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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

354th ACRS MEETING)
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DAY TWO)
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Friday,
October 6, 1989

Room P-110, Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland

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

BEFORE: DR. FORREST J. REMICK
Chairman, ACRS
Associate Vice-President for Research
Professor of Nuclear Engineering
The Pennsylvania State University
University Park, Pennsylvania

ACRS MEMBERS PRESENT:

DR. WILLIAM KERR
Professor of Nuclear Engineering
Director, Office of Energy Research
University of Michigan
Ann Arbor, Michigan

MR. CHARLES J. WYLIE
Retired Chief Engineer
Electrical Division
Duke Power Company
Charlotte, North Carolina

DR. PAUL G. SHEWMON
Professor, Metallurgical Engineering Department
Ohio State University
Columbus, Ohio

ACRS MEMBERS PRESENT (Continued):

DR. CHESTER P. SIESS
Professor Emeritus of Civil Engineering
University of Illinois
Urbana, Illinois

MR. DAVID A. WARD
Research Manager on Special Assignment
E.I. du Pont de Nemours & Company
Savannah River Laboratory
Aiken, South Carolina

DR. HAROLD W. LEWIS
Professor of Physics
Department of Physics
University of California
Santa Barbara, California

MR. CARLYLE MICHELSON
Retired Principal Nuclear Engineer
Tennessee Valley Authority
Knoxville, Tennessee, and
Retired Director, Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, D.C.

MR. JAMES CARROLL
Retired Manager, Nuclear Operations Support
Pacific Gas & Electric Company
San Francisco, California

DR. IVAN CATTON
Professor of Engineering
Department of Mechanical, Aerospace & Nuclear Engineering
School of Engineering and Applied Science
University of California
Los Angeles, California

P R O C E E D I N G S

DR. REMICK: The dinner must have settled well with everybody overnight. Everybody is prompt and early this morning ready to go.

The meeting will now order. This is the second day of the 354th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the committee will discuss and/or hear reports of the following: Generic Issue B-56, Diesel Reliability and Associated Regulatory Guide 1.9, Rev. 3; Generic Issue 87, Failure of HPCI Steamline Without Isolation; continue or discussion of adequate protection, ACRS Subcommittee activities; standardized plant design, the can-do 3; and maintenance of power plants, a continuation of our discussion of that subject.

The meeting is being conducted in accordance with provisions of the Federal Advisory Committee Act and the Government Sunshine Act. Medhoff F. Saltowee is the designated federal official for the initial portion of the meeting.

We have received no written comments or requests to make oral statements from members of the public regarding today's meeting.

The first item on today's agenda is the discussion of Generic Issue B-56, Diesel Reliability, and the associated Regulatory Guide 1.9, Rev 3.

Charlie Wylie is our subcommittee chairman. So, Charlie, I turn the meeting over to you for the next hour.

MR. WYLIE: All right, thank you, Mr. Chairman.

The information for this part of the meeting is contained in Tab 7 of your books. The AC-DC power systems reliability subcommittee held a meeting on Monday of this week. In attendance was Jay Carroll, myself and ACRS consultant, Peter Davis. The purpose of the meeting was to review and discuss with the staff the proposed final resolution of Generic Safety Issue B-56, diesel generator reliability.

To refresh your memories, Generic Safety Issue B-56 on diesel generator reliability is a safety issue which is related to the station blackout rule. It's been around a long time; initiated in 1977. And the staff issued Regulatory Guide 1.155 on station blackout to provide compliance with the blackout rule.

This identified the need for insuring the reliability of the diesel generators to a reliability level of .95 or better. The staff has proposed the resolution of Generic Safety Issue B-56 be accomplished through the issuance of a Revision 3 to an existing Regulatory Guide 1.9, which is entitled "The Selection Design Qualification Test and Reliability of Diesel Generators Used as On-site

Electric Power Systems at Nuclear Power Plants".

The staff's proposed revision to Reg. Guide 1.9 is intended to integrate all the requirements that are now contained in several reg guides and generic letters into a single regulatory guide, to define the principal elements of a diesel generator reliability program which, for the most part, is consistent with present industry practices, and to better define testing requirements to make possible the elimination of cold fast starts, and it reduces the previous requirements for accelerated testing.

In the proposed revision, they have assigned what they call alert levels and remedial actions to be taken if a deterioration of the reliability is encountered for the diesel generators.

In addition, the staff concludes that the issuance of the regulatory guide and the implementation of the diesel generator reliability program will obviate the need to consider diesel generator aging explicitly in the licensing renewal process.

The staff indicates that the resolution of this generic issue and the issuance of Reg. Guide 1.9, Revision 3, will not introduce any new regulatory requirements beyond those currently required for compliance with the station blackout rule.

Reg. Guide 1.9, Revision 3, was sent out for public comments. The staff has considered and incorporated many of those comments in the present draft, and has worked very closely with industry, and specifically NUMARC, in the development of the reg. guide. At the time of the subcommittee meeting there were still some differences between the industry and the staff which they were working on, and perhaps the staff could tell us today how some of those came out.

Generally, the subcommittee felt that the staff was on the right track. The reg. guide, because it's an integration of so many different items and design and maintenance and qualification and testing and reliability and all the other things that we found it somewhat confusing and perhaps would have been better had it been issued in two reg. guides; one covering application design and qualification, and the other separate guide covering the reliability program.

With that, I'll ask Jay, do you have anything to add?

MR. CARROLL: No, I think that's a fine summary. I guess the one thing that we learned at the subcommittee meeting is that there is a difference on the staff. I guess what we're going to be looking at today is what's been signed off by Research and NRR. And I guess Faust and the electrical engineering branch in one area have a disagreement which we will also hear about, because Faust indicated he is going to sign off on the document.

MR. WYLIE: That's correct.

MR. SHEWNON: We will hear about the response to their resolution to the public comment today, and this is a final reg. guide that is going forward now; is that right?

MR. WYLIE: That's correct.

Yes, the staff, represented by Alex Serkiz, is here today to present the staff, and then we have NUMARC representatives here to give us their views of the reg. guide. So I guess we will call on Mr. Serkiz.

(Slide presentation.)

MR. SERKIZ: Thank you, Mr. Wylie.

I think the introduction covers some of the initial slide material that I'll be presenting to you and discussing with you.

As the agenda indicates, the purpose of this meeting is to review with the full committee the resolution of Generic Safety Issue B-56.

My name is Alex Serkiz. I am with the Office of Research and Reactor and Plant Safety Issues Branch. Also, as was indicated by Mr. Carroll, the regulatory guide that was sent to the ACRS does represent the concurred imposition, both by Research and NRR management.

Faust Rosa is here and will speak to you later on a position that is held by he and members of his branch.

I would like to perhaps approach this morning's meeting to look at it from the viewpoint that this is not a new issue. It is related to the station blackout rule as was indicated, and it is a continuation and for the completion of the resolution of A-44. A-44 established these needs.

As you see in your second slide, this indeed does carry out the intent of the items that were covered by Mr. Wylie before. And it has been revised in response to comments obtained from the industry, NUMARC's working group, and it does integrate into a singular regulatory guide program requirements, or rather, program reliability guidance testing, guidance, et cetera.

The regulatory guide, as you note from the title, does contain a variety of guidance that deals both with the testing qualification. And what I would like to speak to you principally this morning about as a full committee are the reliability and reliability monitoring aspects.

If you have questions on other parts of the regulatory guide, certainly we will speak to them. In addition, I think what is important to note is that the definition section of the regulatory guide, namely, Section 2.1, the staff and NUMARC are working on this to make the language of the definition of fail, fail and start, successes as nearly identical as possible with the INPO definitions.

DR. REMICK: Could you elaborate a little bit on bullet 4, eliminate cold fast starts, and tell me a little bit more about that.

MR. SERKIZ: Okay. Let me go through bullet 4 one at a time.

The testing requirements are outlined in the regulatory guide. They were reviewed for consistency with current practices, better phraseology for guidance, et cetera.

With respect to eliminating cold fast starts, the regulatory guide provides guidance that the diesel can be prelubed, preconditions per normal manufacturer's operating conditions so that the surveillance testing is carried out, whereas before through Reg. Guide 1.008, many times the guidance said do a fast cold start, push the button.

DR. REMICK: How about the fast aspect?

MR. SERKIZ: The fast aspect, the regulatory guide sets up an umbrella of reliability program. And in doing so, retains within the schedule of surveillance testing that there will be fast starts every six months. That is a point of agreement to disagree, I prefer to state it this way, between the NUMARC people and the staff.

The staff feels that a fast start, if you will, fast start in terms of every six months, is a necessary condition because there are other considerations in the overall scheme of things; namely, the plant transient of a LOCA which is a different type of transient in a station blackout.

DR. REMICK: Could you contrast that with what are the current, or what is the current guidance?

MR. SERKIZ: Well, the current guidance comes through Reg. Guide 1.008, and the current Regulatory Guide 1.9, Rev. 2, and is tied to tech specs. I would characterize it this way.

That there are plants out there, I don't know the exact number. Faust, you may have a better feel for it. But perhaps on the order of 20 percent of the population because they are tied into tech specs that were written years ago, are retaining for whatever purposes cold fast starts.

Is that a correct statement?

DR. REMICK: What is meant by "fast" in that? I'm trying to contrast the guidance that you are proposing here with what is current guidance.

MR. ROSA: I'm Faust Rosa, chief of the electrical systems branch, NRR.

About 20 percent of the plants retain the fast starts as called for in their technical specifications. And those fast starts simply are, given whatever prewarming and lubing that they have in their design, and some don't have any prewarming or prelubing. They just simply press the button and monitor how quickly the diesel comes up to speed and voltage.

DR. SHEWMON: And they try to bring it up to full power in 10 seconds and do it once a month. Is that --

MR. ROSA: I think they need only go up to about 50 percent of design accident load for those tests.

DR. REMICK: How fast?

What I'm trying to get at once again --

MR. ROSA: Ten seconds.

DR. REMICK: Ten seconds.

MR. ROSA: Usually 10 to 12 seconds.

DR. REMICK: OKay. And now you are maintaining that, but the frequency of that test has been changed from something to six months? Am I understanding you properly or am I misunderstanding?

MR. ROSA: That's correct.

Those tests were required periodically every month.

DR. REMICK: Every month.

MR. ROSA: And at an accelerated frequency if they incurred failures.

DR. REMICK: Okay. So the cold is no longer required if they have a way of preheating, prelubing. The fast is still maintained, but on a less frequent basis.

Is that the change?

MR. SERKIZ: On a less frequent basis, and I would like to speak to that in a subsequent slide.

DR. REMICK: Okay, fine.

MR. SERKIZ: The old method was and the old guidance in Reg. Guide 1.108 and other documents such as the Rev. 2 of this regulatory guide, there were times when you hit a count on a particular diesel of two out of 20, you went into accelerated testing.

We have a different approach here in this regulatory guide to accelerated testings, and that is minimizing it, but tieing it to a problem diesel, which I'll speak to later.

MR. CARROLL: I guess the one piece that you can help the rest of the committee on, I think, Faust, is what happens now on the monthly test? How is it done? It's prelubed and prewired.

MR. ROSA: On a monthly test, most, say 80 percent of the diesels out there have either originally or backfitted the prewarming, prelube features.

DR. REMICK: Okay.

MR. ROSA: So they have the diesel at prewarmed conditions, and the lube system is operating before they start.

Now, they start the diesel. And if they have a slow start capability in their controls, they can come up to rated speed over a period of minutes. And then they can load very gradually. They can load say 10 percent or 15 percent in the first step, to 25 percent after 10 or 15 minutes and so on, until they reach the designed load of the diesel. And then they maintain that for about an hour.

MR. SERKIZ: So we are backing away from the more frequent occurrence that occurs out in the field now.

Sir?

DR. SHEWMON: This sort of screening start, it seems to me, is recommended by nobody for diesel maintenance and reliability. The driving force for it, as I understand it, was originally a guillotine break, which has, (a) never occurred; and (b) shown to be such low credibility that we've instituted leak-before-break. Therefore, many of us would say it's incredible.

Is there some other basis for maintaining these abusive starts even on a six-month basis?

MR. SERKIZ: Well, we are not maintaining it eve on a six-month, because there is a definite difference in the testing description as to the condition of the diesel. So prelubing is an important part of minimizing that type of adverse effect. Prelubing is, I guess, generally speaking, quite widespread.

Ed, would you care to speak to that, because the question came up?

This is Ed Tomlinson, who is --

DR. SHEWMON: Well, you can answer the question you want to answer, but I'd like to come back to mine before we get done.

MR. SERKIZ: All right. I was going to suggest that perhaps Ed describe the starts and the cold start effects.

DR. KERR: No, the question, however, was whether there is any reason other than the guillotine break to have a fast start.

Can you answer that question?

MR. SERKIZ: Yes, sir.

The reason for periodic, although at a reduced frequency, maintaining of a fast start is tied not so much to a guillotine break, but there are other plant transients that the staff feels will be more rapid or more -- they will have an adverse effect and should be handled more quickly in that context.

