSS OF THIS PREFERRED NORMAL COOLING PATH.

THE FERMI 2 DESIGN INCLUDES TWO INBOARD ISOLATION VALVES FED FROM OPPOSITE DIVISIONS. THEREFORE IN THE DEGRADED MODE, IF A SINGLE FAILURE, LOSS OF DIVISION I, CAUSED THE FAILURE OF NORMAL INBOARD ISOLATION VALVE IN CLOSED POSITION, THE OPERATOR CAN OPEN PARALLEL VALVE FED FROM DIVISION II AND THE SHUTDOWN COOLING CONTINUES IN THE NORMAL MANNER.

IF A SINGLE FAILURE, LOSS OF DIVISION II CAUSED THE OUTBOARD ISOLATION VALVE TO FAIL IN CLOSED POSITION, A HAND WHEEL IS PROVIDED ON THE VALVE TO ALLOW NORMAL OPERATION. THE SHUTDOWN WOULD THEN CONTINUE IN A NORMAL MANNER.

IN THE HIGHLY UNLIKELY EVENT WHERE THE RHR SHUTDOWN COOLING PATH BECOMES UNAVAILABLE, AN ALTERNATE PATH IS AVAILABLE. AS DEPICTED IN THIS FIGURE, THIS ALTERNATE PATH UTILIZES A FLOW PATH FROM THE RPV THROUGH THE SAFETY RELIEF VALVES TO THE SUPPRESSION POOL. THE PRIMARY COOLANT CARRYING DECAY HEAT IS DISCHARGED THROUGH THE SAFETY RELIEF VALVE TO THE SUPPRESSION POOL. THE DECAY HEAT FROM THE SUPPRESSION POOL IS TRANSFERRED THROUGH THE RHR HEAT EXCHANGERS TO THE ATMOSPHERE.

CONTAINMENT COOLING MODE OF RHR TO REMOVE DECAY HEAT IS UTILIZED WHERE A LOSS OF COOLANT ACCIDENT INITIATES THE PLANT SHUTDOWN.

THE SIMPLIFIED SCHEMATIC SHOWN HERE DEPICTS THE BASIC OPERATION OF THE SYSTEM. IT CONSISTS OF ESTABLISHING A CLOSED COOLING PATH WITH THE SUPPRESSION POOL.

DECAY HEAT IS TRANSFERRED TO SUPPRESSION POOL IN THE EVENT OF A LOSS OF COOLANT ACCIDENT. HEAT FROM THE SUPPRESSION POOL IS TRANSFERRED THROUGH THE RHR HEAT EXCHANGERS TO THE

RHR SERVICE WATER, WHICH ULTIMATELY DISSIPATES THE DECAY HEAT TO THE ATMOSPHERE VIA RHR COOLING TOWERS.

THE SYSTEM IS CAPABLE OF MAINTAINING THE SUPPRESSION POOL TEMPERATURES BELOW 200°F UNDER THE CONDITIONS OF DESIGN BASIS ACCIDENT (DOUBLE ENDED BREAK IN THE SUCTION SIDE OF THE RECIRCULATION PIPE) WITH DEGRADED SYSTEM OPERATION.

THE SYSTEM CONSISTS OF TWO RHR HEAT EXCHANGERS AND FOUR RHR PUMPS. ONE HEAT EXCHANGER AND TWO PUMPS FORM A LOOP. TWO LOOPS ARE PHYSICALLY SEPARATED AND PROTECTED TO MINIMIZE THE POTENTIAL FOR SINGLE FAILURE CAUSING LOSS OF FUNCTION OF THE WHOLE SYSTEM.

OPERATION OF ONE OF THE FOUR RHR PUMPS, ONE OF TWO RHR HEAT EXCHANGERS AND TWO OF THE FOUR WATER PUMPS IS ADEQUATE TO REMOVE HEAT AND MAINTAIN SUPPRESSION POOL TEMPERATURE BELOW 200°F.

THEREFORE IT IS CONCLUDED THAT THE FERMI 2 DESIGN FOR REMOVAL OF DECAY HEAT IN NORMAL AND DEGRADED MODE IS ADEQUATE.

DECAY HEAT REMOVAL CAPABILITY - NORMAL AND DEGRADED MODES

- NORMAL MODE

  - MAIN CONDENSER
  - RHR SHUTDOWN COOLING MODE OF RHR

- DEGRADED MODE (MAIN CONDENSER UNAVAILABLE)

  - SAFETY/RELIEF VALVES
  - CONTAINMENT COOLING MODE OF RHR

- FURTHER DEGRADED MODE

  - RHR SHUTDOWN LINE INBOARD ISOLATION VALVE NON-OPERABLE
  - BOTH INBOARD OR ONE OUTBOARD ISOLATION VALVE NON-OPERABLE
  - DESIGN BASIS LOSS OF COOLANT ACCIDENT

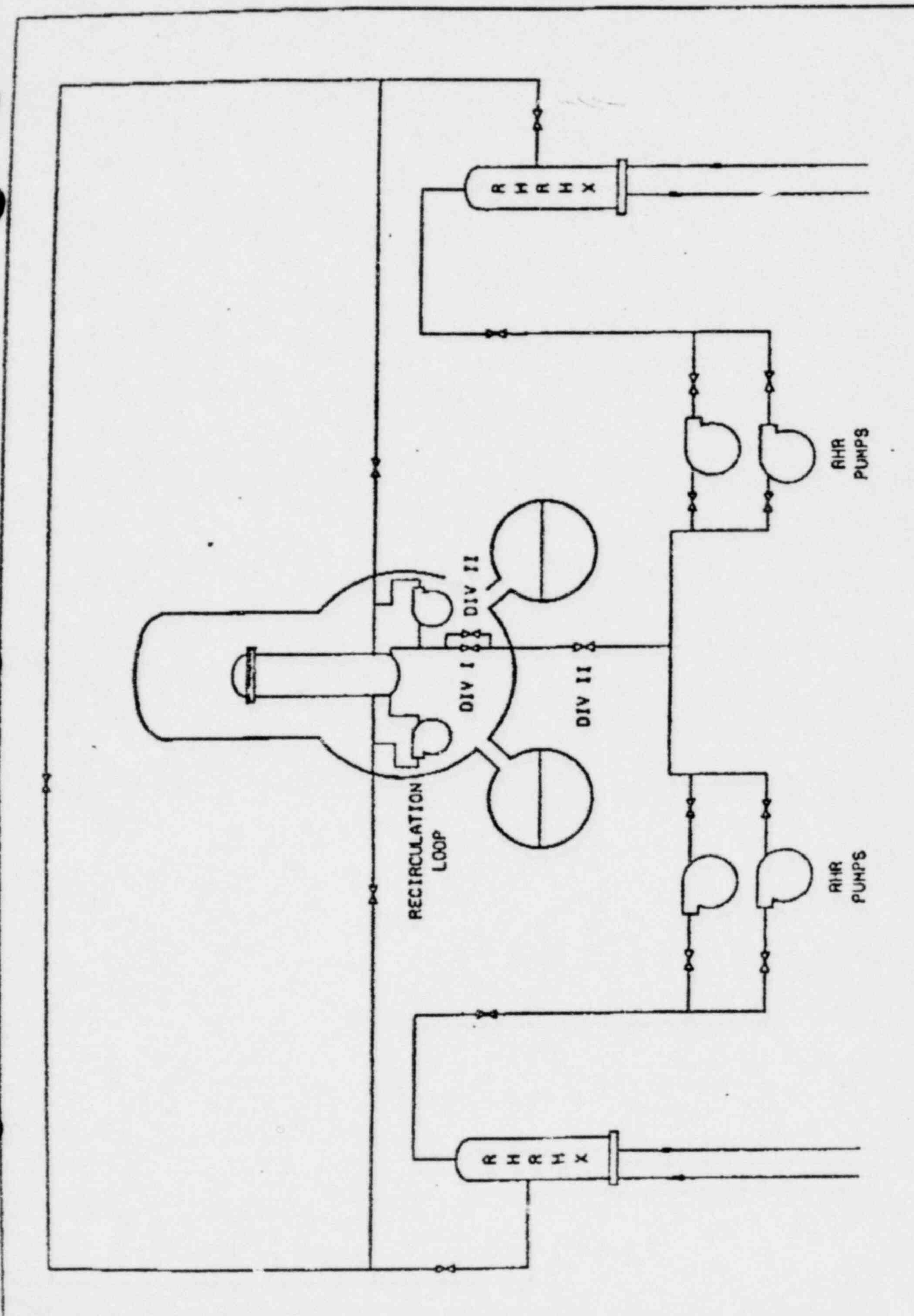


FIGURE 2. RHR SHUTDOWN COOLING MODE

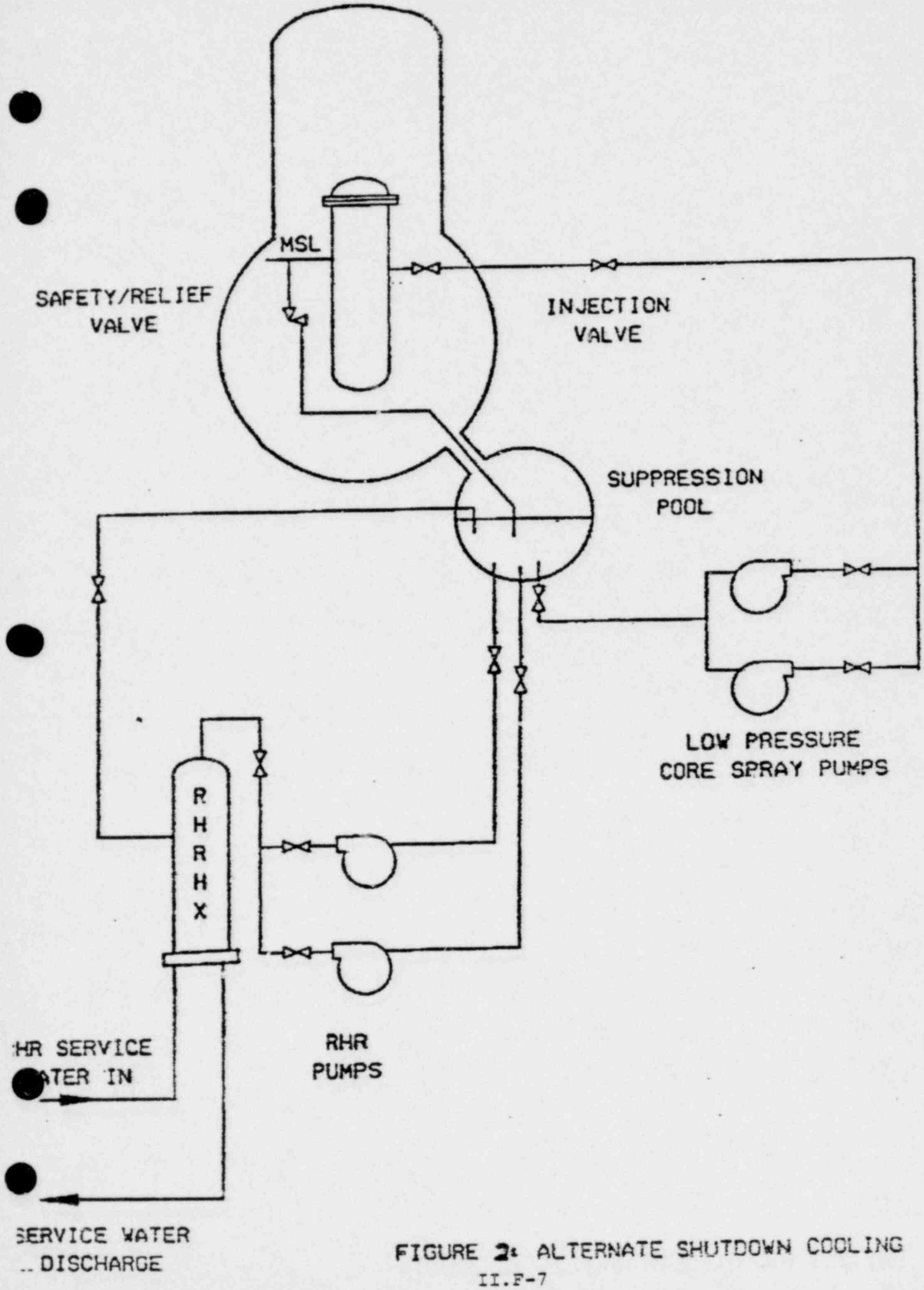
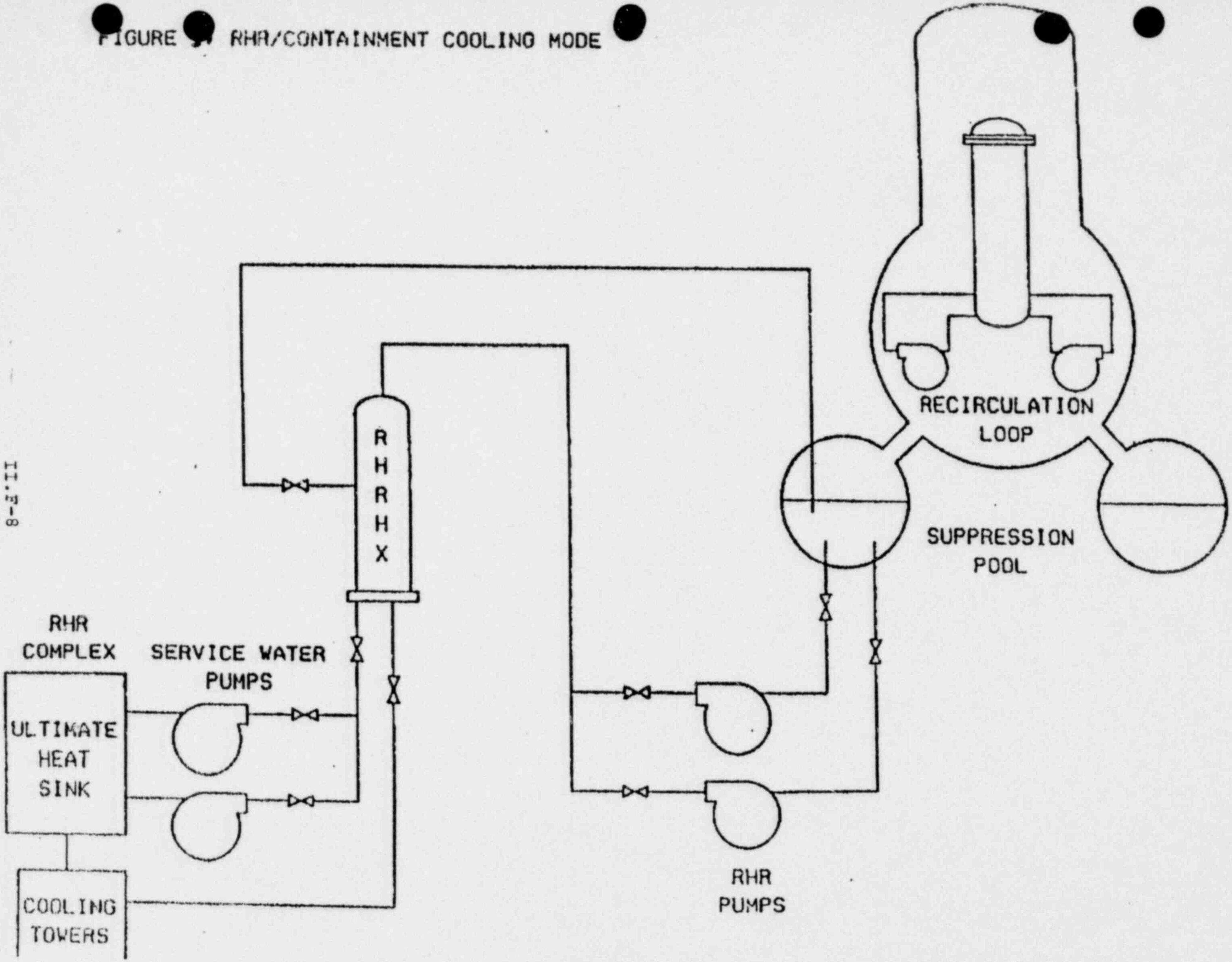


FIGURE 2: ALTERNATE SHUTDOWN COOLING

FIGURE [illegible] RHR/CONTAINMENT COOLING MODE



RELIABILITY OF STATION ELECTRICAL POWER

1. OFFSITE SOURCES

THE FERMI 2 PLANT IS CONNECTED TO TWO SEPARATE OFFSITE POWER SYSTEMS - THE 120 kV AND 345 kV TRANSMISSION SYSTEMS - BY MULTI-CIRCUITED TIES TO INCREASE RELIABILITY. (SEE FIGURE #1).

TWO 345 kV CIRCUITS ON SEPARATE TOWERS AT OPPOSITE SIDES OF THE SAME CORRIDOR CONNECT THE PLANT SWITCHYARD TO BROWNSTOWN STATION. IT THERE LINKS WITH THE SERVICE AREA 345 kV NETWORK WHICH INCLUDES FREE FLOW EXCHANGE WITH CONSUMERS POWER COMPANY.

THERE IS NO UNIT AUXILIARY TRANSFORMER IN THE FERMI 2 DESIGN. GENERATOR OUTPUT IS DELIVERED DIRECTLY TO THE 345 kV SWITCHYARD BY TWO HALF CAPACITY STEP-UP TRANSFORMERS. (FIGURE 3) THE SWITCHYARD ALLOWS FLEXIBLE OPERATION BY EMPLOYING A TWO BUS BREAKER AND A HALF SCHEME. PART OF THE NONSAFETY RELATED PLANT LOAD AND ALL OF DIVISION II ARE FED DIRECTLY FROM THIS SWITCHYARD VIA AN AUXILIARY STEPDOWN TRANSFORMER.

THREE 120 kV CIRCUITS COMPRISE THE OTHER SYSTEM AVAILABLE TO THE PLANT. EACH CIRCUIT IS SEPARATELY SUPPORTED AND SHARES A COMMON CORRIDOR WITH THE 345 kV CIRCUITS FOR FIVE

MILES. THE CIRCUIT SUPPORTS ARE SPACED TO ALLOW AT LEAST ONE CIRCUIT FROM EACH SOURCE TO REMAIN INTACT UPON FAILURE OF ANY SUPPORT. THEREAFTER THE 120 kV CIRCUITS SPREAD OUT IN INDIVIDUAL CORRIDORS TO SEPARATE STATIONS WITH EXTENSIVE LINKING TO THE SYSTEM AT SEVERAL VOLTAGE LEVELS OVER A LARGE AREA.

THE LINES TERMINATE AT THE 120 kV SWITCHYARD, ARRANGED AS A RADIAL DOUBLE FEED BUS, ONE QUARTER MILE AWAY FROM THE OTHER SWITCHYARD (FIGURE 2) THERE IT IS TRANSFORMED TO 13.8 kV AND BROUGHT TO FERMI 2 ENTIRELY UNDERGROUND. NEAR THE SWITCHYARD, FOUR 18.8 MVA GAS TURBINE PEAKING UNITS ARE TIED INTO THE 13.8 kV FEED TO FERMI 2. THE 13.8 FEEDS A SYSTEM SERVICE STEPDOWN TRANSFORMER WHICH SUPPORTS DIVISION I AND PART OF THE NONSAFETY-RELATED LOAD. (FIGURE 4)

2. AC POWER DISTRIBUTION

THE ENGINEERED SAFETY FEATURES SYSTEMS AT FERMI 2 ARE DIVIDED INTO TWO REDUNDANT AND INDEPENDENT DIVISIONS - DIVISIONS I & II. EACH DIVISION HAS AS ITS PREFERRED POWER SOURCE A SEPARATE OFFSITE SUPPLY. (SEE FIGURE 4)

FOUR 2850 kW EMERGENCY DIESEL GENERATORS (EDG) PROVIDE ON SITE AC POWER IN THE EVENT OF LOSS OF OFFSITE POWER. THERE ARE TWO EDG'S PER DIVISION WITH THE GROUP OF BUSES ASSOCIATED

WITH EACH EDG (AND INDEPENDENT OF THE OTHER GROUPS) KNOWN AS A LOAD GROUP. WHILE BOTH LOAD GROUPS WITHIN A DIVISION ARE NECESSARY TO PROVIDE TRUE REDUNDANCY, NORMALLY OPEN INTRA-DIVISIONAL CROSS TIES ENABLE CRITICAL LOADS TO BE THROWN OVER IN THE EVENT ONE EDG IS NOT AVAILABLE. WITHIN A DIVISION, EMERGENCY CORE COOLING LOADS ARE SHARED BETWEEN LOAD GROUPS. PLANT AUXILIARY SERVICES SUCH AS AIR HANDLING, CLOSED LOOP COOLING, ETC., ARE CONFINED TO A PARTICULAR LOAD GROUP. INSTRUMENT POWER SUPPLY FEEDS CAN THROW OVER BETWEEN LOAD GROUPS. EDG AUXILIARY SERVICES ARE PROVIDED BY THEIR RESPECTIVE LOAD GROUPS.

EACH LOAD GROUP INCORPORATES RADIAL BUS ARRANGEMENT. TWO 4 kV BUSES WITHIN THE LOAD GROUP ARE TIED TOGETHER. ONE CONNECTS TO THE SYSTEM SERVICE POWER TRANSFORMER VIA CABLE BUS FOR THE OFFSITE SOURCE. THE OTHER CONNECTS TO THE EDG. LARGE LOADS ARE POWERED FROM THESE BUSES. EACH LOAD GROUP CONTAINS TWO 480 VOLT BUSES - ONE DERIVED FROM EACH OF THE 4 kV BUSES. SMALLER LOADS, INCLUDING THOSE DOWN TO 120 VOLTS, ARE FED FROM THESE BUSES OR THOSE THEREAFTER.

3. BUS LOSS CONSEQUENCES

THE DESIGN BASIS FOR FERMI 2 IS TO OPERATE ON THE PREFERRED POWER SOURCE WHEN AVAILABLE. WHEN NOT AVAILABLE, TWO LOAD

GROUPS WITHIN A DIVISION ARE REQUIRED. EACH DIVISION'S SAFETY-RELATED EQUIPMENT IS COMPLETELY REDUNDANT TO THE OTHER.