DR. KERR: Which are they?

MR. SERKIZ: I'm concerned about seal failure for one, for example.

DR. SHEWMON: Do you have a matter of under a minute to cope with a seal failure?

MR. SERKIZ: No, sir.

DR. SHEWMON: Pardon?

MR. SERKIZ: No, sir.

DR. SHEWMON: Okay.

MR. SERKIZ: The NRR staff right now has recommended maintaining a six-month surveillance frequency on fast start.

DR. SHEWMON: You've made that clear. I'm trying to find out the basis for that decision.

MR. SERKIZ: Okay. It feels that they have not adequately reviewed the analyses and/or models that have been submitted to show that this should not be pursued further and it would not make a commitment at the present time to remove that remove that requirement.

DR. SHEWMON: What you are telling me is you may remove it. You currently don't have a good basis for it except that you thought it used to be a good basis 10 years ago and you haven't looked at what something yet.

MR. SERKIZ: N, sir. I wouldn't characterize -- the staff right now feels, because they have not completed their review of analyses that have been submitted on these other plant transients and have not adequately gone through and rolled these over into the approved models, feels that it's premature to take out that requirement right now.

DR. KERR: So would it be fair to characterize it as the staff doesn't know any reason for having a fast start, nor does it know any reason for not having a fast start, and since it has been tradition to have a fast start, we're going to continue to have that?

DR. CATTON: But at a lesser rate.

MR. SERKIZ: At a lesser rate.

DR. KERR: Yes, but --

MR. SERKIZ: That's correct. They say they have not completed and concluded their review of these reports that have been submitted.

DR. SHEWMON: Is there a commitment to revisit this within the next year or some reasonable period?

MR. SERKIZ: I don't believe there is a commitment right now that I can cite. But I've been told that when that review is completed it will be logical to come back and revisit it.

MR. CARROLL: What are these reports that you are referring to? Are these owners group reports?

MR. SERKIZ: Yes, they are owner groups reports. There are several EPRI reports and SAC reports that deal with transients in both PWRs and BWRs, deal specifically with the fast start issues. I don't remember off the top of my head what the report numbers are. There are two or three INSAC numbers that I've seen that indicate that the

10-second rule is extremely conservative and not necessary, I'm paraphrasing this. That something on the order of 30, 35 or 50-second start is certainly adequately.

DR. LEWIS: Is there a consensus that -- what Bill said earlier -- that a fast start is -- maybe it was Paul -- is deleterious to the diesels so that the burden of proof should be on those who want to have fast start testing? Is there a consensus on that point?

MR. MINNERS: I think you have to break it down. I think I have the confusion you have, Dr. Lewis, not being a diesel expert.

DR. LEWIS: I'm sorry, I don't have any confusion. I thought I asked a question which had a yes or no answer possible.

MR. MINNERS: Well, then, I had the confusion.

The confusion that you are talking about, fast start of the diesel or fast loading of the diesel. Fast start of the diesel, as I understand it, is a consequence of the control system and most diesels fast start. You press the bottom. They fast start. I guess there are some which have a modified system which doesn't do that, but most fast start.

And I presume what you are asking is, is the continual fast loading of the diesel requirement.

DR. LEWIS: A fast start with load, right. That was the question to which I was hoping for a yes or no answer. Let me repeat it.

Is there consensus that this has negative effects on the diesel and that therefore the burden of proof is on those who want to preserve the fast start and fast load testing?

MR. ROSA: I think I can add one item of information here that might be useful. Even in these fast starts during our test, the loading is manually on the grid, and it takes about two minutes to load the diesel even during one of these fast starts. So it's not as if you load completely in those 10 - 12 seconds.

DR. KERR: But you can't answer the question that Dr. Lewis asked, I take it.

DR. SHEWMON: My impression is that the answer is yes. We could ask NUMARC if the staff can't.

MR. MINNERS: We've got a diesel expert.

Ed, would you like to address the question.

DR. LEWIS: Well, I was hoping the staff could answer yes or not.

MR. MINNERS: Okay.

DR. LEWIS: Because they are the ones who are going to make a decision.

MR. MINNERS: Ed here knows more about this than anybody I know.

MR. TOMLINSON: My name is Ed Tomlinson. I am with NRR and currently representing the tech spec branch.

I'm not sure whether I can answer that question, but perhaps I can explain what happens mechanically in a diesel when it does start.

When a diesel is at rest, the governor sees a maximum differential signal. And as soon as you crank that engine that governor builds up an oil supply pressure and drives the fuel racks to full open. The diesel gets maximum fuel at that point in time.

The purpose of that is similar to the choke in a gasoline engine. It's to provide an over-rich mixture to ensure starting.

Once it fires, the engine accelerates under maximum fuel, and there is no way to control this. It is a mechanical feature of the engine that cannot be changed. As soon as the governor picks up, it will bring the engine back, reduce the fuel rack setting and settle the engine speed out at synchronous.

In terms of this fast cold start, the real concern here is the loading of the engine, not the starting of it. The starting is something we can't control. To minimize the effect of starting, we are proposing to follow industry practice and all planned starts with diesels. That is, namely, to prelube the engine for a period of time, three to five minutes prior to starting, to ensure there is adequate lubrication throughout.

So the fast start per se is something of a misnomer. It's a mechanical feature of the engine that we can't get away from.

The fast loading feature, that's a different story. That is the most detrimental thing that you can do to the engine is to load it rapidly, because you then get a tremendous thermal shock to the engine which can create all manner of problems. This has been reduced.

DR. SHEWMON: So the answer is yes.

DR. LEWIS: I'm trying to -- you know, I appreciate the lecture about how diesels work, some of which I already knew.

But is the answer then yes?

MR. TOMLINSON: Repeat the question, sir.

DR. LEWIS: The question is: Is there a consensus that the fast loading of the diesel is detrimental, which you just said, and that therefore after these reports are analyzed that we've been told there are crucial to a decision, that the burden of proof will be on those who want to persist in the testing procedures that have been used up to now, which include fast loading, rather than the other way around?

I have the impression from the speaker that the presumption is in favor of the status quo, and that the review of all the reports that are available will be directed toward the question of whether one should change the status quo. I'm trying to change the bias a little bit.

Can anyone speak for the staff on that? Maybe that's the problem we have here.

MR. SERKIZ: I'll speak to it because we've had that discussion with the staff, and perhaps I have not stated it as clearly.

I will answer yes to the question that has been phrased several ways around the table. The staff will revisit this, and it has to do with the loading time.

DR. LEWIS: Yes.

MR. SERKIZ: And if the loading time is determined -- if the staff determines that the reports indeed are correctly represented technically and so forth, then that 10-second load time will be revised, will be revisited through the tech specs, because it is the tech specs now that make people adhere to the 10-second.

MR. TOMLINSON: I would like to add something else here. The 10 seconds, there is no requirement to load in 10 seconds. The requirement is to load within 60 seconds once you are synchronized with the grid. It is not to load within 10 seconds.

The only test that is required that actually will fast load the diesel is the 18-month surveillance test where you actually fail at off-site power and have the diesel go through its sequence.

So there is no 10-second loading requirement except for that 18-month surveillance test.

MR. CARROLL: The 10 seconds typically is the tech spec requirement on the start time to parallel?

MR. TOMLINSON: No. The 10 seconds is the time to come up to voltage.

MR. CARROLL: Okay, you haven't paralleled yet.

MR. TOMLINSON: And then there is a parallel step, and that particular step is not defined in terms of time.

Now once you are parallel with the grid, then the requirement is to load within 60 seconds.

MR. CARROLL: Okay. But back to the 10-second issue.

A lot of maintenance manpower is expended on keeping the diesel tuned so that it will accomplish that; is that not true in most of the plants?

MR. TOMLINSON: Yes, that is true. But there is something you should understand about that.

If the start time of the diesel is increased, it means something has happened to it because that 10-second start time is driven more by the mechanics of the engine than by any regulatory guide or tech spec requirement.

MR. CARROLL: Well, and the governor.

MR. TOMLINSON: And the governor.

If something changes and the start time increases, it's an indication that there is something happening to the engine. And the think the maintenance is driven more by that factor than it is by the requirement to meet a 10-second start requirement.

DR. KERR: Yes, but there is something that is happening is only something that is required to get this 10-second start. And if you never need a 10-second start --

MR. TOMLINSON: No, sir, that is not correct.

What is happening is you are seeking a change in the trend of the performance of that engine. And when you see a change in the trend on the engine performance, you have to do something.

MR. CARROLL: But there are cases, won't you agree, where 10 seconds is really pushing the diesel? And if it's just absolutely beautifully tuned, you can make it and it's 10.3 seconds if it isn't or something like that. It's just very minor things in the governor that are impacting that.

MR. TOMLINSON: There shouldn't be a great deal of deviation from one start to the next. We're not talking about any engines that have a great deal of time on them, you know.

MR. CARROLL: Do you know that? Because I guess I have some personal experience where we were forever fighting the 10 seconds. Nothing wrong with the engine. It was just minor little adjustments needed to the governor to --

MR. TOMLINSON: If we're talking in terms of a couple of a tenths of a second --

MR. CARROLL: That's what I --

MR. TOMLINSON: -- I would agree with you, sir.

MR. CARROLL: Yes, that's what I'm talking about.

MR. TOMLINSON: It will usually vary from one time to the next, and I don't really --

MR. CARROLL: And I'm talking about a diesel that on the average would make 10 seconds plus or minus a couple, and it was a pain. We did make some modifications to the governor. We ultimately made it 9.8 plus or minus two-tenths of a second.

MR. TOMLINSON: But once you've made the adjustments to the governor, the response time is changed too.

MR. CARROLL: Yes, yes.

MR. TOMLINSON: This is the point I'm trying to make. The response time should be fairly consistent. And if it changes, I'm not talking about the length of time. I'm talking about the trending of it.

MR. CARROLL: Yes.

MR. TOMLINSON: And if it changes from one time to the next, then it's an indication that something has happened to the engine.

MR. CARROLL: Some kind of a plus or minus.

MR. MICHELSON: Could I just make sure I understand?

I was a little surprised, I didn't know enough, I guess, to realize that the fast start of the engine wasn't the problem. I thought that was part of the problem.

But apparently, as I understand it, the fast start is not what damages the engine. It's the fast loading, if I believe what I hear.

MR. TOMLINSON: Yes.

MR. CARROLL: Or no prelube.

MR. MICHELSON: Well, prelube is a different issue, and that can be easily arranged.

In the accident case, and that was going to lead to my question. The accident case where you don't have time to prelube because you don't know the accident is happening, how much is that going to effect the reliability of start if on the one occasion you did not prelube?

Do we have any feel for that?

MR. TOMLINSON: We don't have any good quantitative data on that. We do know that failure to prelube is in fact detrimental to the point that it can cause catastrophic failure.

MR. MICHELSON: The first time you failed --

MR. TOMLINSON: Not the first time.

MR. MICHELSON: Oh, okay. Because it is going to be only one -- yes, it is only going to be one time we don't prelube. That's the time we've got a demand signal or a real demand.

MR. TOMLINSON: Two of the major suppliers of diesel generators to the nuclear industry have already taken the position that their engines should be inspected for damage after a predetermined number of non-prelube starts.

MR. CARROLL: Is that number like two or like 100?

MR. TOMLINSON: No, it's somewhere in the range of I think about 10. I would have to check on that to be sure.

MR. MICHELSON: So if the loading is the real problem, then does this reg. guide suggest that we change the loading timing and possibly the sequence?

MR. TOMLINSON: Well, the reg. guide does allow for slow loading of the unit except once every six months.

MR. MICHELSON: Now, how slow does the loading have to be to be nondamaging since we do have to load all these big motors, and there's not much you can do about that, but you can wait longer times between the loadings.

MR. TOMLINSON: Well, the typical vendor recommendation would be -- Fairbanks Morris. They recommend increments of 25 percent of the full load at 10 to 15-minute intervals.

MR. MICHELSON: Oh, minute increments?

MR. TOMLINSON: No. Ten to 15-minute increments.

MR. MICHELSON: Yes, okay. In other words, really stretch it out.

MR. TOMLINSON: Yes, you can stretch the loading out --

DR. CATTON: Thirty minutes.

MR. TOMLINSON: -- upwards of a half an hour or more.

DR. KERR: For the typical loss of off-site power, though, you don't necessarily have to have a electric power immediately.

MR. MICHELSON: No. I'm not thinking just for --

DR. KERR: It's only if you have a loss of off-site power and simultaneously some bizarre transient.

MR. MICHELSON: No, our bizarre transient causes the loss of off-site power. That's the one I'm worried about.

DR. KERR: Yes.

MR. MICHELSON: And there is a probability of that transient then, and some of those get interesting, I think, within 30 minutes.

MR. WYLIE: Of course, they are not relaxing the sequencing of those --

MR. MICHELSON: No. I was just trying to figure out what could be done. And apparently loading the way the vendors would like to load them to minimize the problem would he very long loading cycle, if I understand correctly.

MR. TOMLINSON: If I may add to this.

The occasional fast loading of the engine is not going to be that detrimental to it. There is no hard quantitative data that tells us what happens in terms of reliability is a function of fast loading, primarily because the nuclear industry in this particular area is just too new and we don't have a good data base.

But there has been enough experience out here to show that these engines can be fast loaded and they do stand up. It's just a kind of common sense thing that says let's not do anything that we don't really have to.

And the fact that you may have to fast load in the loss of off-site power even or some other plant transient that in and of itself is not going to create a problem.

MR. MICHELSON: Now, there's another aspect of fast loading that now becomes troublesome now that I understand the problem better, and that is, it's entirely possible that the loss of off-site power occurs after the loading of the diesel has already happened and you are into your event. And then suddenly off-site power goes. Because the off-site power doesn't go at time zero. It could take a minute or two or three.

And once you have to reload the diesels, now the sequence starts to change and the demand to get it back on again starts to change, depending on just when that loss of off-site power occurred. And it could be considerably less than 30 minutes, depending on where in a given scenario you say you lose the off-site power.

MR. WYLIE: I hate to cut this off, all of this, but we only have an hour for this subject and we've already used --

DR. REMICK: Thirty-five minutes.

MR. WYLIE: -- thirty-five minutes and we've got to hear from the staff and NUMARC, so we've got to move along. Maybe NUMARC will make a comment on this subject when they get up. Suppose we move along.

MR. SERKIZ: To just sort of come back one time and come out of the item 4, the accelerated testing, as it relates to current reg. guides out there and tech specs, this regulatory guide sets up the position that you should have in place a reliability program. And within the framework of that reliability program, you should look at the successes versus failures on a continual basis.

And in doing so, then, and I'm going to for the sake of time, to go to a chart which you have here, which is about two or three pages from the end of the handout, and it has three columns. And the intent of that is to focus on -- there is a normal action state, that you will continue the surveillance and you will repair failures, et cetera.

But within the framework of the reliability program as in place at the plant, at some point you will get into a mild action state, and that is shown on, and I'll speak to those in a minute, on a chart which is this one, and I'll come back to it, at which time you would undertake a closer look at the underlying causes of failure, seeing if there are patterns, no patterns, et cetera.

Under a strong action state, and we used the term "alert" with the subcommittee, and the subcommittee suggested changing the term "alert" to "action". The reason for that, alert carries other connotations. And it's really intended as identifying a need for sustaining the action that's normally being done under the reliability program, indicating a need for more attention for mild action than strong action.

But coming back to the principal thrust of the reliability program and the principal change really to Regulatory Guide 1.9, we need to keep track of what the apparent reliability state is the diesels for the nuclear unit as a whole. This will be done on a monthly surveillance basis which is surveillance testing, which would be tied into the monitoring of the reliability for the unit. And, of course, it would utilize the targets selected for station blackout.

I would like to draw your attention principally to looking at a simplified count approach, immediately up front saying that it is not the intent to drive anything into an exacerbated state by looking at a 20 or 25 demand sample. And that in the regulatory guide, you have much more detail on how we would combine the respective success/failure starts and a number of diesels on site.

Now, an item that has come up in our discussions with the NUMARC people is that if you have a problem diesel, an individual, I'll call it a problem diesel, that despite all the good things that the reliability program does or can do continues to experience the succession of failure. And the number we focus on is a three out of 20 on a single diesel. Then that would be sufficient cause, from the viewpoint of that machine, that particular EDG, to go into a mild action state.

Now, the reason for looking at the last 20 or 25 demands, it is the last 20 or 25 tests on that machine which give you the most current or relevant data as to the health of that EDG.

We are keeping track on a nuclear unit basis of what's happening in the last 50 and 100, and it is principally on the 50 and 100 demand or test basis that one determines the reliability of a nuclear unit and compares this with as station blackout.

However, if the scenario goes along the line that everything should have been going find and all of a sudden a particular machine comes up and triggers up on a three out of 20, I think you've got to look at it in a little more detail rather than, well, we'll take a look at it, make a repair and we think it's corrected.

If that same machine continues to come up and experience a fourth failure, so you have a four out of 25, we feel at this point that one has to take a very thorough look at the underlying causes, including potentially such things as FEMA analysis, good root cause analysis and so on.

And rather than coming out of it simply by revising what you are doing in a reliability program, it's to demonstrate the effectiveness of the actions taken. And the reason I'm focusing on that is the question has come up on accelerated testing.

Well, verification testing to demonstrate the effectiveness of the actions taken, the corrective actions taken, is outlined in reg. position C.3.3, C.2.33, and that involves seven consecutive failure-free tests. Okay, and those can be conducted at weekly intervals, or at a faster frequency, but not less than every 48 hours. That is a method that the staff feels is a way of demonstrating that the corrective actions have been effective on a near-term basis, not on a long-term basis.

Now, the question comes up in the mind of the staff that the machines just continues to experience failures. You've done everything that you thought should be done. You've gone into a strong action state. You've done root cause analysis. And you come up with a fifth failure.

Well, on that simplistic approach, okay, it is the staff's position that the EDG should be declared inoperable and a determination made of the level of overhaul required. At some point, the staff feels, you have to say enough is enough.

Now since we are discussing or it has been brought up earlier that there are differences of opinion with the NUMARC, let me use this slide to clarify where there has been at least agreement on how to handle some of these concerns.

The staff does not have a problem in moving the individual or problem EDG out as a separate element in a regulatory guide, and indeed we will do it that way.

The NUMARC staff felt, by maintaining that within the umbrella in the totality of the reliability program, that it could just continue to give confusion in the field.

Okay, and that's a valid point.

The intent is to focus on a nuclear unit for all EDGs, and if indeed we do have a problem diesel come into being, to deal with that problem diesel.

There is a difference of opinion between the staff and the NUMARC, at least as of several days ago because we have been meeting with NUMARC to try to get language and so forth cleaned up, declaring the diesel inoperative. And I think I will let NUMARC speak to that difference.

The regulatory guide that you have has the problem or individual diesel embedded in the overall program. The regulatory guide will be modified to pull this out as a separate element, but the intent will be the same. It will be to focus on what is happening to a bad acting machine or a diesel, an EDG that is starting to reflect a continued higher number of failures in the most recent tests.

As indicated in several other slides and in comments, we have worked closely with the NUMARC people. This, which is a table out of your regulatory guide, gives you a road map. There are sections in a regulatory guide that are not reflected in a NUMARC document for very evident reasons.

The NUMARC document does not deal with design considerations. The regulatory guide, Revision 2, and previous revisions dealt with design considerations. Those are retained in the current version.

The NUMARC document, for example, does deal with definitions, and this is the area that we are trying to get as exact a complement -- I beg your pardon.

The definitions exist in both documents and I think we are down to three sentences with respect to the question of exactness. So I would say we are 99 percent there.

The test descriptions are in a regulatory guide. The test descriptions are not repeated or reflected in the NUMARC document. This, again, the test descriptions on pre-operational testing, 12-month testing, et cetera, have to be carried in a regulatory guide because they have been as part of the licensing requirements and tech spec requirements. The same goes for this.

The reliability goals are noted in both documents. We've discussed differences of opinion on design basis accident.

Recordkeeping guidance and reporting guidance is contained in a regulatory guide. There is referral made to it in the NUMARC Document D. What we are asking for in recordkeeping is to keep records available on site that are consistent with the INPO information that is recorded and maintained. Reporting criteria, the reporting criteria would depend on whether you were into a mild or a strong action state. We feel the strong action state will be driven principally by the problem diesel, should it occur. And you will hear other views expressed that it will never occur. The industry record is that -- the overall industry record is that 98 percent overall industry reliability.

The staff feels that dealing with the problem diesel and reporting such occurrence as outlined in the regulatory guide is consistent for complete closure on it.

The principal change to the regulatory guide deals with the reliability program. The reliability targets are set by station blackout. You need surveillance, monitoring, maintenance. These are all, in terms of elements, consistent with what is being done at the better plants from the viewpoint that they have implemented these.

There is a problem close out, a data capture utilization and assigned responsibilities and management oversight.

I would like to stop here for two reasons. Time is going on. Give NUMARC some time. I would like to come back and talk to the committee about implementation if that's all right with the committee.

MR. WYLIE: I will call on NUMARC then.

MR. MARIAN: Thank you very much, Mr. Wylie.

I would like to take a couple moments and just elaborate on a few points that were made this morning, and they will be in my --

MR. WYLIE: I don't believe we can hear you.

MR. MARIAN: Can you hear me now?

MR. WYLIE: Yes.

MR. MARIAN: Okay. NUMARC has been working with the staff for the last year and a half to reach a consensus and develop complementary documents to resolve the B-56 issue. We have focused our efforts through the NUMARC station blackout working group that developed the resolution of the station blackout effort.

The working group relied on the B-56 task force that Al Serkiz alluded to, to develop an Appendix D document and coordinate comments on draft Regulatory Guide 1.9, Revision 3.

The task force was comprised of representatives from EPRI, INPO, and utilities representing the diesel generators that existed in the nuclear plants today, including TDI, DeLaval, Cooper Enterprises or whatever they are called today. I believe it's called Enterprise at this point in time.

The purpose of this total effort was to develop a diesel generator reliability program that offers a means of maintaining the diesel generator target reliabilities chosen for station blackout. Those reliabilities are .95 or .975.

Secondly, the purpose was to provide a basis for resolution of the B-56 issue. Our revised Appendix D document has been reviewed by the staff and comments have been submitted. We are currently working towards obtaining staff endorsement of our Appendix D document.

Our Appendix D document was developed on proven successful methods that are responsible for the present high diesel generator reliability. The INPO plant performance indicator program data and the EPRI data that was published in INSAC 108 indicates that diesel generator reliability has been greater than .98 since 1983.

And I would like to, for your benefit, okay, you've just received a copy of a couple graphs. Let me just go through those.

The first one is the summary of the unreliability average on an industry-wide basis as published in the INSAC document.

The second is the unreliability averages published by the INPO data covering the periods of 1986 through 1988.

The third is a composite of all that data for all the years from '83 to 1988. And on the right-hand side, I apologize if you can't see it, there is a line drawn that indicates the .975 station blackout reliability target and the .95 reliability target.

Now, keep in mind the bar charts are based on unreliability.

The average reliability that was established by the INPO and EPRI data is .986.

Recognizing that the intent of the B-56 issue was to increase reliability to .95, we believe that our revised Appendix D document, coupled with the acknowledged industry performance, offer sufficient basis for closure of the B-56 issue.

As part of the resolution of the station blackout issue industry, through the NUMARC station blackout working group, developed the NUMARC 8700 document that contained the initial version of Appendix D, which essentially included five elements of a program that was being considered for maintaining the station blackout diesel generator target reliabilities.

The Appendix D document strictly focused on reliability, and as such did not address accelerated testing. We believe the concept of accelerated testing is fundamentally inconsistent with the reliability program that focuses on addressing both actual and potential failures.

The concept of accelerated testing was originally structured to provide assurance to the staff to accumulate test data to place a judgment on reliability.

In the early 1970s, when the B-56 issue was established, diesel reliability was estimated from LER reviews to be on the order of .93. The intent of the B-56 issue was to improve reliability to .95.

We acknowledge the fact that this was an appropriate consideration for that point in time. But, gentlemen, current industry-wide average reliability is .98, and it has been since 1983. The program delineated in Appendix D is structured to maintain this high level of reliability, consistent with the NUMARC 8700 document and the station blackout rule.

However, in sustaining the commitment to attain complementary documents, we agreed to consider a reduced form of accelerated testing in our document. It is not presented as reliability. Rather, it is structured as an action to be taken following the performance of an individual diesel generator.

This compromise from a reliability focus was offered to address what we perceived to be the staff's fundamental concern of a problem diesel generator exhibiting four failures during the most recent 25 demands.

As Al Serkiz correctly indicated, we are currently working with the staff towards a consensus resolution. Nevertheless, we firmly believe action in the form of accelerated testing is unnecessary and contrary to a reliability-based program. We are interested in your thoughts on these fundamental differences.

We believe a regulatory guide strictly focused on a reliability program should be issued to be complementary with industry's Appendix D document. This is important to assure clarity and a proper focus for station personnel who may implement this guidance.

The current structure of Reg. Guide 1.9 addresses endorsement of an IEEE standard that provides design testing and qualification requirements for diesel generators under design basis accident conditions.

Gentlemen, the station blackout event is not a design basis accident per the station blackout rule.

We believe that this will add additional confusion on the part of the people at the plants. We believe the reg. guide is confusing in that it mixes the IEEE standard with reliability program. The IEEE standard deals with design qualification and testing.

Furthermore, the diesel --

MR. CARROLL: Testing meaning preoperational testing?

MR. MARIAN: Yes. Preoperational and qualification testing.

MR. CARROLL: But not surveillance testing?