LOSS OF DIVISION I AC POWER SUPPLY WILL RESULT IN LOSS OF THE FOLLOWING EQUIPMENT:

1. RHR LOOPS A & C

2. CORE SPRAY LOOPS A & C

3. INBOARD ISOLATION VALVES

4. ONE REDUNDANT SYSTEM OF CONTROL & EQUIPMENT ROOM VENTILATION

5. ONE REDUNDANT SYSTEM OF STANDBY GAS TREATMENT

6. ONE REDUNDANT SYSTEM OF COMBUSTION GAS CONTROL

LOSS OF DIVISION II POWER SUPPLY WILL RESULT IN LOSS OF THE FOLLOWING CRITICAL EQUIPMENT:

1. RHR LOOPS B & D

2. CORE SPRAY LOOPS B & D

3. ONE REDUNDANT SYSTEM OF CONTROL & EQUIPMENT ROOM VENTILATION

4. ONE REDUNDANT SYSTEM OF STANDBY GAS TREATMENT

5. ONE REDUNDANT SYSTEM OF COMBUSTION GAS CONTROL

4. DC ONSITE POWER SYSTEM

THE DC ONSITE POWER SYSTEM FOR ENRICO FERMI UNIT 2 CONSISTS OF THE FOLLOWING FULLY INDEPENDENT BATTERY SYSTEMS:

A. DIVISION 1 260/130V BATTERY 2 PA (ESF LOADS ONLY)

B. DIVISION 2 260/130V BATTERY 2 PB (ESF LOADS ONLY)

C. BOP 260/130V BATTERY 2PC (NON-ESF LOADS)

D. 48/24V BATTERY 21A (NON-ESF INSTRUMENT LOAD)

E. 48/24V BATTERY 21B (NON-ESF INSTRUMENT LOAD)

BATTERIES C, D AND E FEED ONLY NON-ESF LOADS SUCH AS MAIN TURBINE AUXILIARIES, THE ANNUCIATOR, AND NON-ESF CONTROL AND INSTRUMENTATION. LOSS OF ANY OF THESE SYSTEMS WILL

NOT IMPAIR THE ABILITY TO SAFELY SHUTDOWN THE REACTOR.

A TYPICAL SYSTEM DIAGRAM OF AN ESF 260/130V CENTER TAPPED BATTERY DESIGN AT ENRICO FERMI 2 IS SHOWN IN FIGURE #5. BOTH POWER FOR MOTOR-OPERATED VALVES AND SMALL AUXILIARY PUMPS, AND CONTROL POWER REQUIRED BY THE DIVISION ARE SUPPLIED BY THE SAME BATTERY.

LOSS OF THE DIVISION 1 260/130V BATTERY WOULD RESULT IN A LOSS OF THE FOLLOWING ESF FUNCTIONS:

a. RCIC SYSTEM AND ITS AUXILIARIES

b. DEPRESSURIZATION UTILIZING THE AUTOMATIC SRV'S

c. CONTROL POWER TO DIVISION 1 ESF SWITCHGEAR AND EDG'S

d. CONTROL POWER TO THE DIVISION 1 ECCS SYSTEMS.

IF ACCIDENT CONDITIONS EXIST SIMULTANEOUSLY WITH THE LOSS OF THE DIVISION 1 BATTERY SYSTEM, REDUNDANT EQUIPMENT IN DIVISION 2 WOULD PERFORM THE REQUIRED SAFETY FUNCTIONS.

IF THE DIVISION 2 260/130V BATTERY SYSTEM WERE TO BE LOST DURING AN ACCIDENT, THE FOLLOWING ESF FUNCTIONS WOULD BE UNAVAILABLE:

A. HPCI SYSTEM AND ITS AUXILIARIES.

B. OUTBOARD ISOLATION VALVES.

C. CONTROL POWER TO DIVISION 2 ESF SWITCHGEAR AND EDG'S.

D. CONTROL POWER TO DIVISION 2 ECCS SYSTEMS.

THE DIVISION 1 BATTERY AND DIVISION 1 AC WOULD POWER REDUNDANT EQUIPMENT PERFORMING THE REQUIRED SAFETY FUNCTIONS.

5. THE ACTION FOR RESTORING OFFSITE AC POWER IN THE EVENT OF A LOSS OF THE GRID

DETROIT EDISON COMPANY HAS AN "EMERGENCY ELECTRICAL PROCEDURES OPERATING GUIDE" WHICH DESCRIBES THE PROCEDURES TO BE FOLLOWED BY THE ELECTRICAL SYSTEM SHIFT SENIOR SYSTEM SUPERVISOR IN ORDER TO MAINTAIN SYSTEM RELIABILITY FOR CASES OF A DEFICIENT BULK POWER SUPPLY. IN CASE OF A COMPLETE SYSTEM SHUTDOWN, THE GUIDE PROVIDES VARIOUS METHODS FOR RESTORING THE ELECTRICAL SYSTEM TO NORMAL IN AS SHORT A TIME AS POSSIBLE. ONE FEATURE

OF THE GUIDE IS THAT EACH DETROIT EDISON GENERATING FACILITY WITH "BLACK START CAPABILITY" WILL INITIATE THEIR OWN RETURN TO NORMAL GENERATING MODE. THEN AUXILIARY POWER WILL BE RESTORED TO THE OTHER DETROIT EDISON GENERATING FACILITIES, INCLUDING FERMI 2 BY RE-ENERGIZING THE TRANSMISSION NETWORK IN A SEQUENCE WHICH WOULD VARY BECAUSE OF THE AREA ENCOMPASSED BY THE BLACKOUT CONDITION. AFTER THIS IS ACCOMPLISHED, THE FULL TRANSMISSION NETWORK AND CUSTOMER LOADS WILL BE RESTORED IN SMALL INCREMENTS.

RESTORATION OF OFFSITE POWER TO FERMI 2 IS EXPECTED TO BE ACCOMPLISHED BY THE BLACK START OF TRENTON CHANNEL GENERATING FACILITY AND THE RESTORATION OF THE 120 kV NETWORK SUFFICIENTLY TO SUPPLY FERMI 2 VIA ONE OF THE 120 kV LINES TO INTO FERMI 1 SWITCHYARD (SEE FIGURE 6) THE 345 kV OFFSITE SOURCE WOULD BE RESTORED BY RE-ENERGIZING THE 345 kV AT BROWNSTOWN STATION VIA ONE OF DETROIT EDISON GENERATING FACILITIES OR FROM OUR 345 kV INTERCONNECTING TIES TO THE NEIGHBORING UTILITY SYSTEM. THE TIME REQUIRED TO RESTORE THE FERMI 2 OFFSITE POWER SUPPLIES WILL VARY WITH THE NATURE OF THE GRID BLACK-OUT AND THE EXTENT OF ANY ASSOCIATED DAMAGE TO THE TRANSMISSION LINES. IN ANY EVENT, THE TOTAL SYSTEM RESTORATION IS EXPECTED TO BE COMPLETED WITHIN A DAY. THE INDUSTRY EXPERIENCE IN RESTORING SYSTEMS HAS BEEN LESS THAN ONE DAY.

A LESS EXTENSIVE GRID DISTURBANCE COULD ISOLATE FERMI 2 WITH THE TOTAL LOSS OF OFFSITE POWER THROUGH THE UNLIKELY SIMULTANEOUS OR OVERLAPPING OUTAGES OF THE 120 kV STATIONS; BROWNSTOWN, CUSTER AND LUZON AND THE 345 kV FEEDS FROM BROWNSTOWN STATION OR BY THE LOSS OF THE FERMI 2 TRANSMISSION CORRIDOR. THE RESTORATION TIME OF ANY ONE LINE, FOR SUCH AN EVENT, WOULD VARY CONSIDERABLY WITH THE EXTENT OF THE DAMAGE ASSOCIATED WITH THE OUTAGE. AN INDICATION OF TIME TO RESTORE ANY ONE LINE IS THE AVERAGE RESTORATION TIME OF ALL DETROIT EDISON TRANSMISSION LINE LOCKOUT OUTAGES (EXCLUDING MOMENTARY OUTAGES WITH AUTOMATIC RECLOSING). THE AVERAGE RESTORATION TIME FOR A 120 kV LINE LOCKOUT OUTAGE IS 13 HOURS. THE AVERAGE RESTORATION TIME FOR A 345 kV LINE LOCKOUT OUTATE IS 9.3 HOURS ON THE DETROIT EDISON TRANSMISSION SYSTEM.

FOR THOSE INCIDENTS IN WHICH SUBSTANTIAL DAMAGE IS DONE TO TRANSMISSION STRUCTURES, DETROIT EDISON HAS EMERGENCY PROCEDURES IN WHICH TEMPORARY STRUCTURES WOULD BE USED TO RESTORE THE TRANSMISSION LINE IN AS SHORT A TIME AS POSSIBLE. DETROIT EDISON CONSTRUCTION AND CONTRACT CREWS WOULD BE USED TO MAKE THE NECESSARY REPAIRS.

6. LOSS OF OFFSITE AC POWER DUE TO ONSITE EQUIPMENT FAILURES

DUE TO THE INDEPENDENCE OF ENRICO FERMI 2 OFFSITE POWER SOURCES AND THE METHOD OF ONSITE DISTRIBUTION OF POWER,

NO SINGLE EQUIPMENT FAILURE CAN CAUSE THE LOSS OF BOTH OFFSITE AC POWER SORUCES.

THE MOST CREDIBLE EVENT LIKELY TO CAUSE A LOSS OF ALL OFFSITE POWER IS A TORNADO THAT STRIKES THE COMMON TRANSMISSION CORRIDOR FOR THE 345 kV AND 120 kV LINES LEAVING THE SITE. SHOULD THIS HIGHLY IMPROBABLE EVENT OCCUR THE FOUR INDEPENDENT DIESEL GENERATORS WOULD AUTOMATICALLY START AND LOAD RESTORING POWER TO THE ESF BUSES.

RESTORATION OF AN OFFSITE POWER SOURCE WOULD BE ACCOMPLISHED AS DESCRIBE PIN PRECEEDING PART 5.

7. STATION BLACKOUT

DETROIT EDISON DOES NOT BELIEVE THAT A COMPLETE LOSS OF AC POWER IS A CREDIBLE POSSIBILITY AT FERMI 2. COMBINING THE LOSS OF THE TWO INDEPENDENT OFFSITE SOURCES WITH THE FAILURE OF FOUR INDEPENDENT DIESEL GENERATORS GOES WELL BEYOND REQUIRED PROBABILITIES.

EVEN WITH SUCH INCREDIBLE EVENTS A UNIQUE FEATURE OF THE FERMI 1 SWITCHYARD CAN SUPPLY ADEQUATE POWER TO ONE OF THE FERMI 2 ESF DIVISIONS. FOUR COMBUSTION TURBINE GENERATORS,(CTG) RATED 18.8 MVA EACH, USED FOR PEAKING PURPOSES, ARE LOCATED JUST OUTSIDE THE FERMI 1 120 kV SWITCHYARD. COMBUSTION TURBINE

GENERATORS #1 (SEE FIGURE 4) IS EQUIPPED WITH A BLACK START FEATURE CAPABLE OF BEING INITIATED EITHER FROM THE FERMI 2 CONTROL ROOM OR LOCALLY AT THE PEAKING UNIT ITSELF. THIS UNIT IS CAPABLE OF STARTING AND ACCEPTING LOAD WITHIN TEN MINUTES.CTG #1 HAS ADEQUATE CAPACITY TO EASILY POWER DIVISION I ESF LOADS REQUIRED TO ENSURE SAFE SHUTDOWN OF THE REACTOR.

IN THE TIME PERIOD BETWEEN REALIZATION BY THE OPERATOR THAT NO EDG'S ARE AVAILABLE AND THE LOADING OF CTG #1, ADEQUATE ACTION CAN BE TAKEN UTILIZING SYSTEMS POWERED ONLY BY DC. EMERGENCY OPERATING PROCEDURES AT FERMI 2 COVER THE ACTIONS NECESSARY TO MAINTAIN REACTOR VESSEL WATER LEVEL USING THE STEAM-DRIVEN, DC-POWERED, HPCI AND RCIC SYSTEMS. IN ADDITION, ANY ACTIONS NECESSARY TO RESTORE ONSITE POWER, INCLUDING RESTART OF THE EMERGENCY DIESELS AND BLACK START OF CTG #1, WILL BE INCLUDED IN THE ABNORMAL OPERATING PROCEDURES.