MR. MARIAN: That's true.

The reliability program that we've structured is in support of the station blackout rule, which is directly coupled to regulation where the IEEE standard is not and should not be coupled to any form of regulation.

Additionally, gentlemen, the 1984 version of the IEEE standard is currently under revision by the Nuclear Power Engineering Committee of IEEE.

As Al Serkiz correctly indicated, there are several open items that are still open to discussion, and I would like to touch on these because they will answer some of the specific questions and concerns that were raised earlier.

The industry-wide INPO plant performance indicator program definitions currently used by utilities should be followed to minimize confusion. In 1986, the NUMARC Executive Committee approved an initiative to address AC power availability. And this initiative called for each utility to monitor emergency AC power unavailability utilizing data provided to INPO on a regular basis.

We are currently awaiting resolution of three definitions dealing with start failures, load run failures and exceptions.

In Generic Letter 84-15, the staff accepted diesel generator reliability and the availability data as a basis for the reduction and the frequency of cold fast starts. We believe the reliability and unavailability data collected since 1983 supports further reduction.

The current regulatory guide calls for a fast start from standby conditions once every six months. This is contrary to another industry initiative that calls for each utility to reduce or eliminate cold fast starts of emergency diesel generators through changes in technical specifications or other appropriate means.

Our position, gentlemen, is that we believe the frequency for fast start tests should be on an 18-month interval. Plants which are approaching a 24-month refueling cycle through license amendments should be handled on a plant-specific basis.

We do not see any basis for the fast start test on a six-month interval. An analysis that was conducted by General Electric indicates that for certain BWRs, that 118 seconds is an acceptable time period. Another analysis conducted by Westinghouse indicates that 53 seconds is an appropriate period.

MR. CARROLL: Those number of seconds are loading time or total time for push button?

MR. MARIAN: They are start and be ready to load. I'm referring to a draft report that was developed in the NRC's N-PAR program that refers to those tests. And let me just check this a second, if I may.

Strictly focused on starting time.

MR. WYLIE: What was 118 seconds?

MR. MARIAN: Okay. I will read from the report.

GE study indicating a start time could be increased to 118 seconds in a typical BWR-6, and there is some additional clarifications.

DR. KERR: What does that time refer to, from something to something?

MR. MARIAN: Push the button up, up the rated speed and voltage and ready to accept load.

MR. WYLIE: First load.

DR. KERR: Thank you.

MR. MARIAN: The current regulatory guide identifies three separate tests for simulating loss of off-site power, safety injection, auto start and a combined loss of off-site power in conjunction with the safety injection auto start. These involve fast starts, gentlemen. These three fast starts are unnecessary because verification of off-site power and safety injection auto start signals can be verified up to the point of starting the machine without actually starting the machine. And this is done by verifying that you've got the appropriate signal characterized through your instrumentation.

We recommend that decreasing the number and frequency of fast starts consistent with industry initiatives and the manufacturers' recommendations to minimize wear and tear of diesels and improve reliability. We believe the combined test demonstrates a most conservative conditions in terms of load demand.

We recommend that those tests be separated and that utilities should be allowed to use the combined loss of off-site power and size test in lieu of the independent. We think that this will effectively reduce fast start stress and wear consistent with the Generic Letter 84-15, as well as the intent of the station blackout rule.

MR. MICHELSON: Excuse me. When you talk about fast starts now, do you mean just the starting of the engine or fast loading as well?

MR. MARIAN: Starting and sequence loading. The current regulatory guide, as Al Serkiz indicated, offers a statement regarding diesel generator inoperability.

We believe that this is inappropriate for a regulatory guide and expect this to be addressed in the standard tech spec arena. NUMARC has a separate working group addressing the technical specification improvement program.

We further believe the descriptive nature of determining a level of overhaul for the problem diesel is totally inappropriate.

This idea of identifying a problem diesel, taking action in the form of accelerated testing, addressing inoperability and suggesting a consideration for overhaul, is detrimental to reliability, gentlemen, because these actions are being structured in a prescriptive fashion to be totally independent of the failures on that diesel generator.

MR. CARROLL: The tactical word, or phraseology, is major overhaul, is it not?

MR. MARIAN: I believe, while we're still working on that, I believe the word "major" was removed.

There is one more point.

The current regulatory guide in position 4 identifies recordkeeping that we believe does not directly relate to start and load run data necessarily to support investigations or assessments of diesel reliability.

The regulatory guide would requests specific out of service time that we believe is beyond the scope of reliability data.

This has the potential for further misuse of industrywide performance indicators.

And I thank you for the time to talk to you. Do you have any questions?

Yes, sir.

DR. SHEWMON: I have a question. I guess it is for both you and the staff.

I am surprised at how low these unreliability numbers are in that I had thought they were up closer to .95, or I'm sorry, .05.

MR. MARIAN: Yes.

DR. SHEWMON: Is there any difference in the definition of this than what I am likely to have bumped into in PRA data or in listening to the staff? Or is it just my lack of being up to date on things?

You are allowed to say yes to that.

MR. MARIAN: The definitions of the type of data that is collected is established in the INSAC document and is also established in the INPO plant performance indicator rules. And they are consistent.

I believe that data has been misinterpreted by people.

And you know, the INSAC document has been accepted by the staff as an appropriate assessment of diesel performance for that time period.

DR. KERR: The unreliabilities as you characterized them at least were average I take it across the industry?

MR. MARIAN: Yes. Based upon actual --

DR. KERR: Do you have any idea what plant might have a worse unreliability than that and how bad it might be?

MR. MARIAN: Not without reviewing the data. There is a table in this document that summarizes the data submitted for each plant and their diesel generators.

I would have to go through that table to answer your question.

MR. WYLIE: Could you supply that to whoever in the committee would like a copy of that?

MR. MARIAN: Yes. We can do that. Okay. We will make that available.

MR. CARROLL: Do you know if all three plants are down near .95?

MR. MARIAN: No. But let me offer something out of the INPO data.

The INPO data that has been collected shows a cycle of activity where a plant may fall below the .05 value for a period of about a year or so.

Now, the following year, as data is collected and assessed, that utility will show marked improvement and be above that value.

Now, that indicates that the utilities are assessing those failures and responding to them accordingly with appropriate corrective action. And over a period of two years, there is a stability of activity in terms of diesel generators that shows it is above .05.

MR. SERKIZ: I think the point has been made, Mr. Chairman, that there are indeed plants that do slide down into an unavailability level of .05.

I have seen that in other data that has been presented. I think from that viewpoint, and consideration being given that we don't have a two-year cycle on station blackout, is the reason the regulatory guide is structured as a reliability program that follows things on a monthly basis.

MR. WYLIE: Mr. Chairman, how much more time can we take? Do we have to wrap up?

DR. REMICK: I would like to limit it to five minutes if we possibly can. Mr. Serkiz said he did want to come back on instrumentation.

Is there any way of making that very short?

MR. SERKIZ: Do you care to hear Mr. Faust's view on the 3 out or 20 versus 4 out of 25?

DR. REMICK: What is the pleasure of the committee?

DR. KERR: Can he do it in two or three minutes?

MR. ROSA: I can do it in two or three minutes.

DR. REMICK: All right. Please.

(Slides being shown)

MR. ROSA: Thank you for affording the two or three minutes.

(Laughter)

MR. ROSA: It is the staff's view that the level of reliability attained by the industry, no matter how high, is irrelevant to the need for a criteria to determine whether in fact a diesel generator reliability has degraded, and to do this in a timely manner.

Diesels wear. And their reliability will be reduced thereby. There are differences in nuclear power plant expertise, and the programs for maintaining reliability. And this is not a constant.

Right now, the figures seem to indicate that generally this level of expertise and reliability attained is high. I don't know that that will continue for the rest of the century.

The difference between the version of the guide that you have and our view is this. At this point here, the guide goes into a mild action state, and as in Figure 1, the column on the, in the middle.

We would, at this point, declare that diesel a problem diesel and require that seven consecutive failure-free tests be performed on a weekly basis.

And this is to simply provide assurance that the corrective actions taken as a result of these three failures have been properly done and that no other problems have arisen.

Now, that is the basis for our position.

Now, in addition to that, in addition to that of course, if during the course of the corrective action testing additional failures occur, then we would require that if five out of 20 failures occurred, then you would go into the 14-day test sequence before restoring the diesel generator to an operable status.

Also, at this point here, you have to maintain a 3 out of 20 or less failure rate in order to -- well, let me put it differently.

If you have 3 out of 20 you go into the seven consecutive failure free tests. If in the course of that testing after you have succeeded with those seven tests you have another failure such that you again have 3 out of 20, you repeat the cycle. And so on.

That, I think, probably is the gist of our position.

We have prepared writeups of the relevant paragraphs, three of them. Well, two paragraphs and a revision of Figure 1, which the staff can pass out to you, for your perusal at leisure.

And that essentially completes my presentation.

I would like to make one more point, though, before completing this.

Going back to cold fast starts. The position in the guide does not require loading for that six-month fast start. It only requires that the time required to reach rated voltage and frequency is within the specified time, and continued operation for five minutes, unloaded.

So that I hope would put that issue to rest.

DR. REMICK: Dr. Shewmon had a question.

DR. SHEWMON: Yes. I was going to ask whether these failures were, whether a 10.5 second start would be a failure or if they were something really serious. You have answered the question.

MR. ROSA: If you will look at that section of the guide that identifies or defines exceptions.

DR. SHEWMON: Fine.

MR. ROSA: Those exceptions state that anything like operator error or some failure in the initiation system --

MR. CARROLL: Page 15 of tab 7.

MR. ROSA: -- are not counted as failures. There is a whole list of exceptions there.

DR. SHEWMON: You've answered my question. Thank you.

DR. REMICK: Any further questions?

(No response)

DR. REMICK: Anything else, Mr. Wylie, at this moment?

MR. MARIAN: May I just make a brief little statement, gentlemen? Thank you. Alex Marian.

I would just ask you gentlemen to review the EPRI INSAC data and review the data that we provided you that shows the accumulation with the INPO data which suggests that reliability has been on the order of .98 since 1983, and reach your own conclusion and recommendations.

And thank you.

MR. MICHELSON: Just one clarification from NUMARC.

MR. WYLIE: We have about one more minute.

MR. MICHELSON: Yes. Do you agree then that the fast start alone of the diesel is not the objection but rather the fast loading?

MR. MARIAN: No, sir. We do not agree with that.

MR. MICHELSON: Well, I haven't heard your argument anywhere that has contradicted the rather strong position, and the confusion I had is which is the problem or where is the problem. And you are saying it is in the fast start as well?

MR. MARIAN: This may take about a minute and a half.

Let me indicate that it is our belief that the fast starting and loading, inconsistent with the manufacturer's recommendations, increases stress and wear on the diesel generators.

There is current research data that is being developed out of the NRC's nuclear plant aging research program that offers a detailed discussion on the type of stressors that come into play during fast starting and the loading period.

MR. MICHELSON: My question was --

MR. CARROLL: -- respond to what we heard earlier, it is an inherent characteristic of the diesel. Once you push the button, ten seconds.

MR. MARIAN: I am not that familiar with the mechanics of the diesel generator and the fuel injection systems but I do know that utilities have been using diesel generators for black start cap[ability at fossil plants and they do not bring them up to speed and voltage in ten seconds.

MR. MICHELSON: You didn't quite answer my question. My question was just the fast starting alone.

You answered it, that you said fast starting and loading was a problem.

MR. MARIAN: Yes, sir.

MR. MICHELSON: Now how about fast starting alone?

MR. MARIAN: Both, yes. Independently, yes.

MR. MICHELSON: Okay. Independently they are both a problem?

MR. MARIAN: Yes.

MR. MICHELSON: Okay.

MR. MARIAN: That is our position.

DR. REMICK: A final minute for the staff.

MR. SERKIZ: I would like to just make the point that what is shown in this reg guide and the underlying basis for this regulatory guide was to put into a regulatory guide those proven practices that have been also given back to us from industry as such, as through NUMARC, EPRI, et cetera.

The staff does not feel that the regulatory guide is writing new language for something that is being well out there. On the contrary.

The regulatory guide reliability portion of the program and reliability monitoring is indeed consistent with the NUMARC approach.

I would like to comment on a couple of things. I think the INPO definitions are very important. And just to clear the record, we are using identical language except for two or three sentences in Section 2.1.

And it gets down to beauty in the eye of the beholder. And we have to work that wrinkle out.

On the six-month testing, there has been a discussion, and I will go into that. There is a difference of opinion. But you heard Ed Tomlinson give you a mechanic's view, if you will, on how a diesel operates.

The staff right now, for the reasons I cited that it is has not completed its review of some of these reports, is sustaining leaving a six-month test in there.

I would like to clarify, because there has been, Alex Marian made a point about combining some tests.

I would draw your attention to Table 2 in the regulatory guide, which is at the end.

MR. CARROLL: I'll find it for him.

MR. SERKIZ: But what Alex is referring to there are tests that are run --

MR. CARROLL: 32.