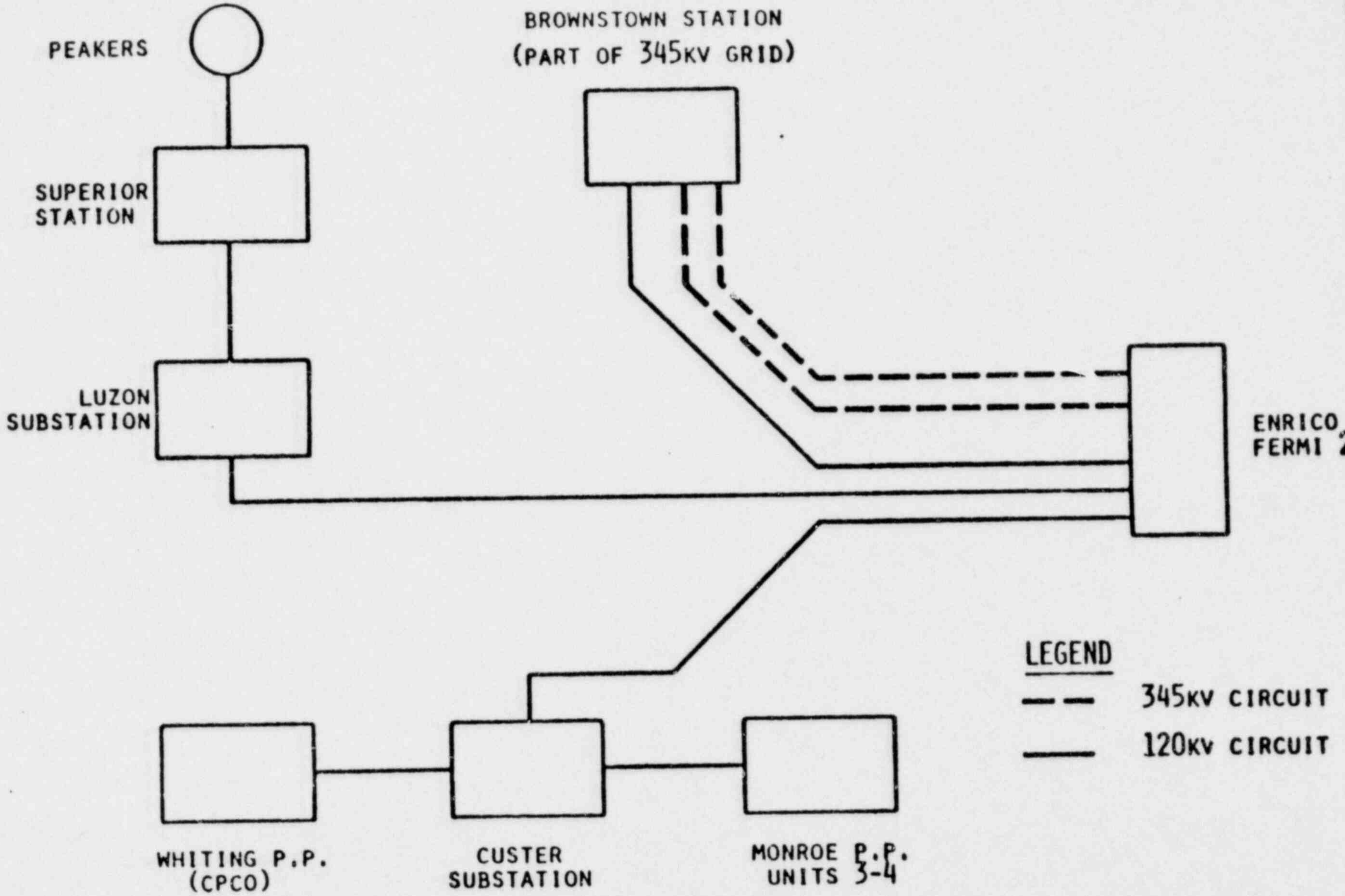


FIG. 1 ENRICO FERMI 2 OFFSITE POWER SOURCES

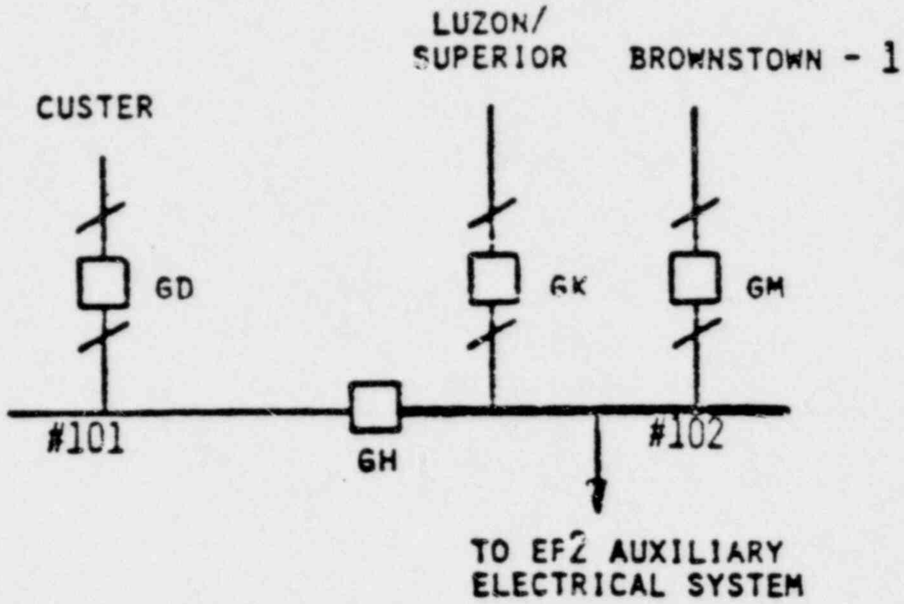


FIG. 2 ENRICO FERMI I 120KV SWITCHYARD

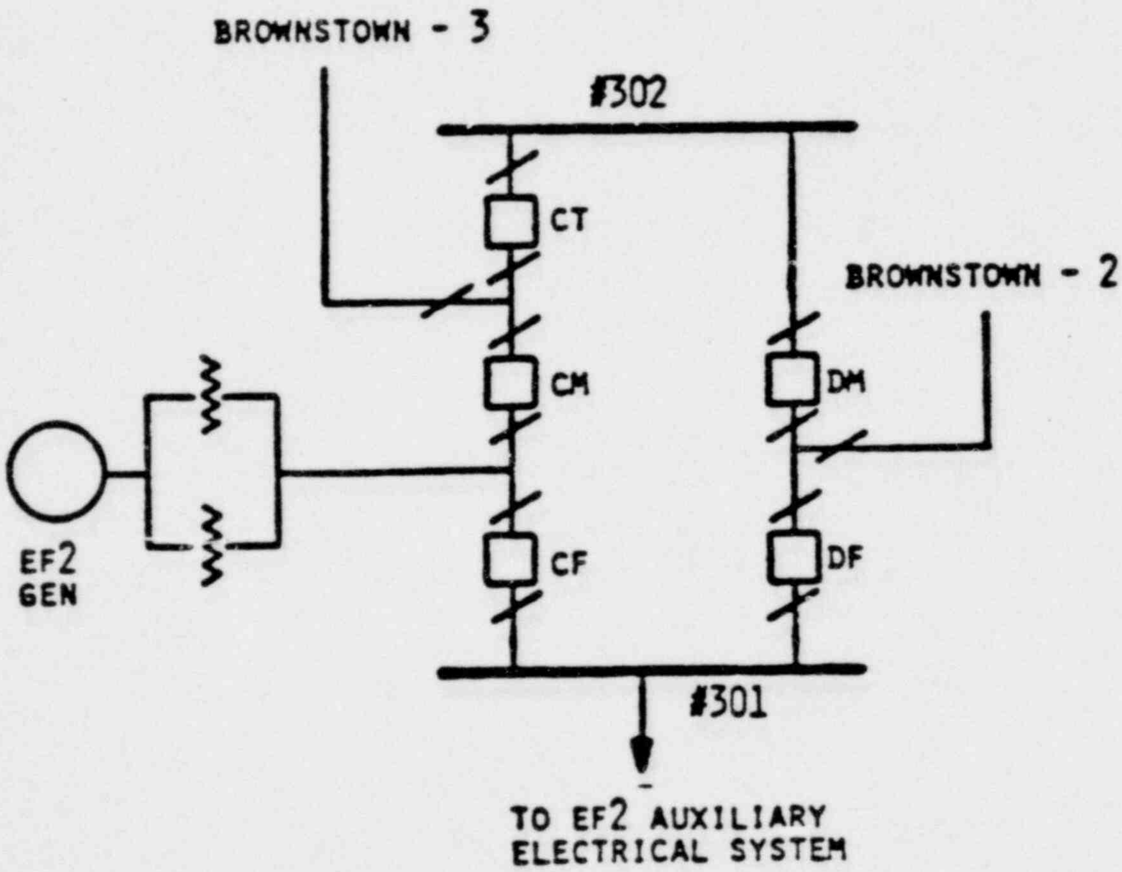


FIG. 3 ENRICO FERMI 2 345KV SWITCHYARD

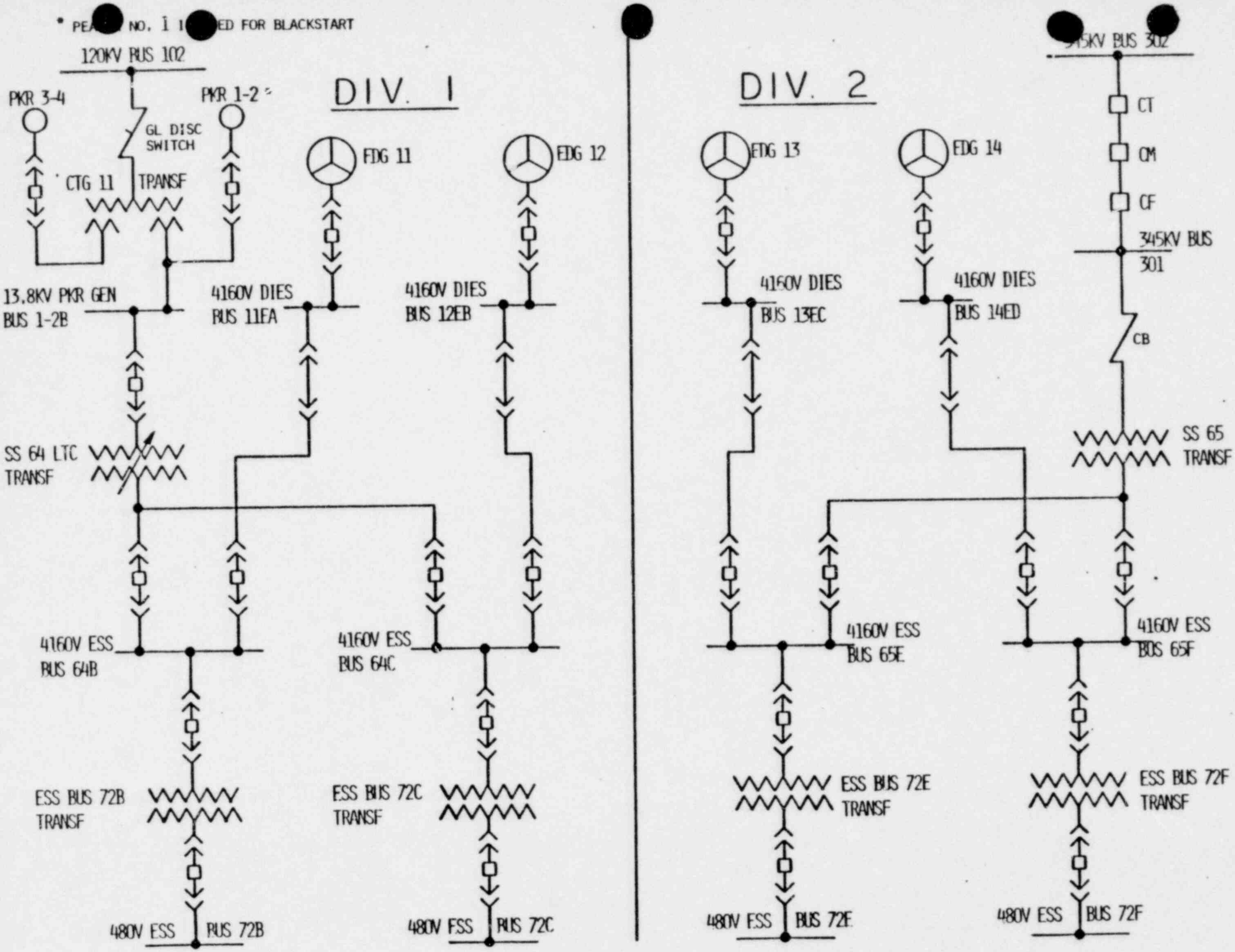
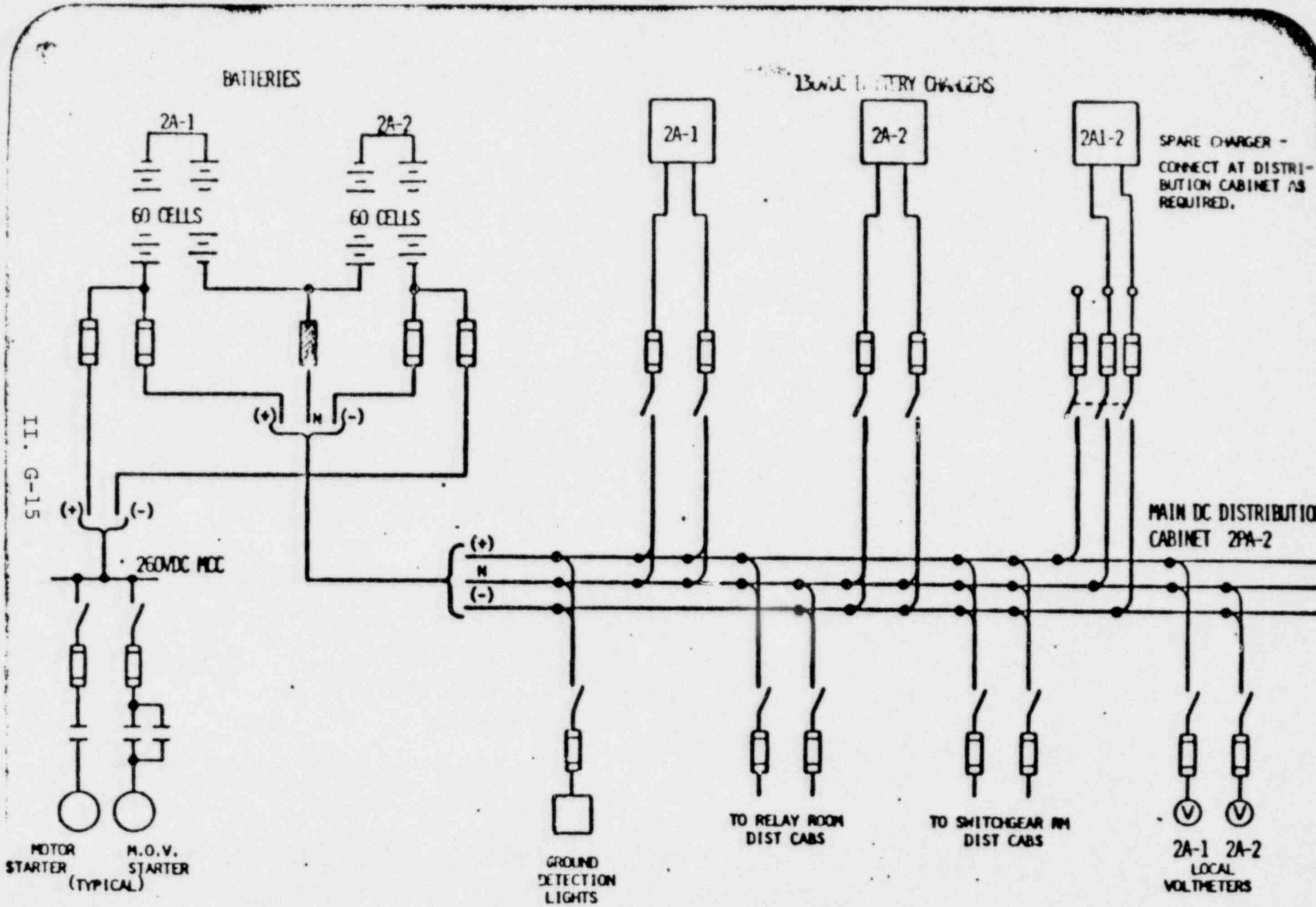


FIG. 4 ENRICO FERMI 2 AUXILIARY ELECTRICAL SYSTEM

II. G-14



II. G-15

FIG. 5 ENRICO FERMI 2 260/130VDC DISTRIBUTION SYSTEM

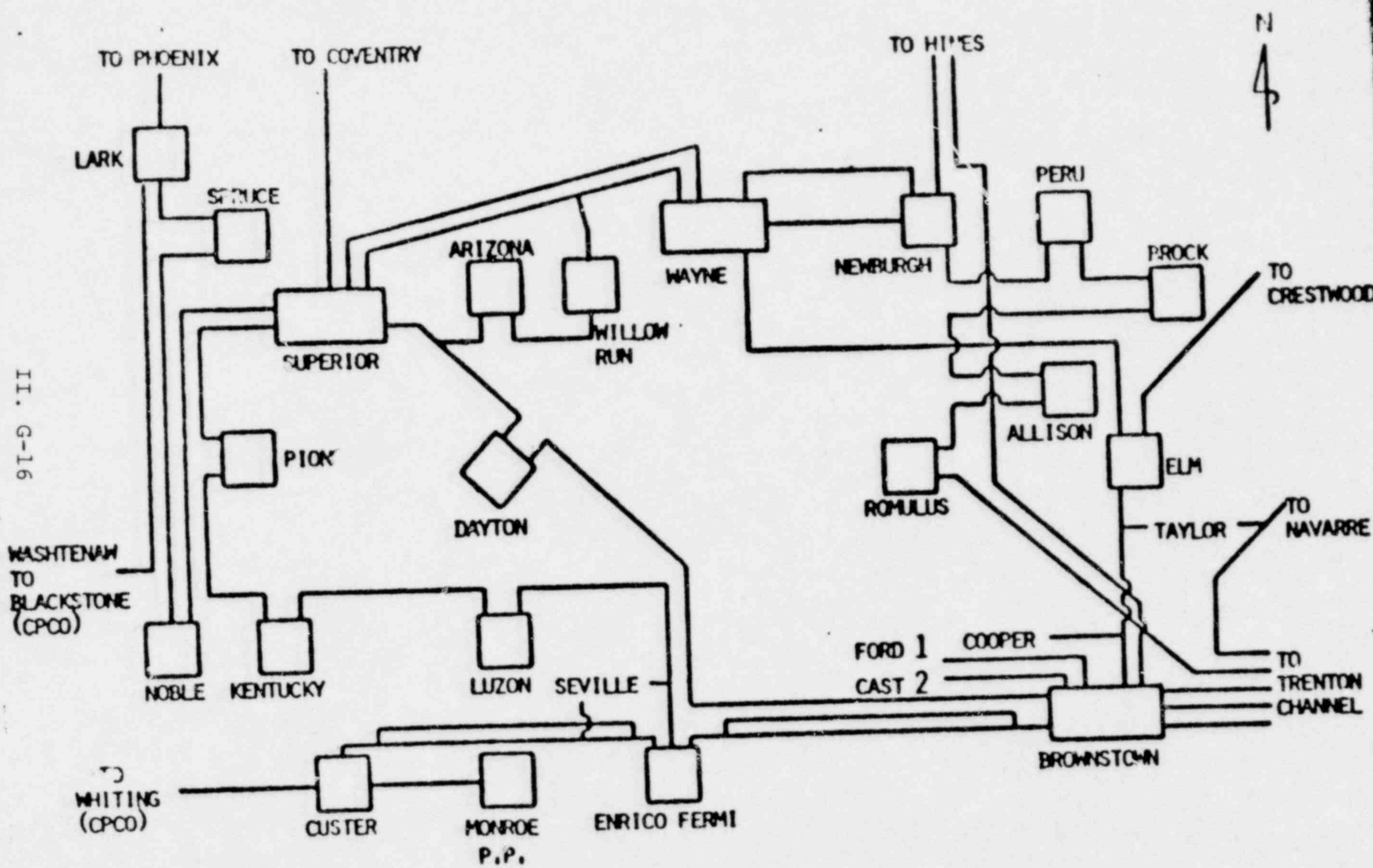


FIG. 6 120KV TRANSMISSION IN 1983 - SOUTH AREA

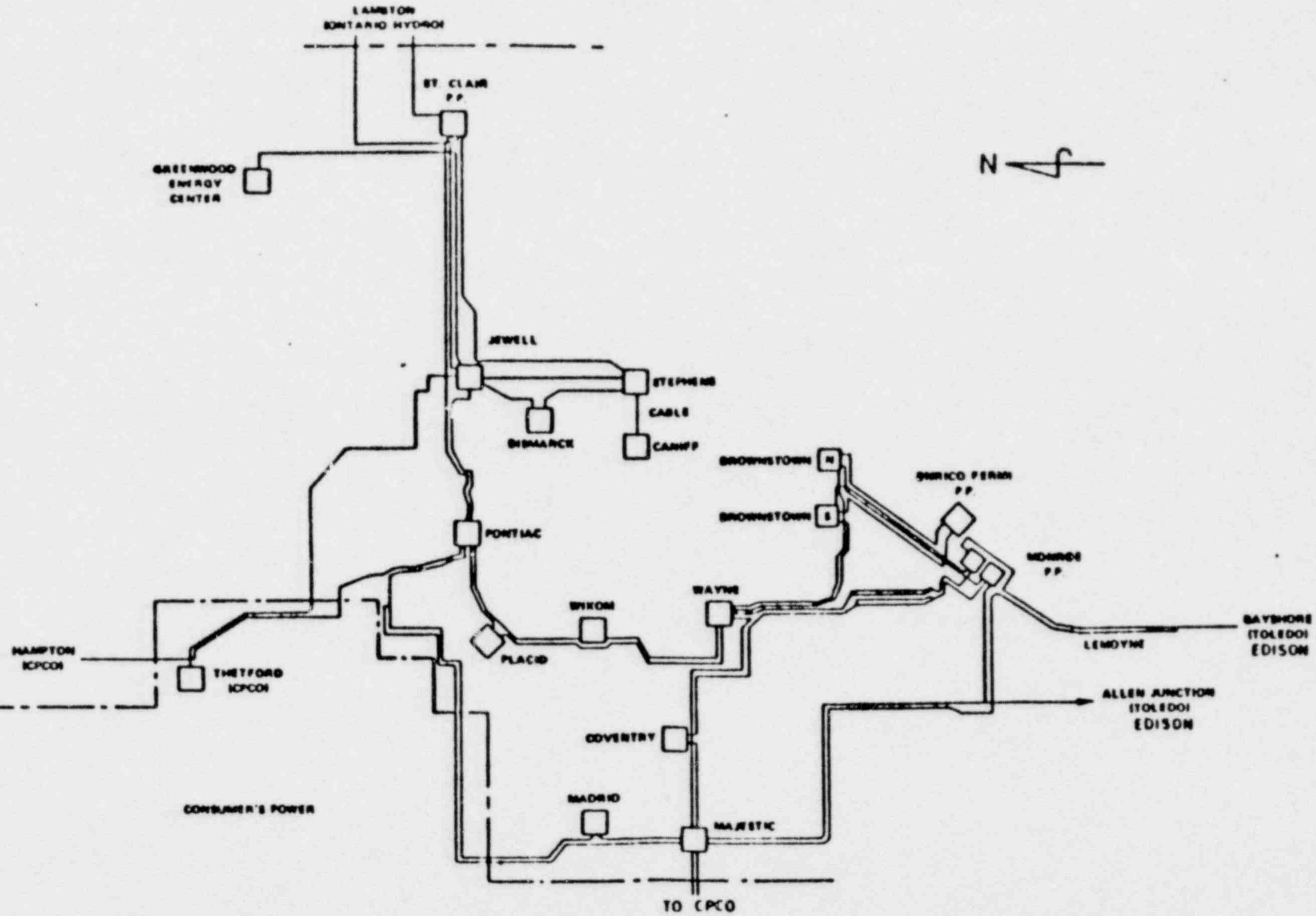
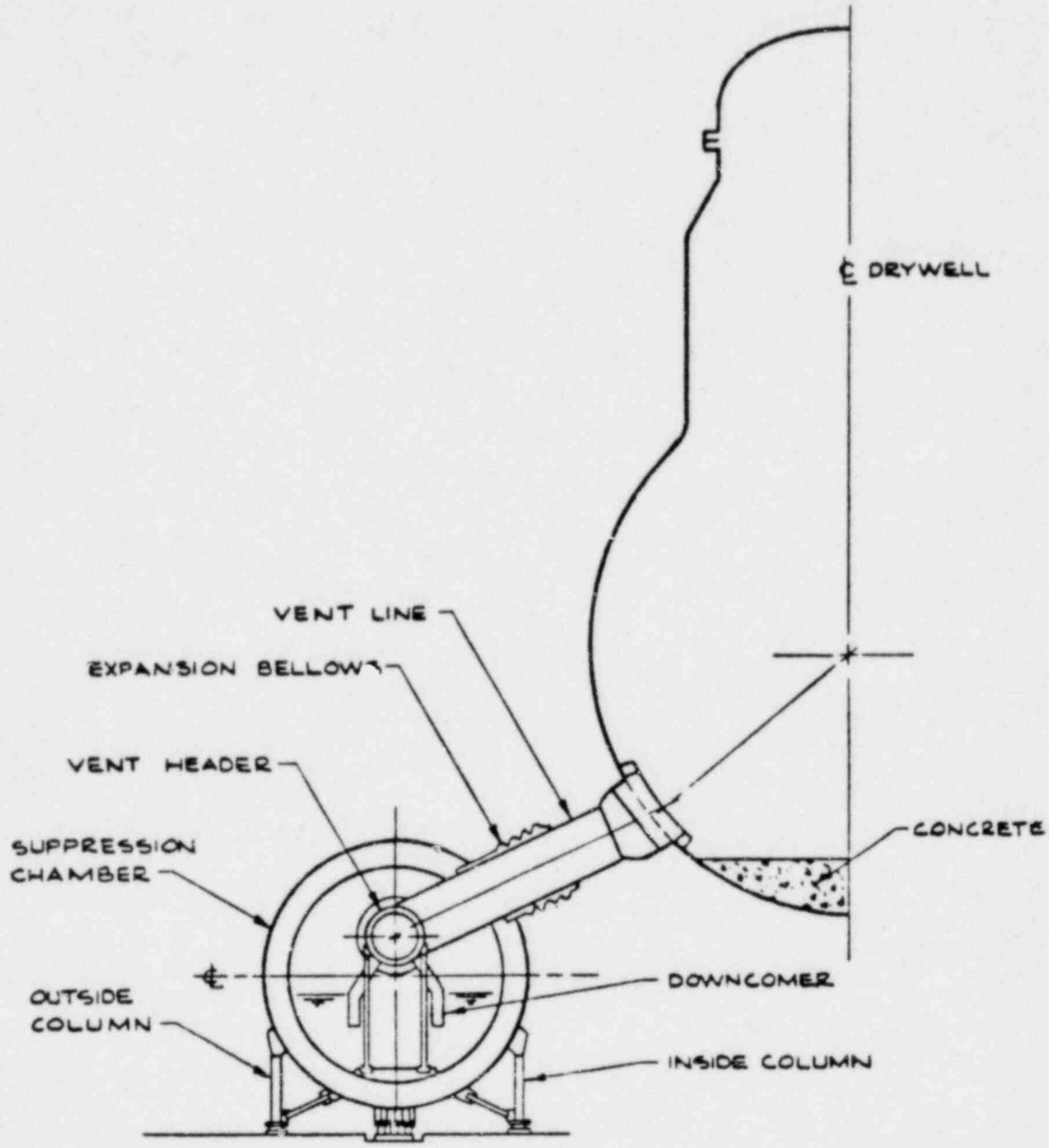


FIG 7 345KV TRANSMISSION IN 1983



GENERAL ARRANGEMENT OF CONTAINMENT - SCHEMATIC

COMPLETED MODIFICATIONS

CATEGORY			DESCRIPTION	APPROX. MOD. DATES
MAJOR	TORUS		RING GIRDER REINFORCEMENT	6/79
MAJOR			COLUMN REINFORCEMENT	10/78
MAJOR			COLUMN CONNECTION REINFORCEMENT	12/79
MINOR	VENT SYSTEM		DOWNCOMER SHORTENING	2/80
MAJOR			VENT HEADER/DOWNCOMER STIFFENING & BRACING	11/78
MAJOR			REINFORCED EXISTING VENT SYSTEM COLUMNS & CONNECTIONS	2/79
MAJOR			VENT HEADER DEFLECTOR	2/80
MAJOR			VENT LINE/VENT HEADER STIFFENING	6/79
MAJOR			REINFORCED VACUUM BREAKER TO VENT HEADER CONNECTION	7/79
MAJOR	INTERNAL STRUCTURES	MONORAIL	ADDITIONAL SUPPORTS	5/78
MAJOR			STRENGTHEN EXISTING SUPPORTS	5/78
MINOR			EXTENDING MONORAIL	5/78
MAJOR		CATWALK	ADDITIONAL SUPPORTS	8/78
MAJOR			GRATING (DELIVER TO SITE)	3/80
MAJOR	SRV PIPING		REROUTED PIPING IN WETWELL	4/80
MAJOR			ADDITIONAL WETWELL SUPPORTS	4/79
MAJOR			REINFORCED V. L. PENETRATION	11/78
MAJOR			ADDED QUENCHER/RAMSHEAD SUPPORTS	1/80
MINOR	TORUS ATTACHED PIPING		ADDED TORUS INTERNAL SUPPORTS	4/80

CURRENT MODIFICATIONS

CATEGORY		DESCRIPTION	EST. COMPLETION	REMARKS
MAJOR	TORUS	MITERED JOINT SADDLES	2/82	FABRICATIONS AND FIELD WORK UNDERWAY
MAJOR		ADDITIONAL COLUMN ANCHOR BOLTS	4/82	FABRICATION AND FIELD WORK UNDERWAY
MAJOR	SRV PIPING	QUENCHER	9/82	MATERIAL ORDERED
MAJOR		QUENCHER SUPPORTS	9/82	MATERIAL ORDERED
MINOR	INTERNAL STRUCTURES	CATWALK GRATING INSTALLATION	9/32	

RESPONSE TO NUREG-0588 EQUIPMENT ENVIRONMENTAL QUALIFICATION

EQUIPMENT COVERED

- o CLASS IE EQUIPMENT REQUIRED TO MITIGATE A POSTULATED ACCIDENT.
- o INSTALLED TMI LESSON-LEARNED EQUIPMENT.
- o DISPLAY INSTRUMENTATION MENTIONED IN THE EMERGENCY OPERATING PROCEDURES WHICH IS ASSOCIATED WITH SAFETY-RELATED SYSTEMS, POST-ACCIDENT SAMPLING AND MONITORING, AND RADIATION MONITORING.
- o EQUIPMENT ASSOCIATED WITH ONE PREFERRED PATH TO BRING THE PLANT TO MAINTAIN IT AT THE COLD SHUT-DOWN CONDITION.
- o CLASS 1E EQUIPMENT, LOCATED OUTSITE THE PRIMARY CONTAINMENT WHICH IS EFFECTED BY HIGH RADIATION DURING POST-LOCA RECIRCULATION OF CONTAINMENT FLUIDS.

REVIEW CONCENTRATES ON EQUIPMENT EXPOSED TO HARSH ENVIRONMENTS. (LOCA/HELB INSIDE AND HELB OUTSIDE THE PRIMARY CONTAINMENT.)

SINCE FERMI 2 CP SER WAS ISSUED PRIOR TO JULY 1, 1974, THE REVIEW WAS BASED ON GUIDELINES OF NUREG-0588 CAT. 2.

FERMI 2 EQUIPMENT QUALIFICATION ASSESSMENT PROGRAM

* PERSONNEL

  o NUTECH, INC.: (CONTRACTOR) NSSS PORTION

  o WYLE LABS: (SUB-CONTRACTOR) NON-NSSS PORTION

  o DETROIT EDISON: OVERALL PROGRAM ADMINISTRATION
    PLANT OPERATING/EMERGENCY
    PROCEDURES

* OUTLINE OF PROGRAM

  o DEVELOPMENT OF A MASTER EQUIPMENT LIST: FSAR, TECHNICAL SPECIFICATION, EMERGENCY OPERATING PROCEDURES, P&ID'S, ELECTRICAL SINGLE LINE AND SCHEMATIC DIAGRAMS.

  o RETRIEVAL OF QUALIFICATION RECORDS: GE, DECO., EPRI BWR OWNER'S GROUP, WYLE LABS, EPRI EQUIPMENT QUALIFICATION DATA BANK AND VENDOR FILES. DEMONSTRATED DATA IS SUMMARIZED IN QSR'S.

  o EVALUATION OF DATA: AGAINST REQUIREMENTS OF NUREG-0588 CAT. 2 AND FERMI 2 ENVIRONMENTAL CONDITIONS. EVALUATION IS DOCUMENTED IN QER'S.