MR. SERKIZ: -- are in the pre-operational and 18-month surveillance testing. So this is not a problem that is occurring every month or every six months. And there is a difference of view between the staff and NUMARC and we are still trying to get that situation finalized.

The point I want to make is we are talking about tests, the combining of two types of tests into one, in a pre-operational state or 18-month testing state.

I'd like to make this point on recordkeeping which is in Section C-4 of the regulatory guide.

We have three paragraphs. Paragraphs 2 and 3 of that section contain within the regulatory guide that type of information that indeed INPO uses to build their data base and come up with what they call availability or unavailability.

So the staff is not asking for anything new. The staff is not asking for something to be sent to headquarters. The staff is simply putting in a regulatory guide the same information that we think INPO is using. And I have to say "think" because I have never seen INPO's complete document.

DR. KERR: Yes. But now are you requiring it be kept in a way which probably requires it be around for ten years and that it be inspectable by NRC inspectors. And while to you that may not seem like a burden, it can be.

I mean, maybe it is necessary.

MR. SERKIZ: Well, if you don't have a definition of information that is normally maintained and consistent with an INPO view or an industry view, and we have a situation of a diesel that is degraded --

DR. KERR: You may need the data for some purpose. But I don't think it is valid to assume that this does not put an added burden on a licensee. It does.

MR. MINNERS: We agree with you. We're just saying the data collection is not any different. The storage may be different.

DR. KERR: Is it the view of the staff that the current diesel reliability is unacceptable?

Or is this reg guide simply an effort to maintain it at the level which it now is?

MR. MINNERS: It is the latter.

DR. KERR: So there is no particular reason to assume that it needs improving. But there is a feeling that without the regulatory guide it would deteriorate?

MR. MINNERS: I think on some individual plants yes, we think it may need improving, at different times.

DR. REMICK: Gentlemen, I have to cut off the discussion at this time. We have already overrun.

I thank very much the staff for coming down. I thank the NUMARC representatives. I think it has been extremely informative. I'm sorry we did not schedule more time for the discussion. But we must move on, because we have other important agenda items.

So thank you again.

Let's move, then, to the next topic, which is generic issue 87, failure of the HPCI steamline without isolation. And our Vice Chairman is Chairman of that Subcommittee.

And so, Carlysle, I turn the meeting over to you.

MR. MICHELSON: Thank you, Mr. Chairman.

We are going to shift the subjects slightly from that shown in the agenda.

And what I would like to do first of all is give you a report of a subcommittee meeting which was held on October 3 that discussed this item as well as two or three other items.

And it is necessary to have a little of the background of that subcommittee meeting to properly understand the position being taken on generic issue 87.

So in view of that, also, the subcommittee determined that it wasn't necessary to have staff presentations, since most of the meeting had to do with a better understanding of what we had in front of us, not new material necessarily.

But I will give you a subcommittee report that covers this and that should suffice.

The Subcommittee on Mechanical Components held a subcommittee meeting on October 3. Those attending were Catton, Carroll, Siess, Wylie, and myself. Our consultant was Peter Wold.

We discussed four subjects at this subcommittee meeting. The first subject was to obtain some clarifications of generic letter 89-10 which had been discussed with the committee earlier. You had complete briefings on it, and we wrote a letter on it. But there was some clarification needed for better understanding.

A second item that we discussed was the recent operating experience and test results of some flow interruption tests being done at the Karlstein facility in Germany.

A third item was a status report on the action plans for motor-operated and check valves.

And the final item was the discussion of generic letter 89-04, which was inservice testing.

At the conclusion of those items, we did discuss then the generic issue 87 and what position this subcommittee should take in terms of a letter.

On the clarification of generic letter 89-10, the committee wrote a letter on this subject on May 9, 1989 in which we essentially endorsed the generic letter with some comments.

And we transmitted these comments in our letter to the staff. I believe the staff has acceptably adopted all but one of the comments. And that was then the subject of further exploration to be sure they understood the comment and to make sure that we understood why they had rejected it.

The comment in particular was in item 2 in our letter which dealt with, it dealt with the problem of the design basis to be used in determining the settings for motor-operated valve switches.

What I was concerned about and wrote a memo on, which you had received a copy of, was the concern that if you find that you have a valve which meets the design basis in mind at the time the valve was --

DR. KERR: Carl, excuse me. Do we have the memo to which you refer?

MR. MICHELSON: Unfortunately, it was in last month's handout. I thought it was in this month's, too.

DR. KERR: That's okay. If we don't have it --

MR. MICHELSON: But I don't think it got in here. Is that right? The copy of the September 4 memo? Is it in there? Page 8. It was in last month. Yes. Here it is. Yes. It is on Page 8.

DR. KERR: Okay.

MR. MICHELSON: And that memo was just to make sure that the committee members as well sa the staff understood the implications of what the committee had said in their letter. And so I assume that all the members have taken a look at this. And what we wanted the staff to do was just to discuss it and get some clarification.

And to put it very simply, the problem is this. At the time you build a plant, you build it to the best criteria that you have in front of you at the time. You specify the components based on such criteria.

Now, as the plant proceeds in age and time, we get a little smarter and we think of other criteria that it probably should have been designed to and wasn't designed to.

Now, the staff views this change of criteria as a backfit if you have to change the components.

My concern was though, how should one view such criteria, such new criteria, if all you have to do is go back and readjust the old components as opposed to changing them? And this was the issue then we basically discussed with the staff, is this question of readjustment.

It turned out that there were a number of interesting problems that resulted, including when you find that the old criteria really aren't good in terms of today's thinking, is that even a reportable deficiency or how do you handle such findings? Do you report them, if at all, and do you accommodate them, if at all?

And when we wrote our letter, and I would like to read that Item 2 now, when we wrote out letter we said this:

Although no change in the existing plant design basis is intended by the generic letter, we recommend that each licensee be reminded to review the design basis governing the selection of each MOV from the viewpoint of completeness and adequacy in light of current regulatory requirements.

In the meantime, and to the extent possible, current requirements should be reflected in selecting MOV switch settings, in demonstrating operability.

Now, the idea was that if you find that your criteria that you should have used indicate that the switches should have been set differently, it would be, it would seem prudent to set them to today's criteria. And that is what we were trying to say in Item 2.

It turns out that that is not a requirement. It is not covered in the generic letter. The generic letter remains silent for such situations.

So it is up to the committee whether the -- The comment was rejected. It is up to the committee whether we pursue the matter.

The subcommittee decided that it probably bears looking into as a generic issue, a potential generic issue, because it has more implications than just the adjustment of valves.

It has implications on the adjustment of anything, as your understanding of its needs changes. And how are these adjustments made, do they have to be reported, to what extent can you require such adjustments be made, and so forth. It is probably generic issue that needs to be explored.

Now, the Mechanical Components Subcommittee Chairman volunteered to at least look into this and try to set up a meeting to discuss it.

Once we get it evolved a ways, I wonder whether that is the best subcommittee to handle it, since it is a much broader issue than just valves.

So that is where we are at on that.

DR. KERR: I had not seen this memo before. But I may have overlooked it in the mail somewhere. And I see it received much wider circulation than the members of the committee, apparently.

MR. MICHELSON: Yes. It was sent to the staff to provide the questions to be answered or discussed at the subcommittee meeting.

DR. KERR: Okay.

MR. MICHELSON: And so they wanted to know what do you want to talk about, what is your problem, and have tried to identify --

DR. KERR: I can't understand from the memo whether you are raising the issue of the MOVs or the broader issue of backfit.

MR. MICHELSON: It's the narrow issue of the MOVs that we're raising.

DR. KERR: But is it somehow a peculiar question outside of the usual backfit considerations in your view or is the whole backfit arena one that needs further exploration?

MR. MICHELSON: Well, at first I thought it might be unique to the adjustment of valves, but after hearing how the staff responded to what you do in that case in terms of regulatory reporting requirements and how you handle it and so forth, I'm not sure.

I think it is something that could be usefully discussed, to first of all, clear up how you do it for valves, and then to expand onto whether or not there are other applications of the same principles, whatever those principles might be.

DR. KERR: In your view, is it a valid interpretation to interpret what you had in mind as a backfit?

MR. MICHELSON: I would not view it that way. But I do recognize that it isn't just a matter of taking a screwdriver and walking up to the valve and changing the torque switch set.

First of all, I've got to do some calculations and be sure the higher thrust doesn't overstress the valve. I have to change a bunch of procedures. Clearly, it is not a trivial matter to adjust any equipment of that sort.

DR. CATTON: You may have to replace the valve.

MR. MICHELSON: Now, the replacement, see, there's two steps to the issue.

The first step is you've got the old component, can you bring it up to today's standards by adjustment only? That would be one case.

Or the second case would be can I bring it up to today's standards with some additional corrections to other flow parameters, other adjustments or whatever, can I make the thing work.

And then the third case is, no, clearly a replacement. The replacement no doubt is a backfit.

I'm wondering though when you do these nonobtrusive changes, in other words, software changes, people changes, analysis kind of changes, are those also backfits?

And I think some people could argue purely that yes, it is, if it is going to cost money and it is something that we didn't have in our original design.

DR. KERR: Generic letter 89-10 must have been designed to correct the deficiencies in MOVs.

MR. MICHELSON: No. No, It wasn't. It was intended to find out. It was intended to assure yourself that you had them properly adjusted to meet your original, or your approved design basis, whatever that is.

DR. KERR: Let's suppose that somehow one were convinced that this was a serious deficiency throughout the industry. And apparently it is.

Is it something that is likely to be dealt with in the IPE or do we have, I guess we have no way of knowing whether, because there it would show up if an appropriate risk and reliability analysis were formulated.

DR. CATTON: I don't think so.

DR. KERR: If one took into account the failures of MOVs that apparently the staff now believes to be the correct number, you would certainly get different values for core melt frequencies.

DR. CATTON: Yes, but a lot of these valves, that aspect is not going to be included in the IPE.

MR. MICHELSON: Not reflected in the PRA.

DR. CATTON: Because if you have an incorrect design basis for the valve, which is what is going to lead it to trouble, that won't come out in a PRA.

DR. KERR: Now wait a minute. Surely a PRA is based on what should occur and not what the design basis is.

PRAs don't assume double ended pipe breaks, for example.

DR. CATTON: No. But if the pipe is there for isolation in case a small pipe breaks, the PRA is going to assume it works. It's not going to go back and take a look and say gee, is this valve correct for this application. The PRA does not do that. So it would be missed.

DR. KERR: It would seem to me with all the current concern about MOVs that anybody who is doing a conscientious job on their plant would look at those things. But that may be naive for me to think that.

MR. MICHELSON: So generic letter 89-10 is the correct place to address this question. It requires you to go back and check your design basis for each valve and make sure you have it adjusted so that the valve operation will match the design basis requirement.

The question is of course what design basis. Well, it made it very clear, you use whatever your approved basis is, irrespective of whether it might be right or wrong in today's light.

DR. KERR: Is there some reason, I mean there apparently is good reason to believe on your part that earlier design bases were probably incorrect?

MR. MICHELSON: That is a good question. We asked the staff to give us a view on that and they were not real firm in their answer.

It would be very nice to say that yes, every one of these valves, like for instance the reactor water cleanup, those valves indeed were designed for full guillotine breaks downstream of the valve, for instance.

But it is not clear that these early FSARs described such events. And if it weren't described, if it were never committed to, then the valve doesn't have to do it.

DR. KERR: Would we want them to be designed for instantaneous guillotine breaks?

MR. MICHELSON: That is the present basis for doing your analysis of flooding outside of containment.

Now, if you want to say that the valves are not designed for that, then you have to do the flooding analysis accordingly.

DR. KERR: In terms of what we believe is likely to happen, is that a high enough probability event that not having done it is likely to cause serious consequences?

MR. MICHELSON: Well, yes, I think I understand where you are coming from, Bill. And I agree, yes, these are low probability events.

I am taking a little bit different philosophical approach.

I'm saying that if I already have a piece of existing hardware, it should be adjusted to do the job I think it needs to do today. To the extent that it can do that, it should be by adjustment of that old piece of hardware.

I'm not proposing replacing the hardware. I'm only proposing adjusting it to today's standards.

DR. KERR: No, but my question is, do we think that the probability of double-ended instantaneous guillotine pipe break is high enough likelihood one should design for it?

MR. MICHELSON: The way I look at that is I keep asking the PRA expert what is the probability of that break occurring. And they keep giving me numbers about 10 to the minus 4.

And I say okay, if the probability of the break is 10 to the minus 4 and if the probability of the valve to fail to close were 1, I think I'm in deep trouble because I cannot isolate the break and it gets interesting.

DR. KERR: When I have heard you ask questions about the pipe break, I have never heard you use the descriptive phrase double ended instantaneous guillotine.

You, I think, just asked for a probability of a pipe break.

MR. MICHELSON: And then when you ask the PRAs what break you mean, they mean all the way up to the full break.