- o PREPARATION OF QUALIFICATION SUMMARY SHEETS: FOR THE SUBMITTAL. INCLUDED SYSTEM/SUBSYSTEM, EQUIPMENT LOCATION, FLOOD LEVEL, SAFETY FUNCTION, REQUIRED NORMAL CONDITIONS VS. DEMONSTRATED, REQUIRED ACCIDENT CONDITIONS VS. DEMONSTRATED, OPERABILITY REQUIRED VS. DEMONSTRATED, QUALIFICATION REPORT & METHOD, QUALIFICATION STATUS AND REFERENCES.

* QA CONSIDERATIONS

  - o APPROVED QA PROGRAM FROM WHYLE NUTECH.

  - o SUBMITTAL WILL BE UPDATED PERIODICALLY.

* DECO VERIFICATION REVIEW

<u>ENVIRONMENTAL PROFILES</u>

* TEMPERATURE, PRESSURE AND RELATIVE HUMIDTY:

  - o FOR ALL ROOMS AND AREAS CONTAINING 1E EQUIPMENT.

  - o INSIDE THE PRIMARY CONTAINMENT: NORMAL AND ACCIDENT (LOCA/HELB)

- o OUTSIDE THE PRIMARY CONTAINMENT: FIVE CONDITIONS, NORMAL OPERATION, DURING TESTING OF EMERGENCY EQUIPMENT, LOSS OF HVAC DURING NORMAL OPERATION, HELB AND BOUNDING CONDITIONS DURING A DBA-LOCA.

* RADIATION

  - o FOR ALL ROOM AND AREAS CONTAINING 1E EQUIPMENT.

  - o WORST VALVE FOR EACH AREA.

  - o INCLUDES GAMMA AND BETA.

SUMMARIES OF METHODOLOGY AND ASSUMPTIONS USED TO DEVELOP ENVIRONMENTAL PROFILES WERE DESCRIBED OR ATTACHED WITH THE SUBMITTAL.

<u>DEFICIENCIES AND CORRECTIVE ACTIONS</u>

* DEFICIENCIES ARE IDENTIFIED ON QUALIFICATION SUMMARY SHEETS

* CORRECTIVE ACTIONS WILL BE TAKEN. ACTION CAN BE ONE OF THE FOLLOWING:

  - o ANALYSIS TO ASSURE LISTED DEVICE NEED NOT FUNCTION FOR MITIGATION OF SAID ACCIDENTS AND WHOSE FAILURE IS DEEMED NOT DETRIMENTAL TO PLANT SAFETY.

- o ANALYSIS TO ASSURE LISTED DEVICE TAKES ITS ACTION PRIOR TO FAILURE AND NO SUBSEQUENT ADVERSE EFFECTS OF ITS FAILURE WILL OCCUR.

- o RELOCATE OR SHIELD LISTED DEVICE TO PROVIDE AN ACCEPTABLE ENVIRONMENT.

- o SPECIFY A CHANGE-OUT PROGRAM.

- o ADDITIONAL TEST/ANALYSIS.

- o TEST.

- o REPLACEMENT PROGRAM.

* CURRENT ACTIVITIES OF DECO IN RESOLVING DEFICIENCIES.

  - o WITH THE EPRI/BWR OWNER'S GROUP IN JOINT QUALIFICATION PROGRAMS ON COMMON ITEMS (ITEMS THAT ARE BEING USED BY 3 OR MORE UTILITIES).

  - o APPROACHED SEVERAL A/E'S AND INDEPENDENT LABORATORIES AS WELL AS OUR OWN RESEARCH LABORATORY TO DISCUSS RESOLUTIONS OF DEFICIENCIES ON PLANT-SPECIFIC ITEMS.

CENTRAL FILE

- o LOCATED AT FERMI 2 SITE

- o CONTAINING ALL SUPPORTING DOCUMENTS

- o ALL PROPRIETARY RECORDS ARE LOCATED AT VENDOR'S FILES. HOWEVER, SUMMARIES OF THESE TEST REPORTS WERE OBTAINED AND KEPT AT THE FERMI 2 CENTRAL FILE.

EDISON POSITION ON ATWS MODIFICATIONS

DETROIT EDISON HAS BEEN FOLLOWING THE ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) ISSUE FOR A NUMBER OF YEARS BECAUSE OF OUR CONCERN FOR SAFETY AND OUR SPECIFIC INTEREST IN THE IMPACT OF ATWS ON THE DESIGN OF FERMI 2. EDISON ENGINEERS HAVE REVIEWED THE ATWS DOCUMENTATION: THE COMPANY HAS SUPPORTED EARLY ATWS STUDIES BY BOTH THE BECHTEL POWER CORPORATION AND KMC, INC.; AND MORE RECENTLY THE COMPANY HAS SUPPORTED THE $3 MILLION STUDY BY GENERAL ELECTRIC IN RESPONSE TO THE FEBRUARY 15, 1979 REQUEST FROM THE NUCLEAR REGULATORY COMMISSION (NRC) FOR A GENERIC EVALUATION OF ALTERNATIVE 3 AS DEFINED IN NUREG-0460.

IT IS OUR VIEW THAT THE BASIC APPROACH OR EMPHASIS TO ATWS SHOULD BE TO IMPROVE THE RELIABILITY OF THE REACTOR SCRAM SYSTEM RATHER THAN TO ASSUME THE SYSTEM IS UNRELIABLE AND THEN TO PROVIDE EXTENSIVE MITIGATION CAPABILITY. THEREFORE, AS DETROIT EDISON'S SOLUTION TO THE ATWS ISSUES WE PLAN TO IMPLEMENT ALTERNATIVE 2 AS DESCRIBED IN VOLUME 3 OF NUREG-0460 PLUS A MANUALLY INITIATED 86 GPM STANDBY LIQUID CONTROL SYSTEM AND TO PROVIDE OPERATOR TRAINING FOR ATWS EVENTS. WE CONTEND THESE CONSTITUTE A SUBSTANTIAL AND SUFFICIENT IMPROVEMENT IN SAFETY.

THE RECIRCULATION PUMP TRIP (RPT) PROVISION OF ALTERNATIVE 2 HAS BEEN PART OF FERMI'S CONCEPTUAL DESIGN FOR A NUMBER OF YEARS AND WILL BE PROVIDED PRIOR TO FUEL LOADING. RPT EFFECTIVELY LIMITS THE PRESSURE TRANSIENT WHICH OCCURS DURING THE EARLY STAGES OF SOME ATWS SCENARIOS. THE INCORPORATION OF RPT WILL ASSURE NO FAILURES OF THE REACTOR COOLANT PRESSURE BOUNDARY DURING ANY ATWS EVENT AND THUS ASSURE THAT CONTAINMENT INTEGRITY WILL BE MAINTAINED WHILE REACTIVITY IS BROUGHT UNDER CONTROL. THE RPT AND ITS ASSOCIATED LOGIC WILL MEET THE REQUIREMENTS SPECIFIED IN NUREG-0460 VOLUME 3. ALSO, PRIOR TO FUEL LOADING, TRAINING WILL BE PROVIDED FOR PROMPT OPERATOR RECOGNITION OF ATWS EVENTS AND THE MEASURES NECESSARY FOR MITIGATION INCLUDING THE USE OF THE STANDBY LIQUID CONTROL SYSTEM.

THE FINAL ITEM TO BE PROVIDED PRIOR TO FUEL LOAD ARE ADDITIONAL AND DIVERSE LEVEL SENSORS IN THE INSTRUMENT VOLUME ATTACHED TO THE SCRAM DISCHARGE VOLUME (SDV). IT SHOULD BE NOTED THAT THE SDV DESIGN FOR FERMI 2 IS THE NEWER, IMPROVED DESIGN, COMPARED TO THAT UTILIZED AT BROWNS FERRY. THE FERMI DESIGN PROVIDES FOR <u>TWO</u> (RATHER THAN ONE) INSTRUMENT VOLUMES THAT ARE CLOSELY COUPLED TO THEIR RESPECTIVE SCRAM DISCHARGE VOLUMES. THIS ELIMINATES THE CHANCE FOR WATER TO BE HELD UP IN THE SDV AND, MORE IMPORTANTLY, MAKES IT IMPOSSIBLE FOR WATER FILLING THE SDV TO GO UNDETECTED. THUS, WHILE THE PRECISE CAUSE FOR THE PARTIAL SCRAM FAILURE AT BROWNS FERRY IS STILL UNKNOWN, IT IS HIGHLY UNLIKELY

THAT IT COULD OCCUR AT FERMI DUE TO THE DIFFICULTY IN PROVIDING A SCENARIO THAT WOULD ALLOW WATER IN THE SDV TO GO UNDETECTED. EVEN SO, MEASURES ARE BEING TAKEN TO IMPROVE THE RELIABILITY OF WATER DETECTION IN THE INSTRUMENT VOLUME AND TO PROVIDE A VENT DESIGN THAT WILL FURTHER REDUCE THE CHANGES OF MALOPERATION OF THE SDV SYSTEM. IT MIGHT ALSO BE NOTED THAT THE CLOSE HYDRAULIC COUPLING DESIGN UTILIZED IN FERMI 2 ALSO PROTECTS IT FROM LOSS OF AIR TO THE CONTROL ROD DRIVE AIR SYSTEM SO THAT THE AIR DUMP SYSTEM PROPOSED IN THE GENERIC SER-"BWR SCRAM DISCHARGE SYSTEM" IS NOT REQUIRED.

IN ADDITION, DETROIT EDISON HAS INITIATED A REQUEST FOR THE ENGINEERING AND HARDWARE NECESSARY TO IMPLEMENT THE ALTERNATE ROD INJECTION (ARI) SYSTEM AND TO ALLOW SIMULTANEOUS USE OF BOTH SLCS PUMPS TO PROVIDE A PUMPING CAPACITY OF 86 GPM. ARI PROVIDES A METHOD INDEPENDENT OF AND DIVERSE FROM THE REACTOR PROTECTION SYSTEM OF DUMPING AIR FROM THE SCRAM AIR HEADER THEREBY PROVIDING BACKUP TO THE ELECTRICAL PORTION OF THE SCRAM SYSTEM.

THE ATWS PACKAGE DESCRIBED ABOVE IS WELL UNDERSTOOD, IS COST EFFECTIVE AND SHOULD EASILY REDUCE THE ATWS RISK BY A RANGE OF SEVERAL FACTORS TO AN ORDER OF MAGNITUDE OR MORE (G.E. PREDICTS A FACTOR OF 100 RISK REDUCTION FOR ALTERNATIVE 2).

THIS ASSERTION IS BASED ON A DESIGN GOAL FOR RPT AND ARI THAT WILL EASILY REDUCE THE ELECTRICAL FAILURE POTENTIAL BY AN ORDER OF MAGNITUDE AND THE REALIZATION THAT THE REMAINING MECHANICAL FAILURE COMPONENT WOULD LIKELY RESULT IN PARTIAL SCRAM FAILURES THAT CAN BE ADEQUATELY HANDLED BY WELL THOUGH OUT OPERATING PROCEDURES COUPLED WITH THE LIQUID BORON INJECTION SYSTEM.

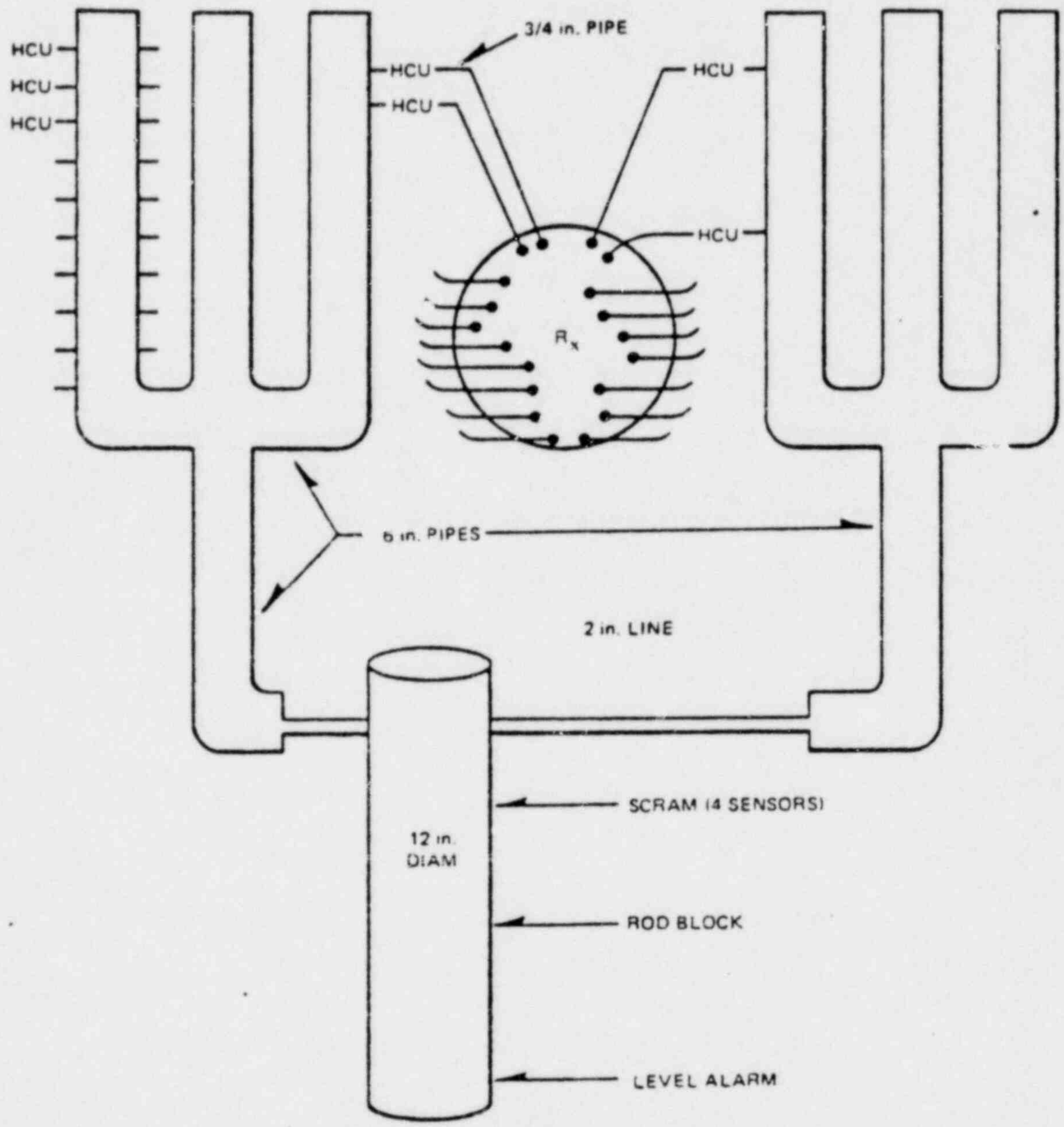
IT SHOULD BE NOTED THAT DETROIT EDISON IS CONCERNED WITH THE PROPOSED FIXED DESCRIBED IN NUREG-0460 THAT GO MUCH BEYOND THE ABOVE MENTIONED APPROACH. WE HAVE CONCERNS WITH THE EXTENSIVE INTERACTION WITH OTHER PLANT SYSTEMS IMPOSED BY YET TO BE DESIGNED ALTERNATIVES 3 AND 4 (NUREG-0460), AND THE INCREASED POTENTIAL OF INADVERTENT BORON INJECTION ASSOCIATED WITH AUTO-BORON. THE INCREMENTAL BENEFIT THESE MITIGATION-ORIENTED ALTERNATIVES PROVIDE BEYOND THE IMPROVEMENT ACHIEVED BY THE MEASURES WE ARE TAKING AT FERMI APPEAR TO US TO BE DISPROPORTIONATELY SMALL TO THE PROBLEMS THEY CREATE. FOR THESE REASONS, NO FURTHER ATWS MODIFICATIONS ARE BEING PLANNED FOR FERMI 2.

SCRAM DISCHARGE PIPE BREAK

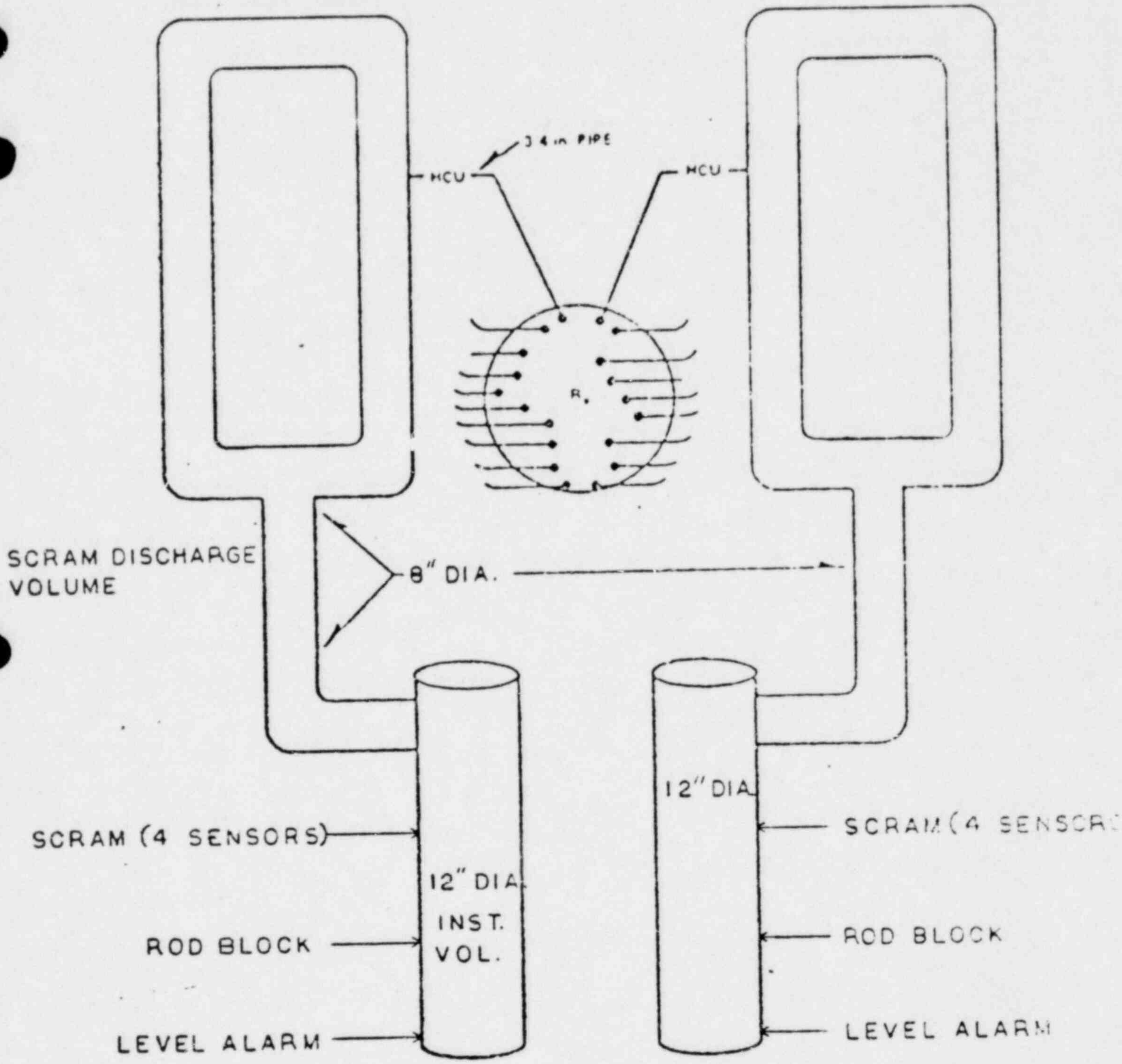
THE NRC HAS ISSUED THE OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA (AEOD) REPORT ENTITLED, "SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM" AS

DRAFT NUREG-0785. AN NRC REQUEST FOR A GENERIC AND PLANT SPECIFIC RESPONSE IN 45 AND 120 DAYS, RESPECTIVELY, WAS ATTACHED TO THE REPORT. DETROIT EDISON HAS FILED THE GENERIC RESPONSE AND REFERENCED THE GENERAL ELECTRIC REPORT NEDO 24342.

A GENERIC SAFETY EVALUATION REPORT (SER) FOR THE SCRAM DISCHARGE VOLUME PIPE BREAK IS SCHEDULED TO BE ISSUED IN LATE JULY BY THE NRC. DETROIT EDISON WILL REVIEW THE FERMI 2 DESIGN FOR COMPLIANCE WITH THE SER CRITERIA. WE EXPECT THAT THE SER WILL BE ISSUED AND OUR REVIEW COMPLETED FOR THE FERMI 2 PLANT SPECIFIC 120 DAY RESPONSE REQUIRED IN THE LETTER FROM ROBERT TEDESCO DIVISION OF LICENSING.



Scram Discharge Volume; Early BWR 4



FIRMI 2 SCRAM DISCHARGE VOLUME

DETROIT EDISON'S PROPOSED ATWS FIX IN COMMENTS TO NUREG-0460

RPT

OPERATOR RESPONSE TRAINING

SDV MODIFICATIONS

86 GPM SLCS (MANUAL)

ARI

FERMI 2 ATWS COMMITMENTS

RPT

OPERATOR RESPONSE TRAINING

SDV MODIFICATIONS

HYDROGEN CONTROL

A COMPOSITE OF SYSTEMS IS USED TO DETECT AND CONTROL THE OXYGEN AND HYDROGEN CONCENTRATIONS IN PRIMARY CONTAINMENT. FOLLOWING A LOCA, HYDROGEN GAS COULD BE GENERATED AS A RESULT OF THE METAL-WATER REACTION AND BOTH HYDROGEN AND OXYGEN WOULD BE GENERATED AS A RESULT OF RADIOLYTIC DECOMPOSITION OF RECIRCULATING COOLANT. THE CORROSION OF CONTAINMENT MATERIALS WAS EVALUATED AND DOES NOT REPRESENT A SIGNIFICANT HYDROGEN SOURCE. THE CONTROL OF COMBUSTIBLE GAS MIXTURES IN THE PRIMARY CONTAINMENT IS ASSURED BY THE FOLLOWING PLANT SYSTEMS:

HYDROGEN/OXYGEN SAMPLING

THERMAL HYDROGEN RECOMBINERS

INERTING SYSTEM

PURGE SYSTEM

HYDROGEN/OXYGEN SAMPLING

THE HYDROGEN AND OXYGEN CONCENTRATIONS ARE CONTINUOUSLY MONITORED AND DISPLAYED IN THE CONTROL ROOM. THERE ARE

TWO SEPARATE SAMPLING SYSTEMS, EACH IS REDUNDANT AND INDEPENDENT. THE SAMPLING SYSTEMS ARE SEISMICALLY AND ENVIRONMENTALLY QUALIFIED ENGINEERED SAFETY FEATURE SYSTEMS. TO ASSURE REPRESENTATIVE SAMPLING, MULTIPLE PORTS ALLOW GAS TO BE DRAWN INTO THE MONITORING SYSTEM FROM SEVERAL LOCATIONS IN THE CONTAINMENT. AN ALARM INITIATES WHEN THE OXYGEN CONCENTRATION REACHES A PRESENT ALARM TO SIGNAL THE OPERATOR IS TO START THE CGCS SYSTEM.