Now, I've never seen PRAs grade according to the size of break, that the probability --

DR. KERR: I personally think the probability of a double-ended instantaneous guillotine pipe break is zero.

DR. CATTON: The "instantaneous" I'm not sure is relevant.

MR. MICHELSON: I will take the "instantaneous" back.

DR. KERR: But that has a significant influence on what happens, doesn't it?

DR. CATTON: You are talking about valves that have to close against full pressure. You can get the full pressure over a period of time. And it doesn't have to be instantaneous.

So the question really is will the valve close against full flow?

MR. MICHELSON: Full break flow. And then you have to define the break.

DR. CATTON: Yes. But it could break over a period of time. The instantaneous is not relevant.

MR. MICHELSON: Yes, the instantaneous is not essential.

DR. KERR: Yes, but you are going to ask it to close when the pipe originally starts breaking, aren't you?

Or are you going to wait until it is all the way --

DR. CATTON: There's a time delay.

DR. KERR: Okay.

MR. MICHELSON: 90 percent closed may not be good enough in terms of environmental effects, if you are unable to complete the closure.

If you can complete the closure, by some other means, then you are in better shape.

But I, as I say, my position would be that we ought to adjust these valves as best we can to meet today's understanding.

DR. KERR: For the recommendation on what we should do.

MR. MICHELSON: Yes. That was what we said in Item 2 of our letter was that yes, indeed, they ought to be designed, they ought to be adjusted to today's standards.

We didn't suggest they replace the equipment.

DR. CATTON: I think you might find today's standards are lesser than before.

MR. MICHELSON: If they are, then that is even easier.

Now, we asked the staff about this as to whether the utility can on their own even readjust these valves without reporting, they just do it.

It was unclear whether they reported. But I think it came across that they have the prerogative of adjusting these things to today's standards without making too big a fuss about it.

But it was unclear as to the reportability or how they handle it.

DR. KERR: On the basis of your meeting with the subcommittee and your discussion, is there anything that the subcommittee thinks the full committee ought to do further?

MR. MICHELSON: Yes. They suggested that we make this a generic item discussion and go back and explore it in subcommittee again as a generic item. Potential generic item.

MR. CARROLL: Separate from the proposed resolution?

MR. MICHELSON: Yes. And that is where we left it. And I think that is a reasonable way to handle it.

DR. KERR: What title would the generic item have?

MR. CARROLL: It would be a new generic item.

DR. KERR: We have something that is a proposed draft of a letter?

MR. MICHELSON: We haven't got to that yet. That's another issue.

That's not necessarily even related to what we are discussing right at the moment. We are discussing this clarification of 89-10 and whether the committee wants to handle any rebuttal to their Item 2 and I think the answer is we will see when we explore it as a generic item, proposed generic item.

DR. REMICK: You are proposing that at this point to the committee?

MR. MICHELSON: Yes.

DR. REMICK: That is be a possible generic item, and you would have the subcommittee explore that further?

MR. MICHELSON: Yes.

DR. REMICK: Any objection to that?

DR. KERR: Is that the consensus of the subcommittee?

MR. CARROLL: Yes.

DR. KERR: It seems reasonable to me.

MR. MICHELSON: We also felt that the Mechanical Components Subcommittee at least would start it and if it grows out of that we will come back to you to see where you want to put it.

Okay. The second item the committee heard about was we had a short briefing on the progress being made at Karlstein with valve testing.

The Karlstein facility in Germany is a much larger facility and capable of different flow conditions than the one at Wylie Laboratories, where the earlier tests that you have already been briefed on were performed.

The first tests at Karlstein were a repeat of the valve that had failed to function at Wylie. And on retest at Karlstein it still failed to function, for perhaps the same reasons, but the analysis really hasn't been done yet.

MR. CARROLL: I recall that particular valve was the one with the very wide clearances in it between the guides and the gate.

MR. MICHELSON: And it hung up on the edge of the face. And it seemed that these tests were better instrumented, better followed, and it still indicated strange things were happening during closure in terms of thrust requirements on the gate. It wasn't at just the point of closure even. It was in the whole process of closure they were seeing interesting things.

The next series of tests will be done with steam. And there were more water tests yet The other valve will be tested to see how it does.

MR. CARROLL: Well, it has been tested. That was the valve that worked successfully in the Wylie test and also worked successfully in this test.

And a third valve which had not previously been tested by a third manufacturer, which everybody predicted was not going to work because it is a much lighter valve in terms of design and so forth, performed beautifully, with no problems.

MR. MICHELSON: We don't have much time. And I didn't want to go into any more detail than to indicate we got the brief report on it.

The next item that we were briefed on is generic letter 89-04 which is inservice testing.

This letter was issued by the staff without ACRS review. It came to the committee after issuance. And the subcommittee was asked to look into it to see whether or not we thought a letter from the committee was worthy, or was needed for any reason.

The subcommittee examined generic letter 89-04 and determined that a letter would not be needed.

MR. CARROLL: On the issue of our not getting a chance to look at it before it came out, presumably that has been dealt with between Chet and Sam and the revised MOU?

We should have seen that letter, and presumably we will see those kinds of letters in the future.

DR. SIESS: We are working on it.

MR. MICHELSON: Now, the matter immediately at hand, generic issue 87, which is a proposed resolution for the HPCI steamline break without isolation.

This issue concerns the ability of the motor-operated valves in the BWR HPCI steamline to isolate, if you should experience a downstream pipe break. That was the original issue.

The issue was eventually expanded to include the reactor water cleanup breaks or any other system breaks or demands for high Delta-P high flow kinds of operabilities that may not have been perhaps considered.

Some recent tests of course at Wylie Laboratory indicated that perhaps these valves might, their operability might be in question.

And the staff issued them the generic letter 89-10 to cover this operability question.

Now, generic letter 89-10 does require that each licensee have a program that identifies all of its safety-related valves, performs certain kinds of analyses and tests, and take appropriate corrective actions if deficiencies are found. And that program is required to include the HPCI valves, the steamline valves.

So the staff proposes that the resolution of generic issue 87 has been accomplished by the issuance of generic letter 89-10.

The subcommittee reviewed this proposed resolution and agrees that indeed the resolution is acceptable for those cases wherein the existing plant design basis calls for isolation of these valves under breakflow conditions, which is the issue represented by generic issue 87.

However, as I discussed earlier, this operability requirement may not have been in the original design basis for the plant.

And so generic letter 89-10 can be applied only if breakflow interruption is specified in the original design basis.

So we did agree that the issue is resolved except for those cases wherein breakflow interruption wasn't specified to begin with. For those cases the issue isn't resolved and the resolution document remains silent on what you do in those cases.

So I have drafted a proposed resolution letter, which is the yellow copy you have in front of you, to cover this particular point.

And we have the option of doing a first reading of that now.

Let's see. This was due to terminate at 10:30?

DR. REMICK: We have 15 minutes.

MR. MICHELSON: We have 15 minutes. So I was hoping to get a single reading of this letter in in case there are any problems.

So with your permission then, I will do a single reading of the letter.

DR. REMICK: Charlie, do you have a copy? Yellow, Draft 2.

MR. WYLIE: I don't believe so.

(Pause)

DR. REMICK: I think everybody has a copy. Let's go.

(Discussion off the record)

(Whereupon, the lunch recess was taken, the meeting to resume at 1:20 p.m. on the same day, Friday, October 6, 1989.)

(Continued on the next page)

AFTERNOON SESSION

(1:20 p.m.)

DR. REMICK: Gentlemen, if you recall, a meeting or so ago we decided it would be beneficial to have a briefing on the CANDU-3. That time has come. Today is the day. And I would like to turn the meeting over to Dave Ward, our subcommittee chairman.

MR. WARD: Thank you, Mr. Chairman. I will just take a couple seconds.

But several months ago when I recognized that the staff was considering a review of the CANDU plant or possible, and there was interest from AECL, I guess, in getting apparently a design certification in the U.S. with the NRC, although these things were in the future, it seemed to me to be appropriate that the ACRS should hear a little bit about the CANDU design so it would be in a position to make a technical review of it if and when the time came for that.

So today, as I understand it, we are going to see a fairly lengthy television presentation describing the design, but there are also some gentlemen here who will be able to answer questions that we might have after that's over.

Drew Persinko of the staff will take it from here.

MR. PERSINKO: I'll give a brief introduction, a little background information.

On May 25th, the NRC received a letter from Atomic Energy of Canada, Limited, AECL. The letter stated AECL's intent to seek standard design certification under Part 52.

The NRC responded to the AECL letter by letter dated July 6, 1989. In the response letter we requested AECL to submit a licensing review basis, also submittal schedules upon which AECL intends to submit the application and proposed acceptance criteria. That is in accordance with the similar type documents that have been put forth on the ABWR and other standardized plants.

Currently, the staff has written a Commission paper. It's currently in concurrence. It's an information paper for the Commission. The staff is also in the process of developing a Commission paper on the review priorities of the evolutionary and advance reactors. In that paper CANDU-3 will be considered in the prioritization.

The purpose, as was stated, today is to discuss the AECL's organization and the CANDU-3. And with that, I would just like to turn it over to AECL, Gary Kugler, Deputy to the Vice President

MR. KUGLER: Thank you very much, Drew.

I would like to express my thanks, first of all, for having the opportunity to be here and the video will be self-explanatory, I think, but it's appropriate to say a few words by way of introduction.

Before I do so, I would like to introduce also Louis Rib, who has joined our team in our office in Rockville; and Mausimo Bonechi, who came down from Toronto with me. We are here to answer, hope to answer any questions that you may have on the video.

And I just would like to perhaps emphasize a little what may be stated in the tape. And that is, AECL has been looking at the U.S. market for a number of years, and last year my company made a corporate decision to try to play a role in what we believe will be an emerging nuclear market in the next decade and in the century ahead.

At this time we are in the process of developing the CANDU-3 design. It's a small reactor, 450 megawatts, in Canada. We are in the midst of a three-year standard design program. We are also in negotiation with one of the Canadian utilities and we believe later on this year or early next year there will be a formal decision made to build the first CANDU-3 in Canada.

In assisting us in getting established in the U.S., we opened an office in Rockville last year. It's headed by one of our vice presidents, Mr. Denny Shiflett, and it has a dual purpose. We are trying to market and sell some CANDU-derived technology in the services area in the U.S., and primarily, though, our purpose is to try to get a license, a standard design certification for our CANDU-3 product with a view to hopefully getting one CANDU sold in the U.S. this next decade, and playing a significant part in your program thereafter.

And I think the rest will be self-explanatory, and we will try our best to answer any questions that you may have.

MR. MICHELSON: How long is the film?

MR. RIB: The film is two hours, but I'm going to attempt on my feet to edit it for you. There are sections in there -- this is a working film that was taken during the presentation at White Flint with the senior NRC people. There are some sections where the audio is not very good and I'm going to attempt to move through that section and just focus on the technical presentation.

MR. KUGLER: On the tape you will first see Mr. Gordon Brooks, our Vice President and Chief Engineer, giving a general introduction to AECL, and who we are and what we have done. He will be followed by Denny Shiflett, but I believe Louis plans to edit that out because the sound is not very good on that section.

Thereafter, it will be Ken Hedges, who is our project director for the standard CANDU-3 design program that we have underway in Canada now.

After that comes Jerry Hopwood, who is in charge of our licensing department. He will talk about our up-front licensing efforts that we are undergoing vis-a-vis the Atomic Energy Control Board in Canada.

And lastly, he will be followed by Robin Ashwell, who is in charge of our I&C department. He will be speaking about some of the man and machine interface work that we're trying to incorporate into the CANDU-3 design.

(Video tape being shown.)

DR. SHEWMON: What is the time span on that?

MR. KUGLER: It's from the late '60s to present.

(Video tape being shown.)

MR. KUGLER: Actually, I think each reactor size starts at the left-hand side. So it depends on which units, how long they have been operating. But the particular units went in operation in 1971.

MR. RIB: I'm going to move ahead now to Ken Hedge's talk. He's the project director of standard development on the CANDU-3, and he'll get into the CANDU-3 design.

MR. WARD: A quick question.

The enriched uranium comes from LWR?

MR. RIB: Gary, would you pick up on that?

MR. KUGLER: If we went to a slightly enriched uranium cycle, up to 1.2 percent, we could use in current designs of CANDUs, without any change at all, just a change in the fuel management sequence. That we would have to get from an enriched plant.

Other cycles, the recovered uranium cycle, the tandem fuel cycle, they would use spent fuel from LWRs, garbage burner concept, in effect. Spent fuel from LWRs will go into CANDU. There is enough uranium or plutonium that it would still make good fuel for CANDU.

But the slightly enriched uranium cycle would be simply 1.2 percent enriched uranium from an enrichment plant.

MR. WARD: And what are the institutional arrangements there? I mean where are you getting this material?

MR. KUGLER: We hope to be getting it from the international market, from the states, from anywhere else. We have not yet done that. Perhaps in the next five years we may introduce it. All currently operating CANDUs use the natural uranium cycle.

DR. SHEWMON: Is that metallic uranium or uranium oxide?

MR. KUGLER: Uranium oxide.

(Video tape being shown.)

(Videotape being shown)

DR. SHEWMON: Does each plant has its own heavy water plant like that one out in Brunswick?

MR. KUGLER: No. The heavy water is loaded one time in a CANDU-6 reactor about 450 tons. The annual makeup requirements are of the order of 3 to 5 tons.

DR. SHEWMON: And the cleanup --

MR. KUGLER: The cleanup --

DR. SHEWMON: -- leak in the steam generator?

MR. KUGLER: Each plant has its own upgrader. So if you lose some heavy water, it usually downgrades because of light water moisture in the air and then it has to be upgraded back to reactor grade heavy water, which has a purity of about 99.75 percent.

So each plant has an upgrader.

DR. SHEWMON: Is that a still or is that this sulfide chemistry that you use for separation?

MR. KUGLER: I'm not sure about it, but I think it is effectively a still. Whereas the heavy water production plants, which have a capacity of about 800 tons a year, they are the H2S process, where you bubble H2S through ordinary water and you gradually enrich the heavy water portion.

DR. REMICK: What is the location of the heavy water plant?

MR. KUGLER: It is on Lake Huron at the Bruce

Nuclear Power Plant site.

DR. REMICK: At Bruce.

MR. KUGLER: Not too far from Detroit.

(Videotape being shown)

DR. REMICK: Incidentally, did I hear them say that there is a positive coefficient?

MR. KUGLER: Yes. There is a positive void coefficient in case of a loss of coolant accident.

During normal operation, the power coefficient is very close to zero, actually slightly negative. But in case of loss of coolant, there is a positive void effect.

DR. REMICK: And what are the tradeoffs for eliminating that?

MR. KUGLER: You would have to go to slightly enriched fuel and use absorbers in the fuel. With natural uranium fuel, it is not possible to get a negative void effect.

MR. WARD: I guess that must be dictated somewhat by the fuel bundle size and so forth.

MR. KUGLER: That is correct. It is the fuel, it is a cluster type of fuel design. And with that you need fairly large lattice, picture lattice spacing.

And the CANDU reactor is designed almost at the maximum core reactivity to get any amount of burnup. Not quite. In order to save heavy water, we have kept it a little tighter than maximum. But it is very close to maximum, so you can't squeeze much more reactivity out of the core by increasing the lattice patch.

And if you decrease it, then you are suffering on burnup. So we've optimized that lattice pitch of a little over 11 inches.

MR. WARD: Wouldn't decreasing the lattice pitch eliminate the positive void coefficient?

MR. KUGLER: You are quite right. However, CANDU reactors have so little excess reactivity that you very closely get no burnup, very quickly get no burnup if you reduce the lattice pitch. That is why we have fuel on power.

DR. REMICK: With the slightly enriched, the 1.2 percent, do you get away from that problem?

MR. KUGLER: Yes. If we put absorbers in the fuel. So we take a little penalty on fuel burnup. But we get quite a significant amount of additional burnup in going to slightly enriched fuel.

DR. REMICK: Going back to my earlier question about where your heavy water plant is, and you answered Lake Huron, didn't you have one in Nova Scotia at Cape Britton also that had some difficulties? Is that operating at all?

MR. KUGLER: Quite right. We built one on the sea in Nova Scotia, in fact two plants. The first one had some corrosion difficulties. It was initially built by a private concern. AECL took it over, rehabilitated it. We got it into operation.

But at that time, we were foreseeing a very large nuclear power program. And there is not enough demand today. So in fact we are dismantling those plants.

The plants that are now operating on Lake Huron are owned and operated by Interior Hydro, a major utility in Canada. And the current production capacity is adequate for a moderate kind of program.

If there were a large program on the horizon, we would have to build additional capacity.

DR. REMICK: Thank you.

MR. WARD: If you did build additional capacity, do you think that would be the HGS process --

MR. KUGLER: We are developing and looking at cheaper ways of making heavy water, yes. I don't know what process would be used for any future plants. We've got active R&D in process looking at cheaper ways, better ways of making heavy water.

We have to accept the fact that the heavy water cost adds about 10 to 15 percent to the capital cost. And we believe that is offset by the lower fuel costs.

MR. RIB: Are we ready to continue?

MR. KUGLER: Please continue.

MR. RIB: I forwarded the tape to Jerry Hopwoods presentation on safety and licensing approach.

(Videotape being shown)

MR. RIB: That completes Hopwood's section. We have one more section on computer controls.

If you wanted to take a break, now would be a good time. Or we could continue and finish up.

MR. MICHELSON: I think now is a good time to take a break, I think, isn't it, David?

MR. WARD: Yes.

MR. MICHELSON: We will take a break until ten after.

(Whereupon, a brief recess was taken.)

DR. REMICK: We can proceed.

MR. RIB: We are going to the Ashwell talk on computer controls now. And this is the last presentation.

(Videotape being shown)

MR. RIB: That's all.

MR. WARD: Thank you. Thank you, Louis. We have another 30 minutes scheduled for this. And the idea was to provide an opportunity for some interchange and perhaps questions from the committee to the gentlemen who have come today.

I have a number of questions I would like to ask. But I will give someone else a chance to go first.

Does anybody have questions?

DR. SHEWMON: Yes.

What happens when a fuel tube ruptures?

MR. KUGLER: A fuel tube is in effect, a pressure tube failure in effect like a small LOCA as far as the overall coolability is concerned.

First of all, the pressure tube is surrounding by the calandria tube, and we have had pressure tube failures in Pickering, in particular.

In that case, the calandria tube contained the failure and the heavy water leaked out at the end of the channel into the reactor building.

It was noticed as a very gradual loss of heavy water in the heavy water makeup tank, and the reactor shut down the reactor under normal sort of reactor, using normal reactor controlled shutdown procedures.

The safety systems, shutdown systems were not even invoked. It was very expensive --

DR. SHEWMON: And if it had gone through the calandria?

MR. KUGLER: If it goes through the calandria, then the reactor, sorry, the moderator is displaced through a rupture disc from the calandria. And it also will then ultimately spill onto the reactor floor.

But it manifests itself essentially as a small

LOCA in terms of whether there is a need for the emergency core cooling system or not.

In the case of the first failure in Pickering, the emergency core cooling system was not invoked.

DR. SHEWMON: The rods go in and you add makeup water until things calm down, is that correct?

MR. KUGLER: Yes. As I said, in the actual case, the shutoff rods were not even needed.

We have effectively three systems to shut down the reactor. Two are considered special shutdown systems. They only come in if there is a trip signal or scram signal.

The third system, the normal regulating shutdown system, comes in whenever there is a need sensed, either invoked by the operator or a need by various instrumentation to reduce the reactor power.

And it was actually the regulating shutdown system that came in. The shutoff rods, the special safety shutdown system using the shutoff rods, they never even came in, as it was a relatively slow event.

And the long term effect is to simply continue pumping water through the fuel channel. It will leak, and you recirculate that.

In a multi-unit station, which Pickering was, I believe they hooked up to the neighboring units to use cleaner heavy water rather than dirty water off the floor, which is downgraded.

At that time, the operator obviously didn't appreciate the fact that the plant would be shut down for long term for complete replacement of all pressure tubes. It was like a transplant, heart transplant of a reactor, in effect.

MR. CARROLL: I am curious as to what your basic approach is going to be for certification.

Are you going to be in effect forgetting all the licensing history that you had on your reactors and adapt the NRC regulations as something that you are going to meet, or are you going to try to show that your regulations are equivalent or better than the NRC regulations?

MR. KUGLER: In general, we feel that in view of the basic safety objectives being very similar, and I think there are more similarities than differences, we hope to be able to demonstrate that the safety objectives, the NRC safety objectives are met.

And we believe it will be largely a question of how we demonstrate it, putting the documentation into a form that is traditionally expected, also confirming our analysis tools, making NRC staff familiar and confident in some of our experimental verification of those tools. There may be cases where we may have to ask for exceptions.

For instance, a case in point may be the codes and standards applied to pressure tubes, which are zirconium. I don't think there are equivalent codes and standards used in the U.S.

So there may be some cases where CANDU does have some features that don't have a good parallel to go by. And we may have to ask for some exceptions. But in general we hope to be able to demonstrate that the safety objectives and licensing requirements of the U.S. are met.

We are not deluding ourselves that it is going to be simple. We recognize that there is a long road ahead. But we have made that commitment and we intend to proceed.

MR. WARD: The speaker described the experience at the Point Lepreau site, indicating that this was the first or one of the first of the CANDU reactors that was not operated by experienced Ontario Hydro, but that things went very smoothly with that operation.

What was the staff? I mean the implication was they just sort of collected all the boys off the farm and had them run the plant.

Was that really the case, or did they draw experienced staff from, for example, Ontario Hdyro?

MR. KUGLER: They drew experienced staff for their more senior technical and operations staff from Ontario Hydro, from AECL, from people that had had some offshore experience in Pakistan and in India.

But the general working level staff were recruited locally.

New Brunswick is quite a proud little province. They recognize that building a nuclear unit has a lot of local benefits and they wanted to use it as much as possible to create local expertise, create jobs, and not just the mundane construction jobs but also some of the more interesting technical positions they intended to fill. And they have done so by and large.

There is a relatively small number of people that were brought in from the outside. They have an installed grid capacity of a little over 3,000 megawatts. The CANDU-6 puts out 640 megawatts net.

So it is a good fraction.

In terms of generation, they produce about 30 percent of the electricity from the nuclear unit. And Lepreau now has a lifetime capacity factor of more than 91 percent after six and a half years of operation.

MR. WARD: Some of the points made about the severe core damage and severe core damage analysis, and that there was something associated with the number smaller than 10 to the minus 6, and there was some discussion on the tape, but I didn't quite get it.

What was the definition of severe core damage in that case? Was that loss of all the normal heat sinks or was it actual fuel damage?

And then there was reference to a U.S. code package used for analysis. And I am curious about what that code package is and what were the results, I mean what was found out about the nature of severe accidents at the CANDU plant?

MR. KUGLER: Our definition of severe core damage I think would be comparable to what you would use in the U.S. It is largely when you lose a coolable core geometry.

Now, we have one dual failure accident that we postulate which by some persons' definition might be severe core damage. In our cases, it is a design basis dual accident.

And that is the case of a loss of coolant accident, a major LOCA, combined with total failure of the emergency core cooling system.

In Canada we have to satisfy the regulatory that even in that case we do not lose the core geometry.

What happens there, the moderator which surrounds the fuel channel which is not lost during a LOCA, it stays, it is cool, it is not boiling, it is low pressure. And it becomes a heat sink in that case.

Now, the scenario there is that you will have fuel damage, but to the extent that you can still meet the AECS criteria for offsite releases on fuel, on dual failures, which is no more than 25 rem to the most exposed individual. And then there is also a population dose limit. That is a design basis accident. We would not call that a severe core damage.

Whereas I think LOCA plus EEC failure in LWRs would constitute a severe core damage.

Severe core damage you would run into if you went beyond that.

If you also postulate that you lose the moderator as a heat sink, then you would get a deformation of the fuel channels and possibly slumping of fuel channels to the bottom of the calandria, which is our reactor vessel.

It is not clear how the scenario goes from there. We do have the calandria surrounded by a light water shield tank which is also quite a significant heat sink. And it will offer cooling to the calandria shell, and most likely contain any slumped fuel channels.

You know, one can continue the analysis based on various assumptions depending on how pessimistic you want to make the scenario. But that kind of scenario we would call severe core damage.

As to the reference to a U.S. Code package, I am not familiar with it. Perhaps my colleague is?

MR. BONECHI: I think I couldn't understand very well what was, couldn't hear very well what was said. But I think you are referring to some analysis that was done of event sequences that lead to core melt. A study that was done for a CANDU-6 reference reactor in cooperation, which a Dutch organization, KEMA.

And basically, for the consequence analysis of these core melt sequences, yours was made of a consequence analysis code that is basically the same as used in the U.S., which was used also for the WASH 1400.

MR. WARD: Do you know the name of it?

MR. BONECHI: I'm not sure if it CRAC or something like that.

MR. WARD: No the number, let's see, when you say the fuel has access to the moderators, the heat sink, is that -- Well, the fuel tubes have this insulating gas --

MR. KUGLER: Yes. Correct.

MR. WARD: As I understand, the pressure tube has to heat up --

MR. KUGLER: Yes. Actually, what happens, it expands circumferentially and it contacts the calandria and the heat transfer to the moderator is via convection through that route.

MR. WARD: And that is part of the design basis?

MR. KUGLER: Yes. Yes. And we have fully confirmed our analysis methods by doing full scale experiments on fuel channels under those conditions.

DR. KERR: How does the decay heat on shutdown compare with light water reactor oxide fuel, the other type?

MR. KUGLER: I would say very similar. Immediately after shutdown, a theoretical number would be about 7 percent decay heat. I think after a day or so it would be down to 1 percent.