THE PAST-LOCA ON-LINE SAMPLING SYSTEM SERVES AS A BACKUP FOR HYDROGEN/OXYGEN MONITORING.

THERMAL RECOMBINERS

REDUNDANT AND INDEPENDENT THERMAL RECOMBINERS ARE INSTALLED AT FERMI 2 TO ENSURE THAT A COMBUSTIBLE GAS MIXTURE DOES NOT BUILD UP AND IMPAIR THE CONTAINMENT INTEGRITY. EACH RECOMBINER IS INDIVIDUALLY CAPABLE OF LIMITING THE AMOUNT OF OXYGEN IN THE CONTAINMENT ATMOSPHERE TO LESS THAN THE COMBUSTIBLE CONCENTRATION IN CONFORMANCE TO REGULATORY GUIDE 1.7. THESE COMBUSTIBLE GAS CONTROL SYSTEMS (CGCS) CONFORM TO 10 CFR 100 SECTION 50.44 GENERIC DESIGN CRITRIA 41 AND BRANCH TECHNICAL POSITION CSB 6-2.

THE CGCS IS AN ENGINEERED SAFETY FEATURE SYSTEM AND IS SEISMICALLY AND ENVIRONMENTALLY QUALIFIED. THE RECOMBINERS ARE LOCATED IN THE REACTOR BUILDING OUTSIDE OF PRIMARY CONTAINMENT. THE SYSTEM OPERATES TO CONTROL THE MINORITY CONSTITUENT, OXYGEN. THE PROCESS FLOW IS 150 SCFM WITH AN INLET FLOW OF 60 SCFM FOR CONTAINMENT ATMOSPHERE CONTAINING FIVE VOLUME PERCENT OXYGEN.

THE RECOMBINATION TAKES PLACE IN THE REACTION CHAMBER AS A RESULT OF AN EXOTHERMIC REACTION. A BLOWER DRAWS THE CONTAINMENT ATMOSPHERE FROM EITHER THE DRYWELL OR TORUS THROUGH DEDICATED PENETRATIONS. AFTER THE REACTION, THE RESULTANT STEAM IS COOLED AND CONDENSED WITH THE RESULTING WATER AND ANY REMAINING GAS RETURNED TO THE TORUS.

THIS RECOMBINER SYSTEM IS AN INTEGRAL PACKAGE PRODUCED BY ATOMICS INTERNATIONAL. EACH SYSTEM INCLUDES THE SKID MOUNTED HYDROGEN RECOMBINER, A LOCAL POWER CABINET AND A REMOTE CONTROL CABINET IN THE RELAY ROOM. ALL EQUIPMENT NECESSARY TO START A COMBUSTIBLE GAS CONTROL SYSTEM IS LOCATED ON THE MAIN CONTROL ROOM PANEL.

NITROGEN INERTING SYSTEM

THE FUNCTION OF THE NITROGEN INERTING SYSTEM (NIS) IS TO PROVIDE AND MAINTAIN A NITROGEN ATMOSPHERE INSIDE THE PRIMARY

CONTAINMENT, AND TO PROVIDE PRESSURIZED NITROGEN FOR PNEUMATIC SERVICE INSIDE THE PRIMARY CONTAINMENT AND DISTRIBUTION THROUGHOUT THE PLANT. THE PRIMARY CONTAINMENT WILL BE INERTED AND CONTROLLED TO LESS THAN 4% OXYGEN. EVEN IF LARGE QUANTITIES OF HYDROGEN ARE GENERATED FOLLOWING A LOCA THE INERTED CONTAINMENT WILL HAVE INSUFFICIENT OXYGEN TO SUPPORT COMBUSTION OF HYDROGEN.

THE INERTING SYSTEM IS NOT SAFETY-RELATED SYSTEM AND IS NOT DESIGNED TO MEET SEISMIC AND OTHER CRITERIA EXCEPT WHERE PENETRATION AND ISOLATION IS CONCERNED. THE CLOSURE OF CONTAINMENT ISOLATION VALVES AND OTHER SELECTED FEED VALVES IN THE NIS WILL OCCUR FOR LOW REACTOR WATER LEVEL-2, HIGH DRYWELL PRESSURE OR HIGH RADIATION IN THE REACTOR BUILDING EXHAUST. THE ISOLATION SIGNALS AND VALVE ACTUATORS CONFORM TO THE REQUIREMENTS OF BRANCH TECHNICAL POSITION CSB 6-4.

THE INERTING SYSTEM IS COMPOSED OF LARGE LINES TO THE TORUS AND DRYWELL VALVES SIZED 20 AND 24 INCHES, RESPECTIVELY. THIS SYSTEM IS HIGH FLOW AT LOW PRESSURE AS PROVIDED THROUGH THE STEAM VAPORIZER FED FROM A NITROGEN STORAGE TANK. THE INERT SYSTEM IS USED INFREQUENTLY WITH ISOLATION VALVE OPERATION LIMITED TO 90 HOURS PER YEAR FOR CONDITIONS OTHER THAN COLD SHUTDOWN OR REFUELING PER CSB 6-4.

A PRESSURE CONTROLLED NITROGEN MAKE-UP SYSTEM IS PROVIDED AT FERMI 2 FOR ON-LINE MAKE-UP OF NITROGEN DUE TO LEAKAGE. THE DRYWELL IS KEPT SLIGHTLY POSITIVE RELATIVE TO SECONDARY CONTAINMENT. THE MAKE-UP SYSTEM IS COMPRISED OF 1-1/2 INCH LINES TO THE DRYWELL AND TORUS FROM THE PRESSURIZED NITROGEN SYSTEM. THE MAKE-UP SYSTEM IS LOW FLOW AT HIGH PRESSURE AND IS FED THROUGH AN ELECTRIC HEATER FED FROM THE NITROGEN STORAGE TANK. THE ISOLATION VALVES IN THIS SYSTEM COMPLY WITH CSB 6-4.

PURGE SYSTEM

CONTAINMENT PURGE CAPABILITY IS PROVIDED TO EVACUATE THE CONTAINMENT ATMOSPHERE AND TO FUNCTION AS BACK-UP HYDROGEN CONTROL. THE PURGE SYSTEM IS NOT A SAFETY-RELATED SYSTEM AND IS NOT SEISMICALLY QUALIFIED WITH THE EXCEPTION OF THE PENETRATIONS AND ISOLATION VALVES.

THE DRYWELL PURGE INLET AND VENT OUTLET LINES ARE 20 INCHES IN DIAMETER WHILE THE TORUS INLET AND OUTLET VALVES ARE 24 INCH IN DIAMETER. THESE LINES ARE USED TO EVACUATE THE CONTAINMENT ATMOSPHERE DURING THE PURGE (DE-INERT) OPERATION. THE PURGE SYSTEM CONNECTS TO EITHER THE REACTOR BUILDING EXHAUST OF THE STANDBY GAS TREATMENT SYSTEM. THE USE OF THESE ISOLATION VALVES IN THE PURGE OPERATION IS LIMITED TO 90 HOURS PER YEAR DURING CONDITIONS OTHER THAN COLD SHUT-DOWN OR REFUELING PER CSB 6-4.

AN ON-LINE PURGE SYSTEM IS INCLUDED AT FERMI 2. THIS IS A 1-1/2 INCH DIAMETER SYSTEM USED TO VENT CONTAINMENT ATMOSPHERE TO MAINTAIN PRESSURE CONTROL RELATIVE TO THE SECONDARY CONTAINMENT.

BOTH THE LARGE PURGE SYSTEM AND SMALL ON-LINE SYSTEM COMPLY WITH CSB 6-4. THIS INCLUDES THE FIVE-SECOND VALVE CLOSURE, DEBRIS SCREENS, 90-HOUR LIMIT AND DIVERSE ISOLATION (I.E., LOW REACTOR WATER LEVEL-2, HIGH DRYWELL PRESSURE AND HIGH RADIATION IN THE REACTOR BUILDING EXHAUST).

SUMMARY

THE FERMI 2 DESIGN HAS INCORPORATED SEVERAL HYDROGEN CONTROL MEASURES. THE LIMITING OF A COMBUSTIBLE GAS MIXTURE WITH AN INERTED CONTAINMENT AND POST LOCA HYDROGEN CONTROL WITH DEDICATED THERMAL RECOMBINERS ARE THE KEY FEATURES.

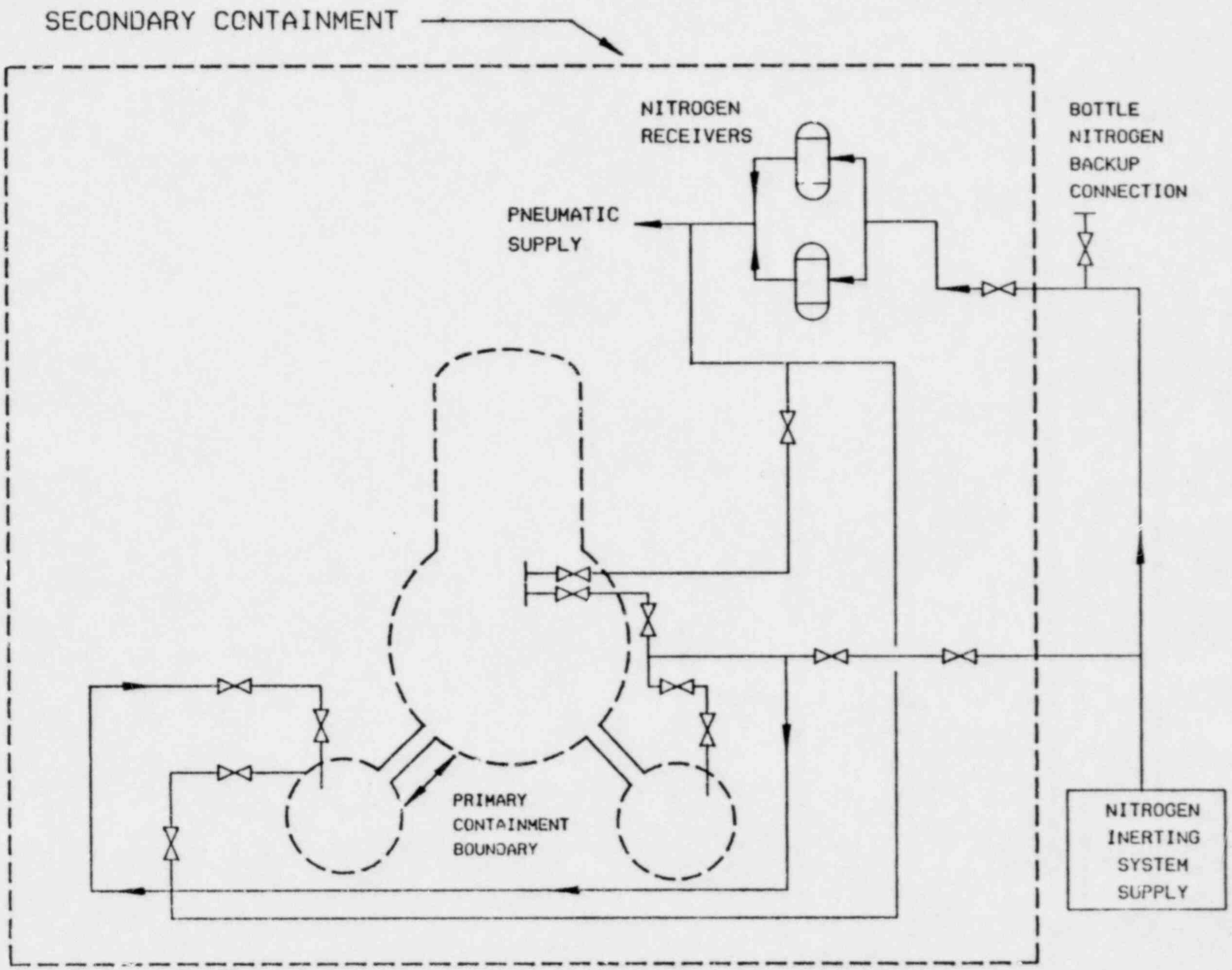
HYDROGEN CONTROL

NITROGEN INERTING SYSTEM

HYDROGEN/OXYGEN MONITORING

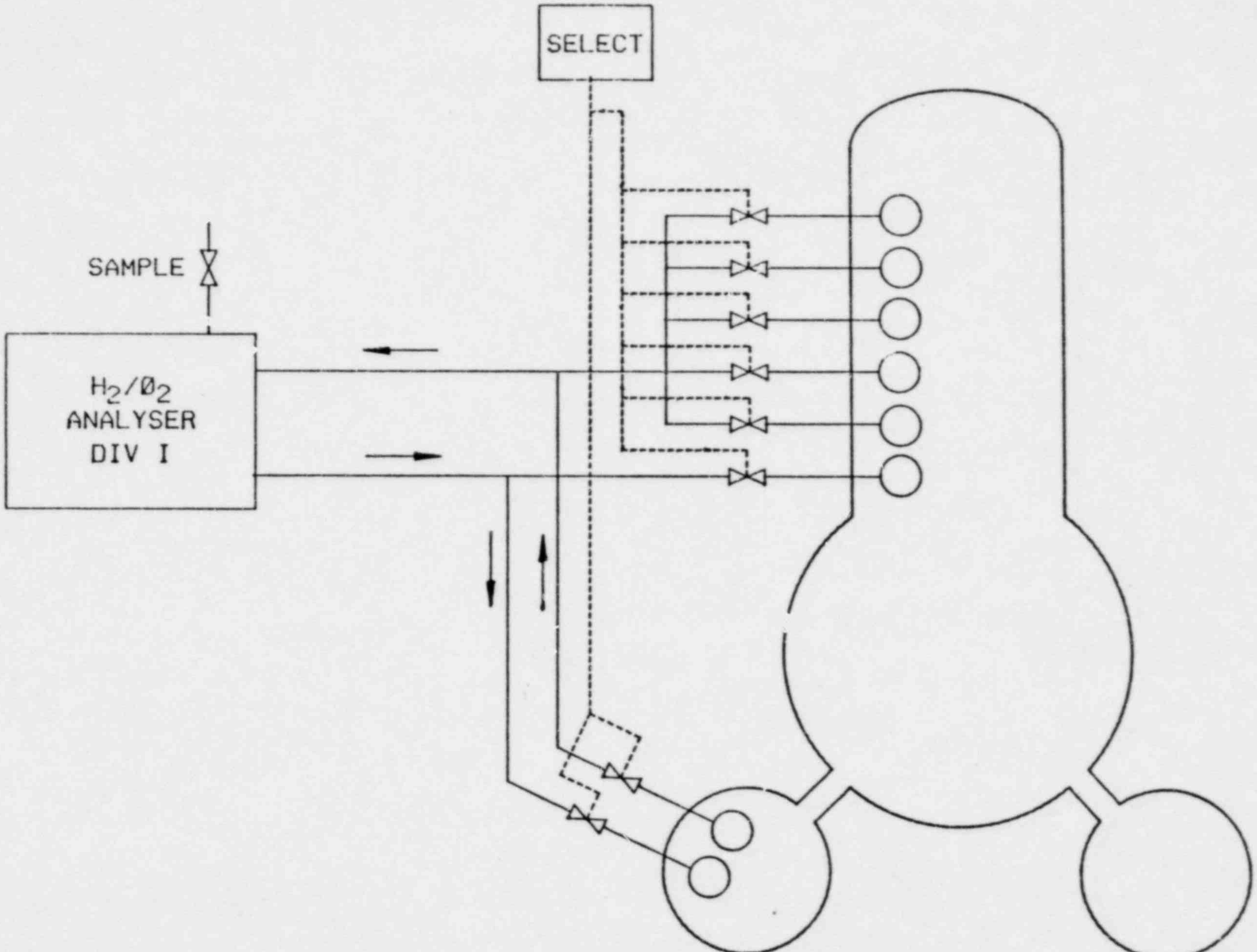
THERMAL RECOMBINERS

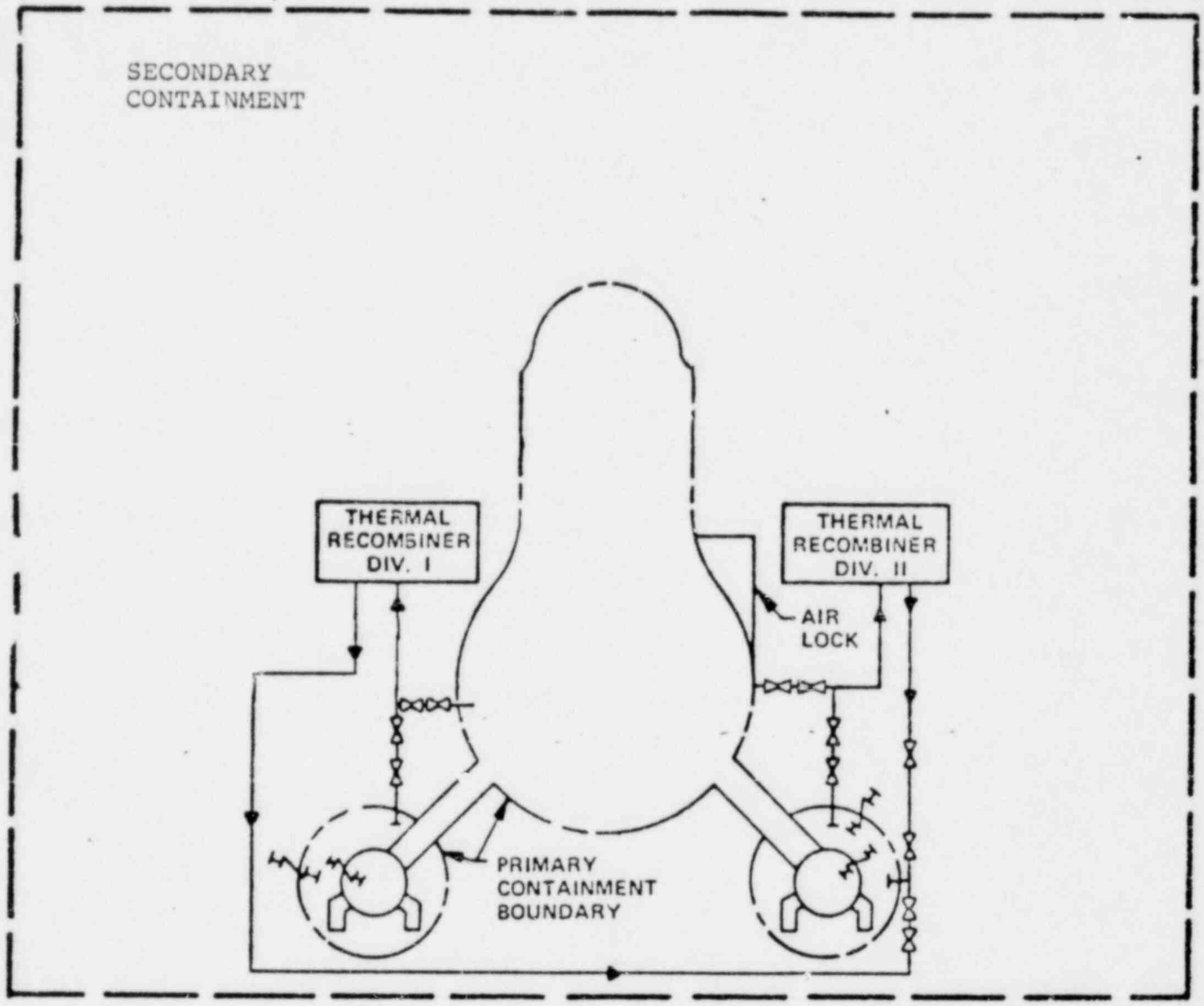
PURGE SYSTEM



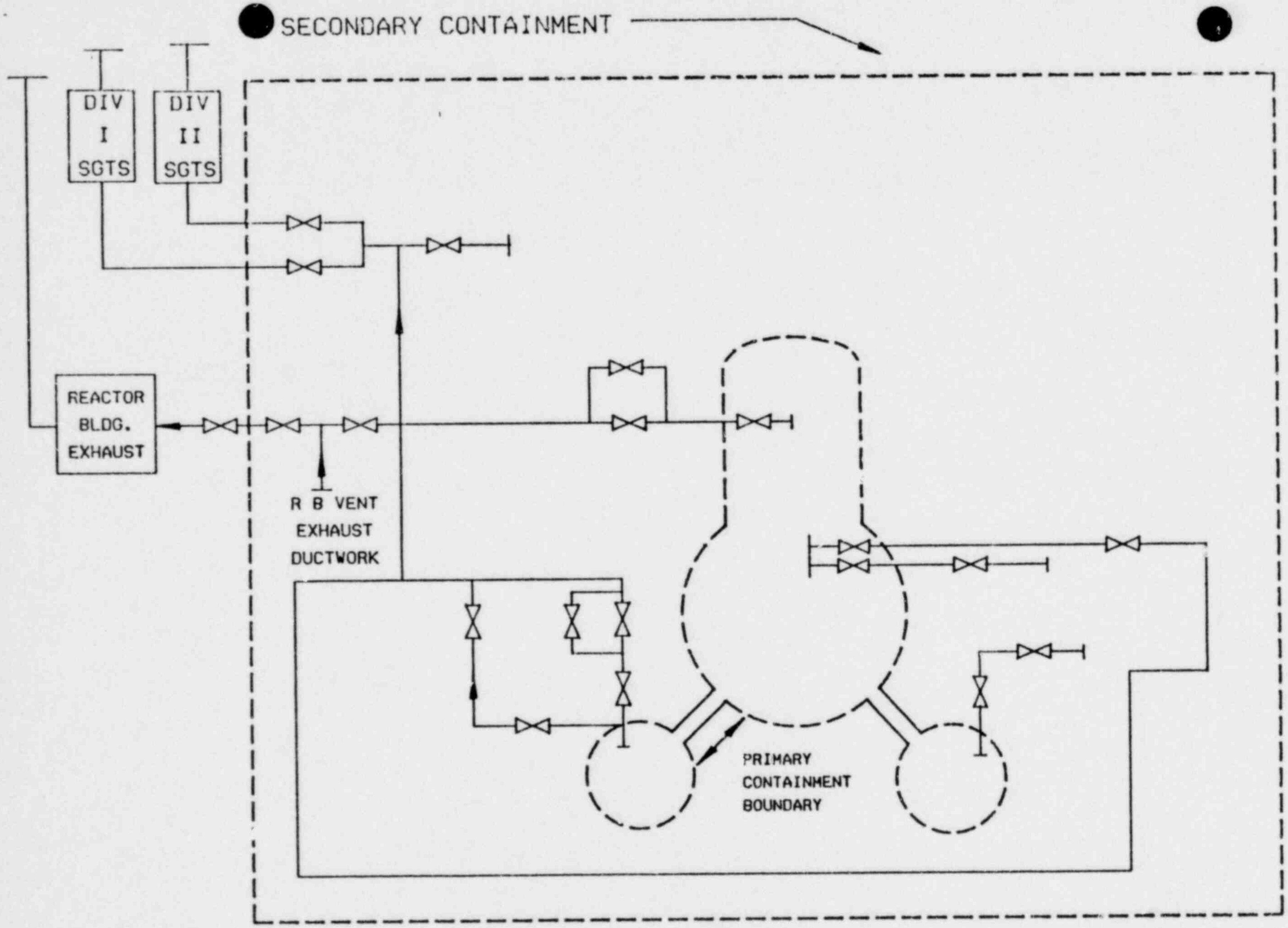
FERMI 2 NITROGEN INERTING SYSTEM

HYDROGEN/OXYGEN MONITORING





FERMI 2 THERMAL RECOMBINERS



FERMI 2 PURGE SYSTEM

TMI ISSUES

\- STATUS -

(BALANCE OF SER TMI OPEN ITEM NOT DISCUSSED ELSEWHERE ON ACRS AGENDA)

NRC SER OPEN ITEM NO.	SUBJECT & COMMENTS
16A	I.C.1 GUIDELINES FOR DEVELOPING EMERGENCY PROCEDURES I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES o BWR OWNERS GROUP EMERGENCY PROCEDURES GUIDELINES UNDER NRC REVIEW o WALK-THROUGH OF FERMI 2 EMERGENCY PROCEDURES SCHEDULED FOR JULY 25-26 AT THE BROWNS FERRY SIMULATOR
16C	I.G.1 TRAINING DURING LOW POWER TESTING o NRC REQUIRES DETROIT EDISON TO CONDUCT A SIMULATED LOSS OF OFFSITE AND ONSITE POWER TEST o DETROIT EDISON HAS COMMITTED TO PERFORM SUCH A TEST USING THE GUIDELINES PREPARED BY THE BWR OWNERS GROUP o DETROIT EDISON HAS COMMITTED TO SUBMIT THE DETAILED TEST PROCEDURE AND SAFETY ANALYSIS AT LEAST 90 DAYS BEFORE FUEL LOADING
16E	II.D.1 TESTING SAFETY RELIEF VALVES o DETROIT EDISON IS PARTICIPATING IN THE BWR OWNERS GROUP PROGRAM AND IS REVIEWING THE PROGRAM DESCRIPTION AND SCOPE TO ENSURE THAT THE RESULTS WILL BE APPLICABLE TO THE FERMI 2 PLANT-SPECIFIC VALVES & PIPING o DETROIT EDISON HAS COMMITTED TO PROVIDE NRC-REQUIRED DOCUMENTATION ON A SCHEDULE AGREED TO BY THE BWR OWNERS GROUP AND THE NRC STAFF

NRC SER OPEN ITEM NO.	SUBJECT & COMMENTS
16F	II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY o FERMI 2 COMPLIES WITH OR HAS COMMITTED TO COMPLY WITH THE NRC REQUIREMENTS UNDER ITEM II.E.4.2 o NRC PLANS TO CONDUCT AN AUDIT TO CONFIRM COMPLIANCE WITH THE REQUIREMENTS GIVEN IN NRC BRANCH TECHNICAL POSITION CSB 6-4 CONCERNING PURGE VALVE OPERABILITY

EMERGENCY PLANNING

1. EMERGENCY SUPPORT FACILITIES

THE CONTROL ROOM (CR), TECHNICAL SUPPORT CENTER (TSC), OPERATIONAL SUPPORT CENTER, (OSC), AND EMERGENCY OPERATIONS FACILITY (EOF) ARE THE FOUR FACILITIES THAT WILL BE USED TO RESPOND TO AN EMERGENCY AT FERMI 2. FIGURE 1 IS A PLOT PLAN OF THE SITE SHOWING THE LOCATIONS OF EACH FACILITY. THE EMERGENCY SUPPORT FACILITIES WHICH ADDRESS THE REQUIREMENTS OF NUREG-0737 AND 0696 ARE DESCRIBED IN APPENDIX H TO THE FSAR.

1.1 CONTROL ROOM

THE CONTROL ROOM (CR) PROVIDES A CENTRALIZED LOCATION FOR DAY-TO-DAY PLANT OPERATIONS. IN THE EVENT OF AN EMERGENCY, THE CR PROVIDES THE INITIAL ONSITE CENTER OF EMERGENCY CONTROL. CR PERSONNEL EVALUATE AND EFFECT CONTROL OVER THE INITIAL ASPECTS OF AN EMERGENCY AND INITIATE ACTIONS NECESSARY FOR COPING WITH THE INITIAL PHASES OF AN EMERGENCY UNTIL THE SUPPORT CENTERS ARE ACTIVATED.

- o CONTINUOUS EVALUATION OF THE MAGNITUDE AND POTENTIAL CONSEQUENCES OF AN INCIDENT

- o INITIAL CORRECTIVE ACTIONS

- o NOTIFICATION OF APPROPRIATE INDIVIDUALS, AND GOVERNMENT AND EMERGENCY SUPPORT AGENCIES

1.2 TECHNICAL SUPPORT CENTER

THE TSC IS A 5000 SQUARE FOOT HARDENED FACILITY ON THE GROUND FLOOR OF A TWO-STORY OFFICE SERVICE BUILDING LOCATED WITHIN THE SECURITY PERIMETER OF THE FERMI 2 PLANT. FUNCTIONALLY, THE TSC PROVIDES INFORMATION ON PLANT STATUS FOR USE BY TECHNICAL AND MANAGEMENT PERSONNEL IN SUPPORT OF THE COMMAND AND CONTROL FUNCTIONS, CARRIED OUT IN THE CONTROL ROOM. THE TSC IS AN EASY 4 MINUTE INSIDE WALK FROM THE CONTROL ROOM THROUGH THE TURBINE BUILDING AS SHOWN ON FIGURE 2.

UNDER EMERGENCY CONDITIONS, THE TSC FUNCTIONS AS THE PRIMARY INFORMATION/COMMUNCIATIONS SOURCE TO THE NRC, THE OSC, AND THE EOF AS INDICATED IN FIGURE 3. ADDITIONALALLY, THE FUNCTIONS OF THE EOF ARE PERFORMED IN THE TSC UNTIL SUCH TIME AS THE EOF IS ACTIVATED.

THE TSC IS DESIGNED TO ACCOMODATE 25 PERSONS REPRESENTING DETROIT EDISON AND THE NRC (FIGURE 4). THE OPEN DESIGN WITH MOVABLE PORTITIONS AND RAISED COMPUTER FLOOR PROVIDES MAXIMUM FLEXIBILITY FOR PERSONNEL AND CRT POSITIONING.

THE HEART OF THE TSC IS A COMPUTER-BASED SYSTEM WITH CRTS AND PRINTERS (FIGURE 5). PART OF THE PERMANENT INFORMATION REQUIREMENTS ARE PROVIDED BY CLOSED CIRCUIT TV (CCTV) POSITIONED AS SHOWN IN FIGURE 6 ON EXISTING CONTROL ROOM DISPLAYS. THE CCTV ALSO PROVIDES THE INTERIM DATA INFORMATION SOURCE UNTIL THE COMPUTER-BASED SYSTEM IS AVAILABLE.

THE SAFETY PARAMETER DISPLAY SYSTEM (SPDS) IS AN IMPORTANT FUNCTION ON THE COMPUTER-BASED SYSTEM AND INCLUDES THE FEATURES SHOWN IN FIGURE 7.

1.3 OPERATIONAL SUPPORT CENTER (OSC)

THE OSC (FIGURE 2) IS A DESIGNATED AREA AT THE NORTH END OF THE THIRD FLOOR OF THE TURBINE BUILDING AND PROVIDES AN ASSEMBLY POINT FOR SHIFT SUPPORT PERSONNEL FOR ASSIGNMENT OF DUTIES IN SUPPORT OF EMERGENCY OPERATIONS. PERSONNEL SUCH AS INSTRUMENT TECHNICIANS, ENGINEERS, MECHANICS, ELECTRICIANS, RADIATION/HEALTH PHYSICS TECHNICANS, EQUIPMENT OPERATORS, ETC., ARE DISPATCHED FROM THIS AREA.

1.4 EMERGENCY OPERATIONS FACILITY (EOF)

THE EOF WILL BE A COMMAND POST FOR THE OVERALL MANAGEMENT OF THE EMERGENCY RESPONSE WITH OFFSITE ORGNAIZATIONS, THE

COORDINATION OF RADIOLOGICAL AND ENVIRONMENTAL ASSESSMENTS, THE DETERMINATION OF RECOMMENDED PROTECTIVE ACTIONS FOR THE PUBLIC, AND MANAGEMENT OF RECOVERY OPERATION (FIGURE 8). THE EOF, (FIGURE 9), DESIGNED TO HANDLE 40 PERSONS, IS IN THE BASEMENT OF THE NUCLEAR OPERATIONS CENTER, AND WILL BE LOCATED APPROXIMATELY THREE-FOURTHS (3/4) OF A MILE SOUTHWEST OF THE PLANT ON OWNER-CONTROLLED PROPERTY.

THE EOF WILL FUNCTION TO PROVIDE ASSISTANCE IN THE DECISION MAKING PROCESS TO PROTECT THE PUBLIC HEALTH AND SAFETY AND TO CONTROL RADIOLOGICAL EMERGENCY MONITORING TEAMS AND FACILITIES ONSITE AND OFFSITE. RADIOLOGICAL AND METEOROLOGICAL DATA AND ADEQUATE PLANT SYSTEM INFORMATION WILL BE PROVIDED TO PERFORM THESE FUNCTIONS. THE EOF IS NORMALLY THE FOCAL POINT FOR THE RECEIPT AND ANALYSIS OF ALL FIELD MONITORING DATA AND THE COORDINATION OF SAMPLE MEDIA.

1.5 ACTIVATION OF EMERGENCY RESPONSE FACILITIES

ACTIVATION OF THE TSC AND OSC IS INITIATED AT THE ALERT LEVEL;THE EOF IS ACTIVATED FOR THE SITE AREA AND GENERAL EMERGENCY CLASSIFICATION. THE FUNCTIONAL STAFFING OF THESE CENTERS IS SHOWN IN FIGURE 10.

THE CONTROL ROOM TSC, AND EOF ARE THE PRIMARY COMMAND AND CONTROL CENTERS, WITH OFFSITE EDISON FACILITIES AS SECONDARY FUNCTIONS. A SCHEMATIC OF THE PRIMARY COMMUNCIATIONS AMONG THE VARIOUS OFFSITE EDISON FACILITIES AND OTHER RESPONSE ORGANIZATIONS IS SHOWN IN FIGURE 11.

2. ROLE OF FEMA, STATE AND LOCAL AGENCIES

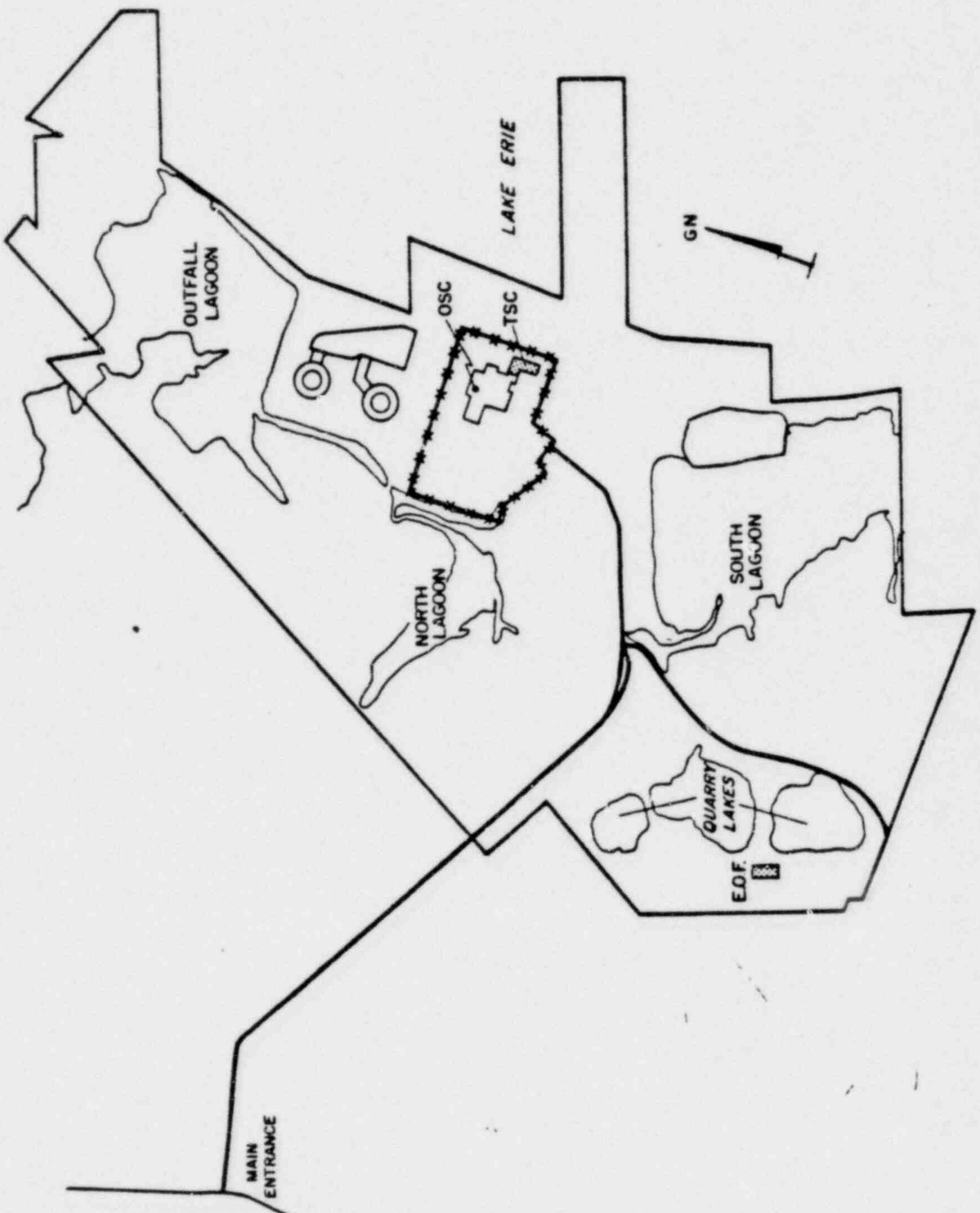
THE RADIOLOGICAL EMERGENCY RESPONSE PLAN (RERP) FOR FERMI 2 WAS UPGRADED TO MEET THE REQUIREMENTS OF 10 CFR 50 AND NUREG-0654/ FEMA-REP-1 AND FILED WITH THE NRC ON MARCH 31, 1981. NRC COMMENTS ON THE PLAN WERE REVIEWED WITH THE STAFF ON JUNE 30 AND RESPONSES ARE DUE FROM EDISON AUGUST 30. A REVISED RERP AND PROCEDURES WILL BE SUBMITTED TO THE NRC JUST PRIOR TO THE FULL-SCALE EXERCISE IN FEBRUARY 1982.

THE FERMI 2 RERP INCLUDES A PLUME EXPOSURE PATHWAY EMERGENCY PLANNING ZONE (EPZ) EXTENDING TO ABOUT 10 MILES AND AN INGESTION PATHWAY EPZ TO 50 MILES. THE EPZ FOR THE PLUME EXPOSURE INCLUES ALL AREAS WITHIN 10 MILES THAT LIE IN MONROE COUNTY AND A SMALL PORTION OF THE SOUTHERN TIP OF WAYNE COUNTY, MICHIGAN (FIGURE 12). THE 50-MILE EPZ INCLUDES PORTIONS OF MICHIGAN, OHIO, AND THE PROVINCE OF ONTARIO, CANADA (FIGURE 13).

THE MICHIGAN STATE PLAN, PREPARED BY THE EMERGENCY SERVICES DIVISION OF THE STATE POLICE, WAS SUBMITTED TO FEMA, REGION V FOR APPROVAL IN FEBRUARY 1981. FEMA WILL BE SUBMITTING ITS FINDINGS ON THE MICHIGAN PLAN FOR THE OPERATING PLANTS TO FEMA HEADQUARTERS IN JULY 1981. ONCE MICHIGAN IS COMPLETED, FEMA INTENDS TO PROCEED WITH THE OHIO STATE PLAN. AT THE PRESENT TIME, SINCE THE PROVINCE OF ONTARIO CANADA IS IN THE 50-MILE EPZ, NOTIFICATION OF ANY EMERGENCIES AT FERMI 2 WILL BE MADE BY THE STATE POLICE EMERGENCY SERVICES DIRECTOR.

THE FERMI 2 FULL-SCALE EXERCISE AND NRC APPRAIISAL PROGRAM ARE SCHEDULED FOR THE FIRST TWO WEEKS IN FEBRUARY 1982, SUBJECT TO APPROVAL BY ALL PARTIES INVOLVED (FIGURE 14). THE MICHIGAN STATE POLICE ARE WORKING TOWARD THIS TARGET DATE AND ARE SCHEDULED TO SUBMIT THE PLANS FOR MONROE COUNTY AND JURISDICATIONS WITHIN WAYNE COUNTY IN NOVEMBER 1981. FEMA HAS BEEN CONTACTED AND IS PRESENTLY REVIEWING THIS SCHEDULE.

FOLLOWING THE FULL-SCALE EXERCISE AND APPRAISSAL, A SCHEDULE WILL BE DETERMINED FOR IMPLEMENTATION OF ANY UNRESOLVED ISSUES IDENTIFIED BY NRC OR FEMA BY FUEL LOAD OR FULL POWER OPERATION.



PLOT PLAN

FIGURE 1

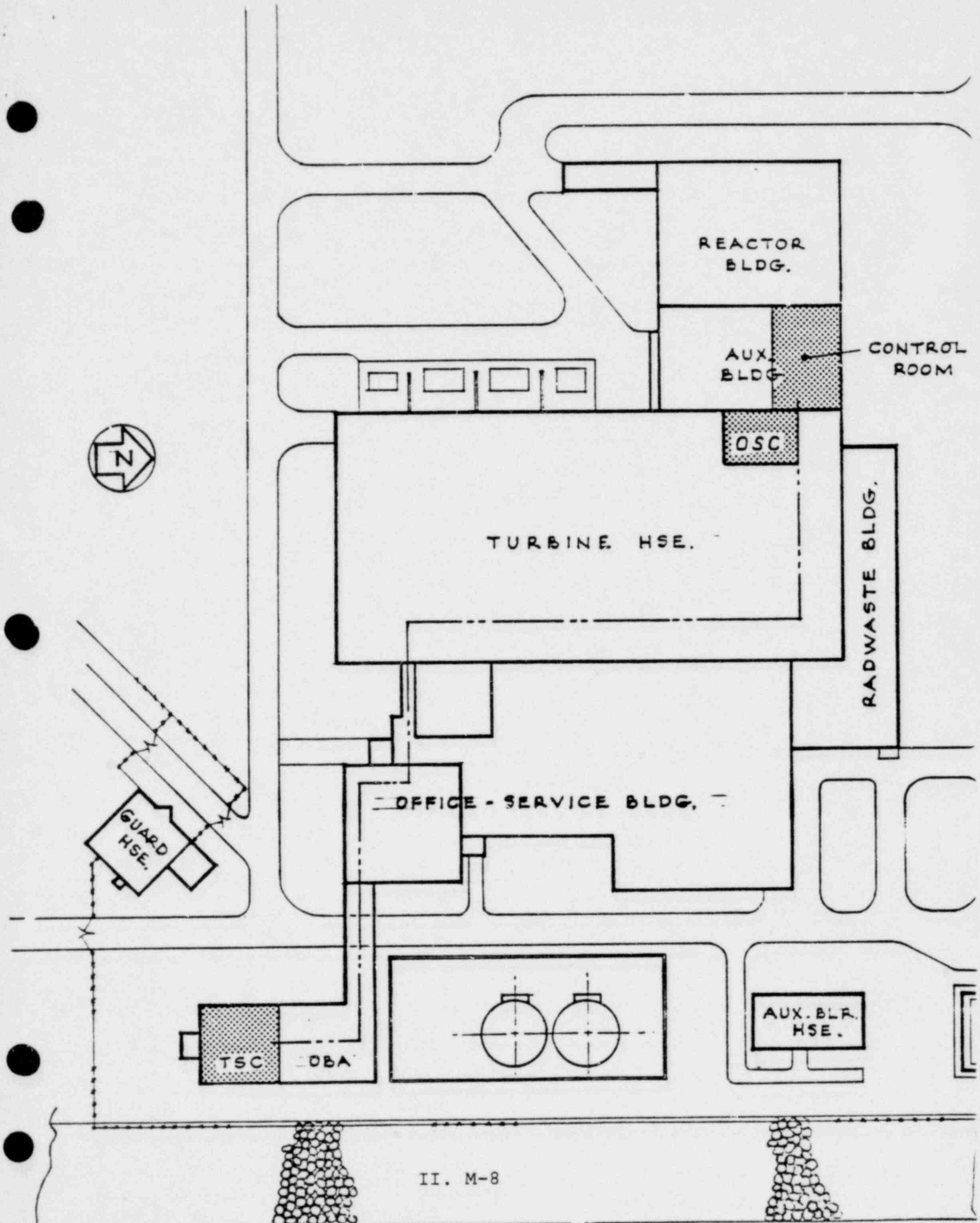
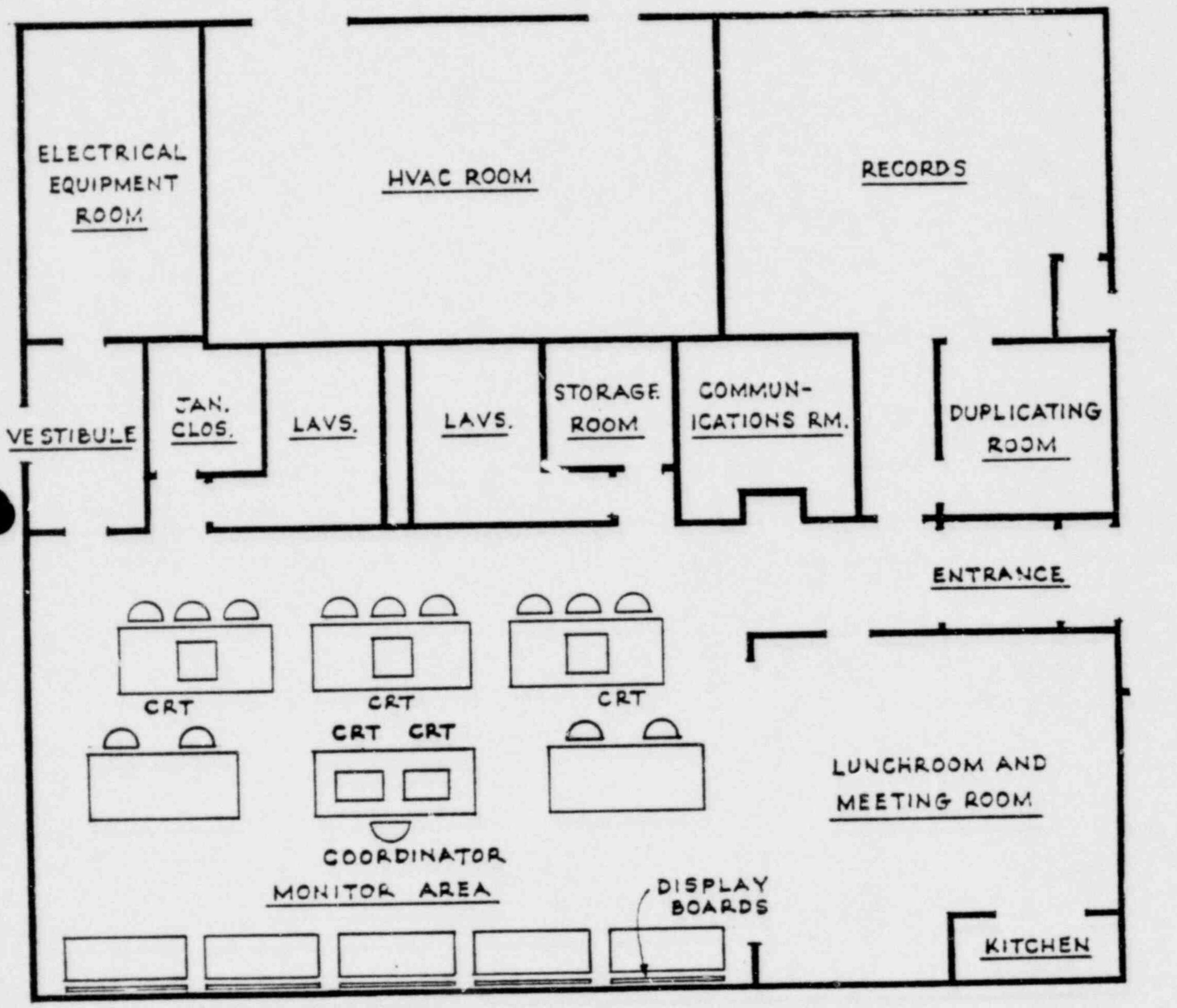

REACTOR BLDG.
AUX. BLDG
CONTROL ROOM
OSC
N
TURBINE HSE.
RADWASTE BLDG.
GUARD HSE.
OFFICE - SERVICE BLDG.
TSC
DBA
AUX. BLR. HSE.

FIGURE 2

FIGURE 3

TECHNICAL SUPPORT CENTER FUNCTIONS

- PLANT STATUS AND DYNAMICS PRIOR TO AND DURING THE ACCIDENT
- PERFORMANCE OF ACCIDENT MITIGATION FUNCTIONS
- NATURE AND TREND OF THE ACCIDENT
- DAMAGE TO THE PLANT AND EQUIPMENT
- STATUS OF THE OPERATION (INCLUDING PERSONNEL ACTIVITY IN THE PLANT)
- AMOUNT OF RADIOLOGICAL RELEASE TO THE ENVIRONMENT
- PREVAILING METEOROLOGICAL STATUS
- PROJECTED ENVIRONS RADIATION AND RADIOACTIVITY LEVEL
- RADIOLOGICAL ASSESSMENT (IMPACT OF RADIOACTIVITY LEVEL ON PUBLIC HEALTH AND SAFETY)



FLOOR PLAN

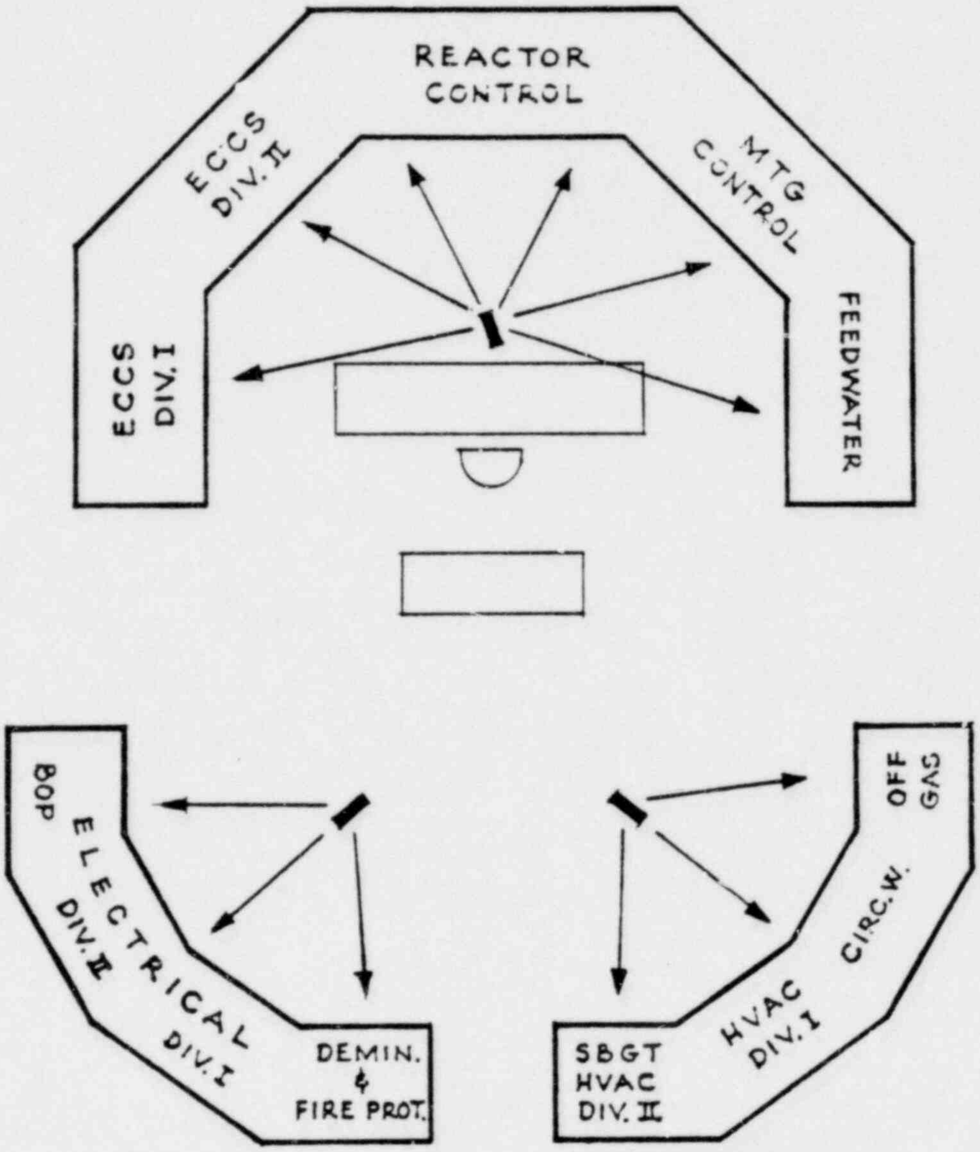
TECHNICAL SUPPORT CENTER

FIGURE 4

FIGURE 5

TECHNICAL SUPPORT CENTER INFORMATION SYSTEM

- Objective: To provide information in the TSC for analysis of the plant conditions
- Functional Spec first draft reviewed in BWR Sub- group's October meeting
- Primarily a computer bases system with CRTs in TSC
- Part of information required in TSC is provided by closed circuit TV from existing control room displays
- Display of independently selectable menus from SPDS
- Edison uses the parameter set developed by the BWR Owners Control Room Subgroup
- Historical data for a selected smaller set of variables
- The data is sufficient to determine the plant steady state and dynamic behavior prior to and throughout the course of an accident
- The TSC receives all data provided at EOF and on the NDL



CONTROL ROOM

CCTV CAMERA LOCATIONS

FIGURE 6

FIGURE 7

SAFETY PARAMETER DISPLAY SYSTEM

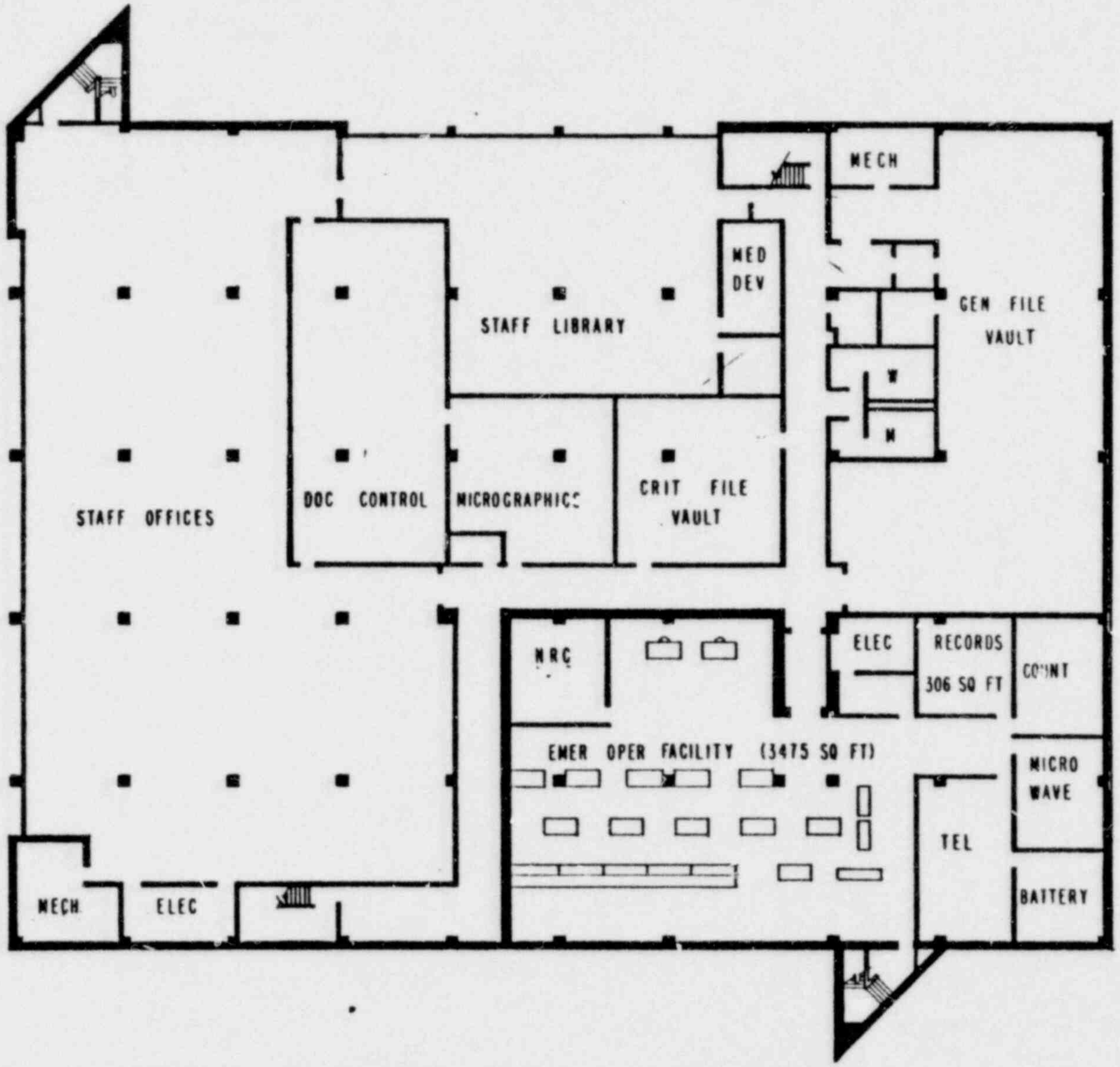
- Functional Spec for SPDS completed
  - Based on Emergency Procedure entry parameters indicating
    - Core cooling
    - Fuel integrity
    - Reactivity
    - Reactor coolant system integrity
    - Containment integrity
    - Radiation released to environment
  - Fifteen variables are required to show the above parameter set. Some are represented by several inputs
- The values of the parameters are validated by a second variable or second input of the same variable
- The computer system is self-checking
- The system is high quality but not seismic
- Design Spec for SPDS scheduled May, 1981
- Edison will place order as soon as Spec available
- One or more displays in Control Room
- One display in TSC
- The processor in computer mezzanine

FIGURE 8

EMERGENCY OPERATIONS FACILITY

INFORMATION SYSTEMS

- THE EOF IS ADDRESSING OFFSITE AND PUBLIC PROTECTION MATTERS.
- METEOROLOGICAL AND RADIATION RELEASE INFORMATION IS PROVIDED.
- EOF PERSONNEL DO NOT DETERMINE PLANT SAFETY, BUT OBTAINS THAT INFORMATION FROM TSC.
- IT IS ESSENTIAL TO ACCURATELY ASSESS THE PLANT SAFETY USING THE BEST QUALIFIED PEOPLE WHICH ARE LOCATED IN THE TSC.
- NO SPDS DISPLAY WILL BE PROVIDED IN THE EOF BECAUSE IT MAY BE MISLEADING TO PEOPLE NOT FAMILIAR WITH THE PLANT SYSTEMS.



EMERGENCY OPERATIONS FACILITY

FIGURE 9

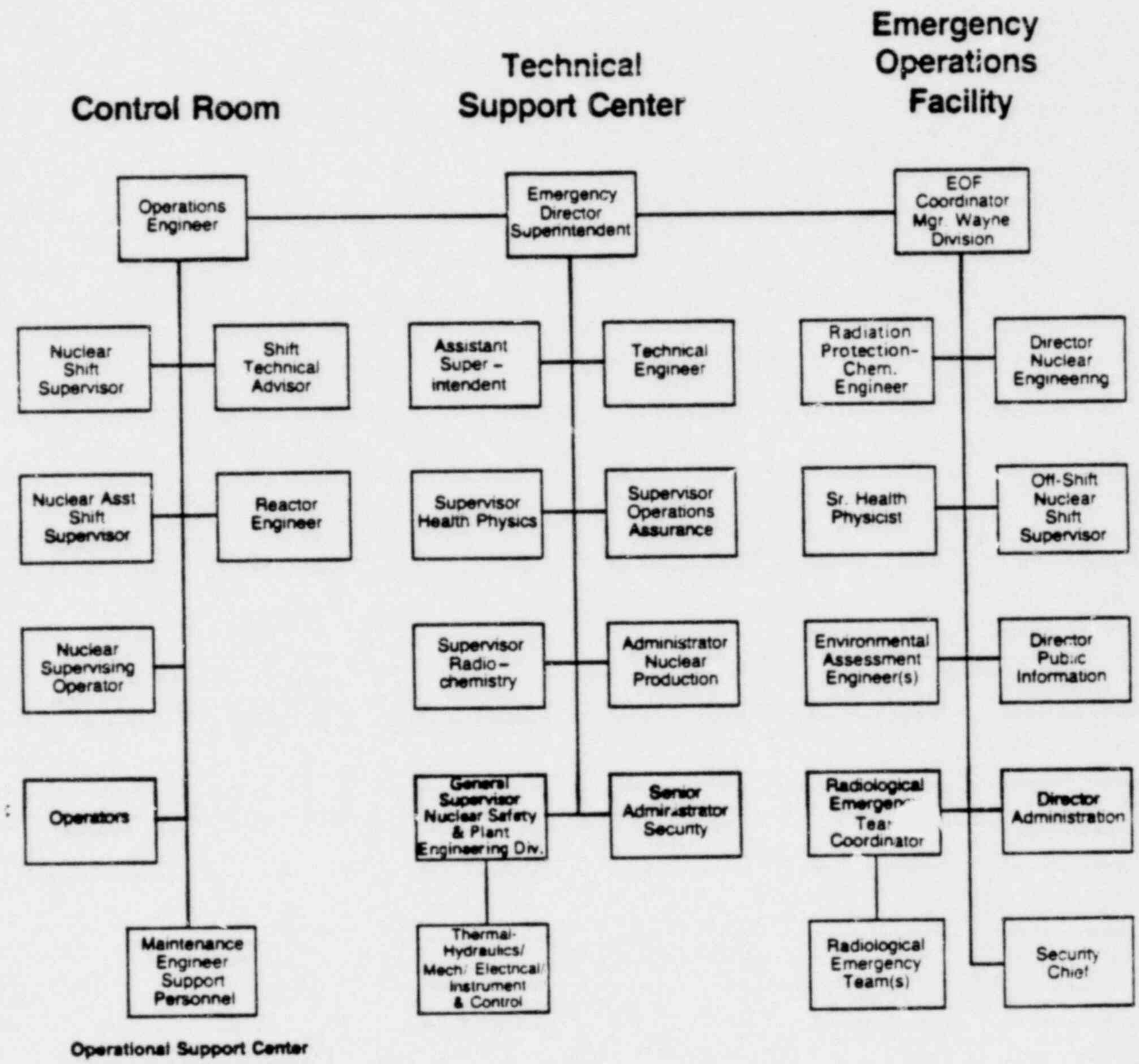
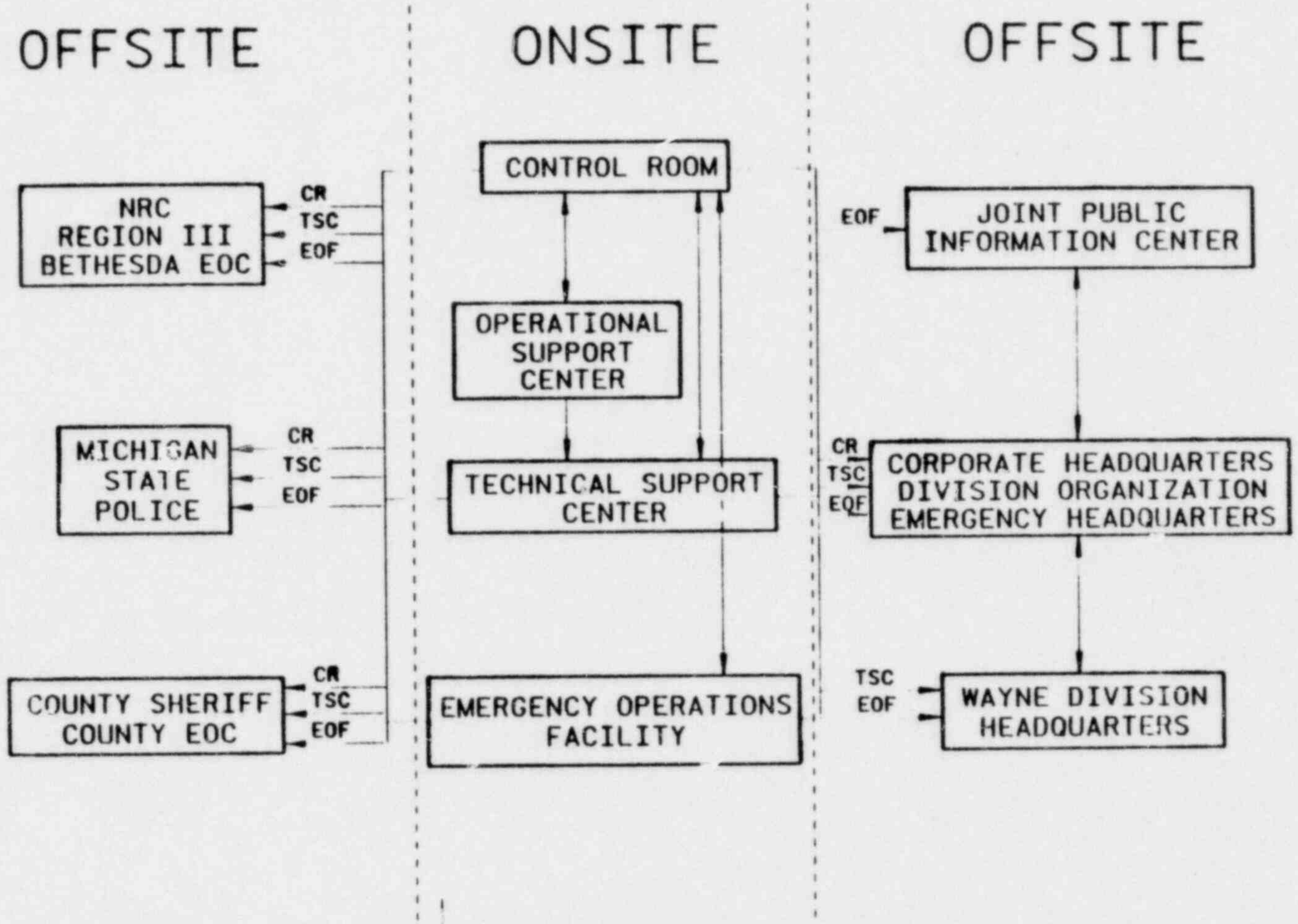

Control Room
Technical Support Center
Emergency Operations Facility
Operations Engineer
Emergency Director Superintendent
EOF Coordinator Mgr. Wayne Division
Nuclear Shift Supervisor
Shift Technical Advisor
Assistant Super-intendent
Technical Engineer
Radiation Protection-Chem. Engineer
Director Nuclear Engineering
Nuclear Asst Shift Supervisor
Reactor Engineer
Supervisor Health Physics
Supervisor Operations Assurance
Sr. Health Physicist
Off-Shift Nuclear Shift Supervisor
Nuclear Supervising Operator
Supervisor Radio-chemistry
Administrator Nuclear Production
Environmental Assessment Engineer(s)
Director Public Information
Operators
General Supervisor Nuclear Safety & Plant Engineering Div.
Senior Administrator Security
Director Administration
Maintenance Engineer Support Personnel
Thermal-Hydraulics/ Mech/ Electrical/ Instrument & Control
Radiological Emergency Team(s)
Security Chief
Operational Support Center

FIGURE 10



RADIOLOCICAL EMERGENCY RESPONSE
COMMUNICATIONS

FIGURE 11

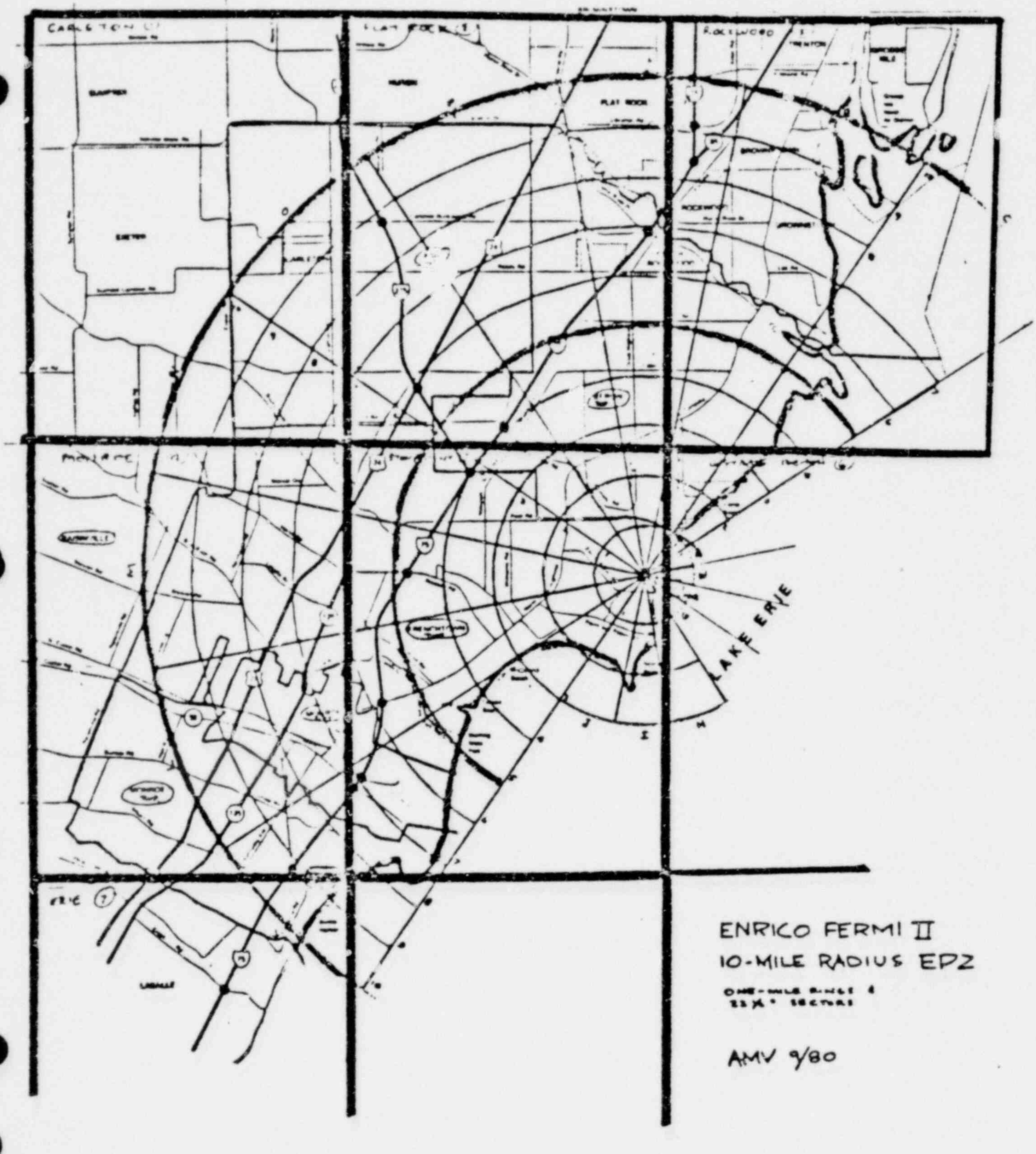
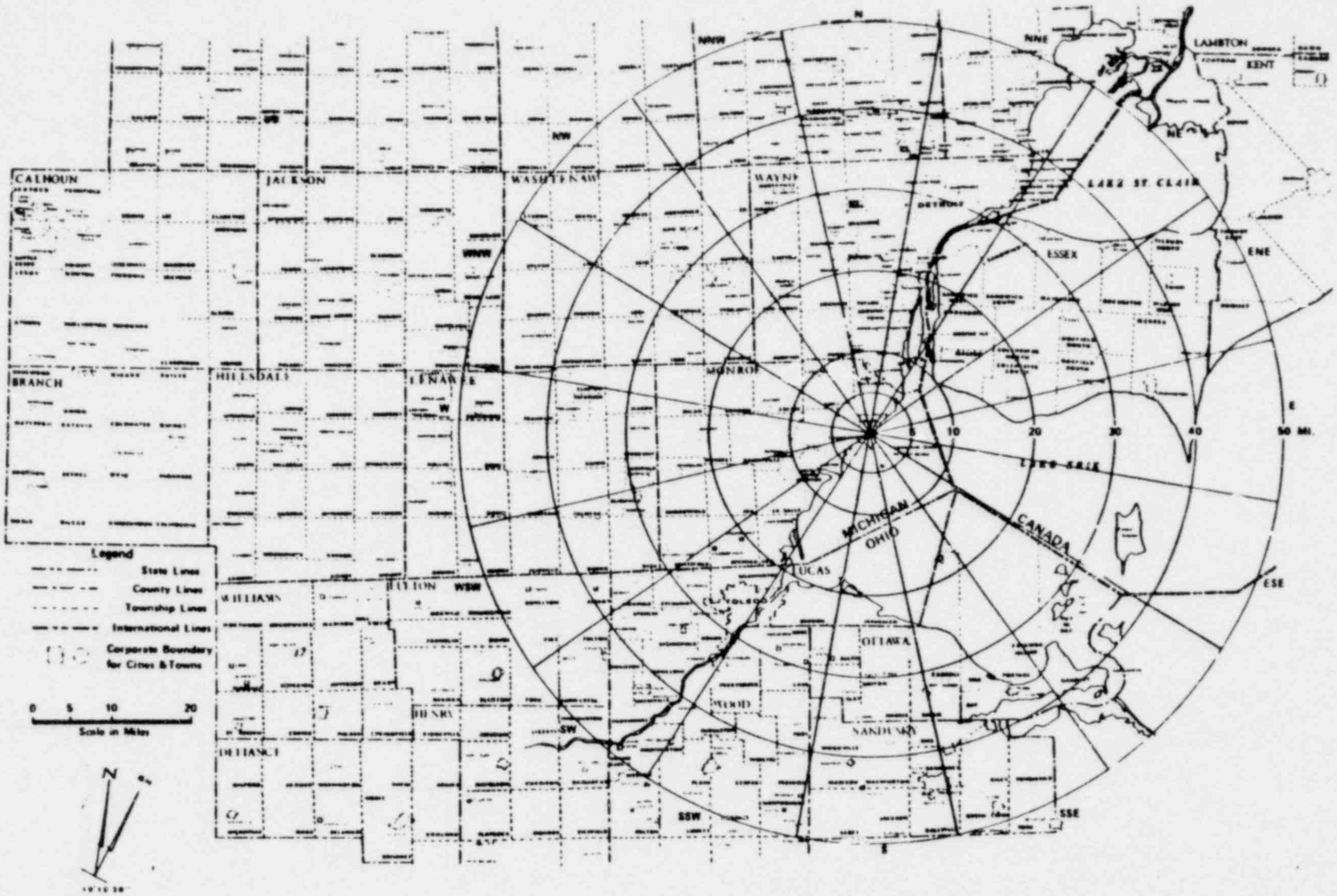


FIGURE 12



ENRICO FERMI ATOMIC POWER PLANT, UNIT 2
50 MILE EMERGENCY PLANNING ZONE

FIGURE 13

FIGURE 14

TENTATIVE SCHEDULE FOR FERMI 2 RADIOLOGICAL EMERGENCY RESPONSE PLAN (RERP)

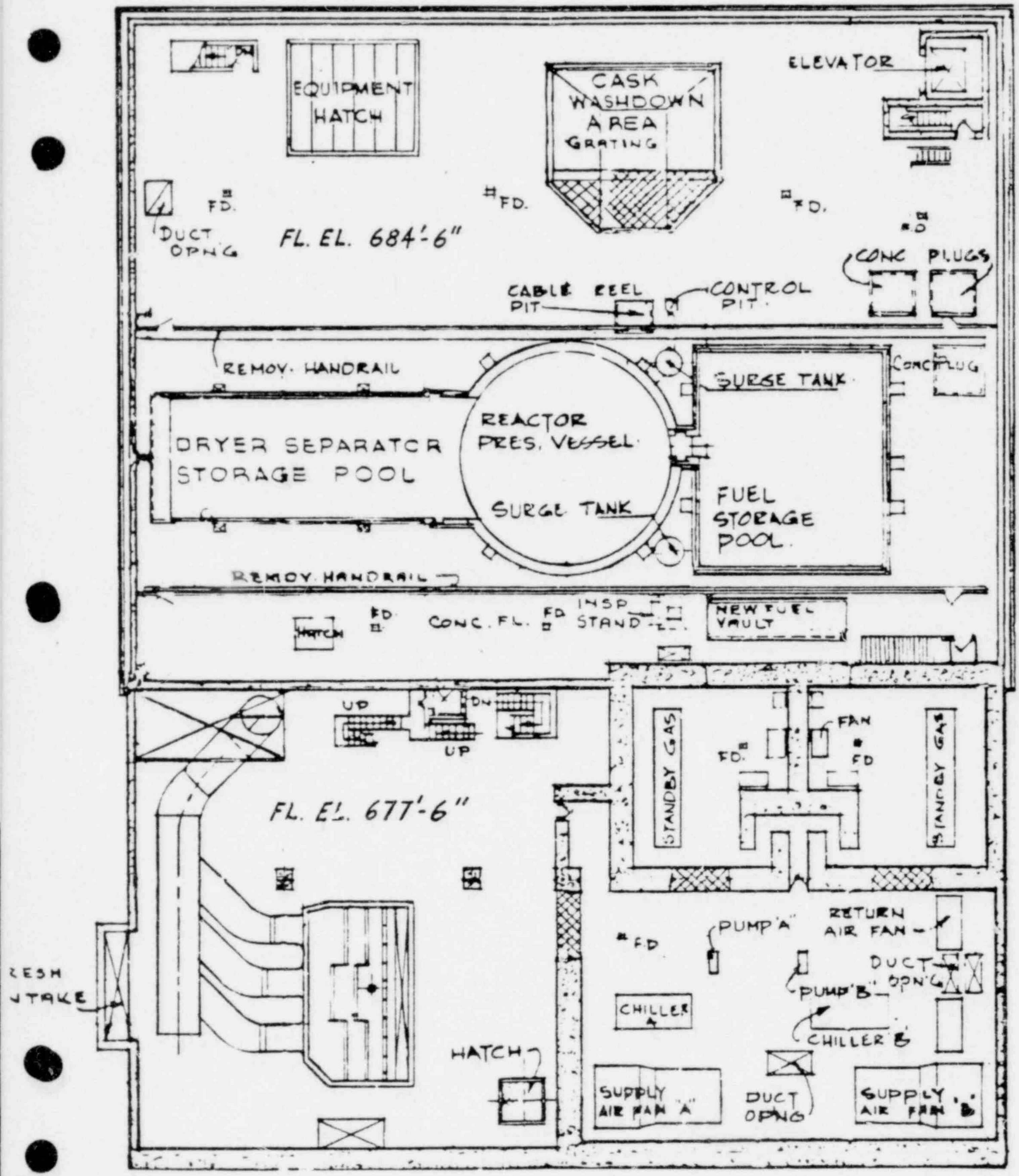
• State of Michigan Plan Submitted to FEMA, Region V	February, 1981
• Fermi 2 RERP Submitted to NRC (10CFR50)	March 31, 1981
• Responses to NRC Comments	August 30, 1981
• FEMA, Region V recommendations to FEMA headquarters on Michigan plan	July, 1981
• Michigan submit to FEMA, Region V for approval • Michigan State Plan • Monroe County • Jurisdictions within Wayne Co. in 10 mile EPZ	November, 1981
• Fermi 2 full scale exercise and NRC appraisal	February, 1982 (first 2 weeks)

SPENT FUEL STORAGE AT FERMI 2

THE SPENT FUEL STORAGE POOL FOR FERMI 2 IS LOCATED ON THE FIFTH FLOOR OF SECONDARY CONTAINMENT ("REACTOR BUILDING"), AS IS TYPICAL FOR BWR'S WITH MARK I CONTAINMENT. THE POOL IS 40 FEED BY 34 FEET BY 39 FEET DEEP.

"HIGH-DENSITY" TYPE STORAGE RACKS HAVE BEEN INCORPORATED TO INCREASE THE NUMBER OF STORAGE CELLS TO APPROXIMATELY 2300 TYPICALLY THE STORAGE CELLS ARE ARRANGED IN 13 X 13 ARRAYS MOUNTED ON FREE-STANDING BASE PLATES. THE CELL WALLS ARE TWO STAINLESS STEEL SHEETS (0.075 INCHES THICK) SANDWICHING A 0.07-INCH BORAFLEX SHEET.

THE CAPACITY OF THESE RACKS GIVES STORAGE FOR SPENT FUEL FOR NINE YEARS OF OPERATION (AT 80% CAPACITY FACTOR) AND STILL HAVE THE ABILITY TO UNLOAD AND STORE ALL FUEL IN THE REACTOR VESSEL.


EQUIPMENT HATCH
CASK WASHDOWN AREA
GRATING
ELEVATOR
FD.
DUCT OPNG
FL. EL. 684'-6"
CONC PLUGS
CABLE REEL PIT
CONTROL PIT
REMOV. HANDRAIL
SURGE TANK
CONC PLUG
REACTOR PRES. VESSEL
DRYER SEPARATOR STORAGE POOL
SURGE TANK
FUEL STORAGE POOL
REMOV. HANDRAIL
HATCH
CONC. FL.
INSP STAND
NEW FUEL VAULT
UP
DN
UP
STANDBY GAS
FAN
FD
STANDBY GAS
FL. EL. 677'-6"
RESH
NTAKE
PUMP 'A'
RETURN AIR FAN
DUCT OPNG
PUMP 'B'
CHILLER A
CHILLER B
HATCH
SUPPLY AIR FAN 'A'
DUCT OPNG
SUPPLY AIR FAN 'B'

II.F.2 - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (ICC)

MINIMUM INSTRUMENTATION FOR ICC MONITORING SYSTEM

ICC INSTRUMENTS (SEE NOTE 1)	REACTOR TYPE (SEE NOTE 2)			
	WESTINGHOUSE	CE	B&W	GE
SATURATION METER	REQUIRED	REQUIRED	REQUIRED	NOT REQUIRED
COOLANT INVENTORY (LEVEL) ABOVE CORE	REQUIRED	REQUIRED	REQUIRED*	REQUIRED
COOLANT INVENTORY (LEVEL) WITHIN CORE	DESIRABLE	NOT REQUIRED**	NOT REQUIRED**	REQUIRED
CORE EXIT T/Cs	REQUIRED	REQUIRED	REQUIRED	CORE T/C REQUIRED*

* NO FIRM DESIGN PROPOSED BY VENDOR.

** NOT REQUIRED PROVIDED THAT CORE EXIT THERMOCOUPLE INFORMATION IS PROCESSED, RECORDED, AND DISPLAYED IN AN ACCEPTABLE MANNER TO FACILITATE INTERPRETATION OF CORE COOLING CONDITIONS IN CONJUNCTION WITH ABOVE-CORE LEVEL INSTRUMENTATION.

NOTE 1: LEVEL INSTRUMENTATION MUST BE TESTED AND EVALUATED FOR LARGE BREAK LOCA SURVIVABILITY AND POST LOCA OPERABILITY.

NOTE 2: REQUIREMENTS ARE BASED ON REACTOR VENDOR PROPOSED INSTRUMENTATION, INTERCHANGEABILITY OF INSTRUMENTATION SYSTEMS IS ACCEPTABLE.

BASIS FOR STAFF POSITION

ON ICC MONITORING SYSTEMS

INSTRUMENT	REFERENCE FOR REQUIREMENT
SATURATION METER	NUREG-0660 NUREG-0737 R.G. 1.97
COOLANT INVENTORY (LEVEL)	NUREG-0660 NUREG-0737 (CLARIFICATION ITEM 6) R.G. 1.97*
CORE EXIT T/Cs	FOR PWR: NUREG-0737 (II.F.2 ATTACHMENT 1) R.G. 1.97 FOR BWR: NUREG-0737 (NOT SPECIFIC) NUREG-0519 (LASALLE SER) R.G. 1.97

*ICC INDICATION RANGE

BWR - FROM BOTTON OF CORE SUPPORT PLATE TO LESSER OF TOP OF VESSEL OR CENTERLINE OF MAIN STEAM LINE

PWR - FROM BOTTOM OF CORE TO TOP OF VESSEL

NEED FOR BWR THERMOCOUPLES

1. DIVERSE WATER LEVEL INDICATION

2. MONITOR CORE COOLING EFFECTIVENESS

3. MONITOR OPERABILITY OF CORE SPRAY

A. NRC INTRODUCTION

LOCATION - ON LAKE ERIE AT MONROE, MICHIGAN

REACTOR - BWR 4, 3292 MWT

CONTAINMENT - MARK I

APPLICATION FOR CP - APRIL 1969

CONSTRUCTION PERMIT ISSUED - SEPTEMBER 1972

APPLICATION FOR OL - APRIL 1975

1. OVERVIEW OF OL REVIEW
2. SER OPEN ITEMS

A.1. OVERVIEW OF OL REVIEW

FIRST REVIEW PERIOD (17 MONTHS)

FSAR DOCKETED APRIL 1975

REVIEW AREAS COMPLETED:

. SITE CHARACTERISTICS (METEOROLOGY, FLOOD, GEOLOGY)

. CRITERIA FOR DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS (CLASSIFICATION, FLOOD PROTECTION, MISSILE PROTECTION, PROTECTION FOR POSTULATED PIPE FAILURES, DESIGN BASIS FOR CATEGORY I STRUCTURES)

. AUXILIARY SYSTEMS (COOLING WATER, HEATING AND VENTILATION)

. RADIATION PROTECTION (SHIELDING)

REVIEW SUSPENDED FOR CONSTRUCTION DELAY - SEPT. 1976

SECOND REVIEW PERIOD (9 MONTHS)

FSAR UPDATED JUNE 1978

REVIEW AREAS COMPLETED:

. MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

. REACTOR PHYSICS

. RADIOACTIVE WASTE MANAGEMENT (APPENDIX I)

REVIEW SUSPENDED FOR TMI PAUSE MARCH 1979

THIRD REVIEW PERIOD (4 MONTHS)

AMENDMENT 33 TO FSAR (RESPONSE TO TMI ACTION ITEMS) MARCH 1981

REVIEW AREAS COMPLETED:

- SITE CHARACTERISTICS (GEOGRAPHY, DEMOGRAPHY, SEISMOLOGY, FOUNDATIONS)
- STRUCTURES (MASONRY WALLS, HIGH DENSITY SPENT FUEL POOL)
- MECHANICAL SYSTEMS & COMPONENTS
- REACTOR (MECHANICAL DESIGN, THERMAL & HYDRAULIC DESIGN, MATERIALS, CONTROL ROD DRIVE SYSTEM)
- REACTOR COOLANT PRESSURE BOUNDARY (MATERIALS, OVERPRESSURIZATION PROTECTION, PRESERVICE INSPECTION LEAKAGE DETECTION)
- CONTAINMENT (FUNCTIONAL DESIGN, ISOLATION VALVES, FRACTURE TOUGHNESS)
- ECCS, RCIC, RHR
- INSTRUMENTATION & CONTROLS
- ELECTRICAL POWER (EMERGENCY DIESEL ENGINE AUXILIARIES)
- CONDUCT OF OPERATIONS
- INITIAL TEST PROGRAM
- QA FOR OPERATION (Q-LIST)
- SAFETY ANALYSES (TRANSIENTS AND ACCIDENTS INCLUDING OFFSITE DOSE CONSEQUENCES)
- FINANCIAL QUALIFICATION
- RESPONSE TO TMI-2 REQUIREMENTS

A.2 SER OPEN ISSUES TO BE COMPLETED IN SSER (AUGUST 31, 1981)

- CONFORMANCE TO 10 CFR 20, 50, 100
- SEISMIC REASSESSMENT OF DESIGN MARGIN
- PRESERVICE TESTING OF PUMPS & VALVES
- SEISMIC QUALIFICATION REVIEW TEAM AUDIT
- BURIED PIPE FOUNDATION CONDITIONS
- CONFORMANCE TO APPENDIX G & H, 10 CFR 50
- CONTAINMENT LEAKAGE TESTS
- PROCEDURE FOR TESTING RHR ISOLATION VALVE INTERLOCKS
- LOSS OF INSTRUMENTATION & CONTROL POWER (IE BULLETIN 79-27)
- FIRE PROTECTION (CONTROL ROOM)
- PHYSICAL SECURITY PLAN
- EMERGENCY OPERATING PROCEDURES (I.C.1, I.C.8, ATWS)
- FEEDBACK OF OPERATING EXPERIENCE (I.C.5)
- CONTROL ROOM DESIGN (I.D.1)
- DEGRADED CORE TRAINING (II.B.4)
- CONTAINMENT PURGE OPERABILITY (II.E.4.2)

TO BE COMPLETED PRIOR TO OL ISSUANCE (NOV. 1982)

- DESIGN OF MODIFICATION TO DIESEL ENGINES
- ENVIRONMENTAL QUALIFICATION OF EQUIPMENT
- UPGRADED EMERGENCY PREPAREDNESS
  (III.A.1.1, III.A.1.2, III.A.2)
- MARK I CONTAINMENT
  PLANT UNIQUE ANALYSIS
  TORUS - ATTACHED PIPING ANALYSIS
- SAFETY ANALYSIS AND PROCEDURES FOR TRAINING DURING LOW
  POWER TESTING (I.G.1)

(6)

TO BE COMPLETED AFTER LICENSE ISSUANCE (LICENSE CONDITIONS)

. ANALYSIS OF FISSION GAS IN FUEL

. TESTS OF FUEL CHANNEL BOX DEFLECTION

. ANALYSES OF HYDRODYNAMIC STABILITY

. ANALYSIS OF MULTIPLE CONTROL SYSTEM FAILURES

. ANALYSIS OF EFFECT ON HIGH ENERGY LINE BREAK ON CONTROL SYSTEMS

. INSPECTION OF LOW PRESSURE TURBINE DISCS

. DESIGN OF POST ACCIDENT SAMPLING SYSTEM

. DESIGN OF INSTRUMENTATION FOR INADEQUATE CORE COOLING