I think it is very comparable.

Our shutdown cooling system, by the way, the equivalent of the residual heat removal system in CANDU is designed to be connected into the primary heat transport system at full pressure and temperature.

So we could in theory connect immediately without waiting to depressurize the heat transport system.

One other feature, which is perhaps noteworthy in the context of loss of coolant accidents is that because we do not have a thick pressure vessel and all our pressure boundary is effectively thin walled piping, we can crash cool the heat transport system very quickly.

And one of the LOCA signals in fact is to immediately depressurize the secondary site, the boilers, to reduce the pressure in the primary heat transport system within literally seconds, to a point where even though the high pressure emergency core cooling stage can inject at high pressure, but you can reduce the pressure very quickly so that you have a good guarantee of getting the emergency core cooling into the system.

(Continued on the next page)

MR. WARD: What's the basis for the containment design? I mean, it looks like a large dry containment as was mentioned, but is it sized to accommodate blowdown of the primary system or just that?

MR. KUGLER: Yes. The basis is a complete blowdown of the primary heat transport system, which establishes the peak pressure, the design pressure.

We have, as we say, a dry containment. And perhaps a saving feature is the fact that there is relatively little energy discharge into containment because the amount of liquid that you lose in case of a LOCA is relatively small because the moderator is separate from the coolant.

So the energy discharge in containment is fairly small and therefore our design pressures in CANDU containments are I believe significantly lower less than would be the equivalent in LWRs.

We do not design for main steamline pressure except to the extent that we have to demonstrate that the reactor building would not be substantially damaged. It can crack, it can leak. But in case of a main steamline break we don't have any fuel failures at all. So there is no radioactivity injected into containment; the concern simply being structural integrity in that case.

MR. CARROLL: Well, in addition to the pressure challenge to containment what other challenges do you think the containment is designed for. I'm thinking, you know, in our large dry containments we're worried about hydrogen; we're worried about core concrete interaction, all of those things.

MR. KUGLER: Hydrogen we also, of course, have to take into account. In the CANDU-6s we did not feel the need for igniters; in CANDU-3 we are putting them in, largely due to a requirement by a regulatory.

I think our analysis would indicate that we are not likely to reach a sufficient hydrogen concentration, even if we postulate all the zirconium cladding to be consumed in water zirconium reaction.

Nevertheless, there could be pockets of a hydrogen concentration and we're putting in igniters for that reason. Other design bases criteria would be external events, aircraft impact, and so on.

DR. SHEWMON: Let me come back, you got a large amount of zirconium in that reactor and you say you can react it all to hydrogen and not raise the concentration of hydrogen up to past 15 percent or something.

MR. KUGLER: I believe our analysis assumes that all the cladding is reacted. I'm not quite sure what our analysis leads to in terms of interaction of the zirconium that's in the pressure tube. I don't think that reaches a sufficiently high temperature to react.

DR. SHEWMON: And then you mix it uniformly throughout your large dry containment and reach that conclusion; is that it?

MR. KUGLER: On an average basis, yes, we have less than four percent, significantly less. But we do recognize that certain areas might have higher than that -- high concentrations than that.

Now, one: it's an interesting situation that the only time we would get into a zirconium water reaction is in case of a loss of emergency core cooling. Only then does the cladding heat up to that temperature.

Now, if you postulate that you lose all of the emergency core cooling then there's not enough water to react with. And if you postulate that you have enough water to react with, then you've also got cooling and then you don't reach that temperature. So regulatory, though, because of the uncertainties does require us to postulate sort of the hypothetical situation that you have both no cooling but still enough water to react with the zirc.

DR. SHEWMON: I suppose you don't have this chimney effect that we do where we can keep furnishing water from the bottom as it boils away and bring it past zirconium on the way out.

MR. KUGLER: No. By the time we get the benefit of an emergency core cooling the sheet temperatures would come down to a point where there wouldn't be a chemical reaction.

DR. SIESS: What's the design pressure for the containment?

MR. KUGLER: In the CANDU-6 I believe it was 18 psi; in the CANDU-3 it's 30.

MR. BONECHI: 200 kpa.

DR. SIESS: I didn't get it.

MR. BONECHI: 200G.

MR. KUGLER: So that's 2 atmosphere gauge.

MR. WARD: And the volume is about what? The volume of containment? I looked in the book and couldn't find it.

MR. KUGLER: I don't know the volume. I know the leak rates that were designed, but I'm not sure of the volume.

MR. WARD: Is it about 2 million cubic of the size; it looks smaller than that.

MR. KUGLER: It's quite small.

MR. WARD: Okay.

DR. REMICK: What's the relationship between the vacuum vessel and the containment? What's the function there and the inter-relationship?

MR. KUGLER: In the vacuum building -- first of all just to clarify, we use only on multi unit stations. For economic reasons it just doesn't make sense to have it coupled to a single unit station.

The vacuum building is effectively under vacuum. It's only a few millimeters mercury pressure. And if you have a LOCA in any one of the reactor buildings the steam and fission products are sucked into the vacuum building, literally sucked and not driven by pressure but actually vacuum. So it forms part of the containment.

And for small LOCAs or most likely LOCAs, the pressure in the building will never reach atmospheric, so if anything there's an in-leakage rather than out-leakage.

I think only in the largest LOCA might the pressure go slightly above atmospheric.

DR. REMICK: Yet they're designed for two atmospheres in CANDU-3?

MR. KUGLER: No, the CANDU-3 is a single unit containment; it's not vacuum.

DR. REMICK: Okay.

MR. KUGLER: It operates just slightly below atmosphere normally. I don't know what the design pressure of our multi unit vacuum building is; I'm sorry, I don't know that.

DR. REMICK: And there is some kind of a valve in between that has to be opened?

MR. KUGLER: Correct. It's a very large -- I think it's more like a membrane or butterfly valve.

MR. CARROLL: Back to my containment challenge question: do you have the equivalent of our big concern about core on the floor?

MR. KUGLER: I don't think that scenario has been a big factor in the design of the containment building because that's an area we get into at much less than 10 to the minus 6 probability. We normally take the attitude that anything less than 10 to the minus 7 is incredible and is not something that we would factor into the design. We would do analysis. We would try to minimize the likelihood of that event happening. But I don't think we would actually factor a potential consequence into the design of the building.

MR. WYLIE: Let me ask a question about your philosophy on the safety systems. I was reading through your literature here and it implies that you design on the basis of meeting a single failure with the safety system out-of-service -- train out-of-service which would imply an N plus 2 design; is that correct?

MR. KUGLER: When we postulate a safety system out-of-service, in most cases we postulate the total system unavailable. That is certain the case for one of the shutdown systems we postulate, it's just not available.

MR. WYLIE: Is that true of ECCS, too?

MR. KUGLER: In the case of ECCS we would also assume that there's absolutely no water flowing in. In other words, the valves don't open. In fact, we make it worse. As I mentioned before, we assume a trickle of water, just enough to react with all the zirc in the fuel cladding. So it's almost a machiavellian type of scenario.

In the case of containment we do not postulate failure of the entire system; we would postulate typically failure of containment isolation. In other words, the containment stays open as one of the failure mechanisms.

But rather than just postulating the failure of a redundant component in a safety system we postulate the failure of the entire system.

Now, in addition to that we will also postulate, in case of a shutdown systems, failures of certain rods of the system that we do take credit for. Typically in the CANDU-6 we have 28 cadmium shut-off rods. We assume that the two best rods, the most reactivity rods don't drop. We make that assumption and we say that one might be out for testing just at the time of the accident and the other one fails to drop for other reasons.

MR. WYLIE: So you could have a complete safety system out of service and still meet a single failure.

MR. KUGLER: Yes. Correct.

MR. WYLIE: So it's an N plus 2 system.

MR. KUGLER: We meet what we call the dual failure criterion, that is a 25 rem to the most exposed person.

MR. WYLIE: It also says that diversity is the use provided protection against, in the safety systems, to accomplish their functions; is that correct?

MR. KUGLER: Yes. For instance, we apply that in particular to the design of the shutdown systems. We use cadmium dropping rods, dropping under gravity for shutdown system one and we use liquid gadolinium nitrate injection into the moderator in the case of shutdown system two. So we have diverse operating physical principles.

We also have a geometric separation in the sense that the rods come in from the top. The poison injection nozzles come in from the side. We try to use diversity in the design of specific components, maybe select a trip logic and instrumentation from different manufacturers and so on. Really trying to maximize the difference or the diversity, but still insisting that the same specs are met ultimately; typically to shut the reactor down in less than two seconds.

MR. WYLIE: I see. Thank you.

MR. WARD: One other question: you mentioned quite low worker exposure levels experienced and expected with CANDU-3; what about the tritium, is that based on detritiating the moderator or is there a lot of work in plastic suits or some combination of that?

MR. KUGLER: Tritium contributes approximately 30 to 40 percent of the man-rem dose at CANDU station. Even including that the total station dose is quite acceptable by world standards. Typically at a CANDU-6 station a total dose would be about 100 rem. We are hoping to improve that by a factor of two on the CANDU-3.

Yes, people do in certain areas have to work in plastic suits to avoid the internal dose.

As to your question of whether we remove tritium, we do not need to remove it for reasons of man-rem exposure in terms of, we can meet what is considered reasonable and certainly the licensing requirements by just allowing the tritium to build up during the lifetime of the reactor. It does build up and reach sort of a saturation level because the half-life being 12 years or so starts to come in.

Ontario Hydro has built a tritium removal plant at its Darlington Station. And initially, because they thought it might be needed in a long-term to keep the man-rem exposure down. As it turns out that's no longer the reason. They're now stuck with a rather expensive facility and they're trying to use it for commercial gain and sell the tritium.

MR. WYLIE: Do you know anybody that needs any?

(Laughter)

MR. KUGLER: Unfortunately, you can't sell it for certain purposes.

MR. WYLIE: I notice your diesel sets come up in 35 seconds to full speed and load within two minutes. I assume that has to be consistent with your safety analysis.

MR. KUGLER: Yes. The diesels are what we call our class 3 system, class 3 electrical system. We have, in fact, two sets of two diesels: one for the group one systems and one for the group two systems.

I don't know whether it came through clearly through the video, but a very important to us at least a safety concept is the two group separation of philosophy where we have two entire groups or sets of systems which can both do or independently shut down the reactor, cool the fuel and monitor the state of the plant in the long-term.

And we separate those systems physically by having them in different buildings. And within the group two systems, that is the safety systems, those systems are also separated from each other. And this affords a very comprehensive protection for common mode or cross link events. Fires, for example, wouldn't break out in the same part of the plant at the same time. These buildings are separated from each other and flooding in a building due to broken pipe or so on.

So we believe that that affords a very high reliability.

MR. WYLIE: I notice also you have the wisdom to use generator circuit breakers as a common practice.

MR. CARROLL: You made a friend with Charlie for life.

DR. SHEWMON: Common practice in some of the better utilities.

MR. KUGLER: I see.

MR. WYLIE: I notice your reactor coolant pipe motors are totally enclosed water coolers; is that a standard practice with you?

MR. KUGLER: I'm not quite familiar with your terminology in the sense that --

MR. WYLIE: The pump motors.

MR. KUGLER: Yes.

MR. WYLIE: Pump motors, cooling --

MR. KUGLER: Yes, they're enclosed, yes.

I just wanted to add one comment: we pride ourselves on the reactor coolant pump seals and we achieved a bit of a note and fame in having been selected to design the O-rings, redesign the O-rings for Morton Theicol for the Challenger seals.

The reason we got to that stage is that we've always worried about losing heavy water, it's expensive. And seals and valves is one place where you get the leaks.

So we put a lot of emphasis on developing seal technology and what we call live loaded valves where you keep constant load on the packing through a set of specific design springs, and just to avoid leakage of heavy water. It has been very good and I think it's also starting to be applied in other areas.

MR. WYLIE: What is the cost of heavy water per pound at present?

MR. KUGLER: Per kilo we would sell it at 300 kilo[illegible] that would make it a little less than $150, Canadian dollars, translate that to about $120 U.S. a pound. And o the spot market today some people in Europe still have a bit of it around and they usually undersell us. But they wouldn't have enough for a reactor load, but for makeup water we have been under-bid.

DR. SHEWMON: But it's over $100 a pound.

MR. KUGLER: Yes.

MR. WARD: Well, thank you very much. We appreciate you coming down.

MR. KUGLER: It's my pleasure.

Thank you for giving us the opportunity to speak to you.

MR. WARD: Mr. Chairman, that's the end of this.

DR. REMICK: All right. I would suggest that we take a 10 minute break at this time, but I would remind you that you're obligated to provide the Vice Chairman a ranking of nominees for membership, we are awaiting those.

So let's return at 4:15.

(Whereupon, at 4:05 p.m. the meeting was adjourned to reconvene at the call of the chair.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name:

Docket Number:

Place:

Date:

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken stenographically by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

/s/ Joan Rose

(Signature typed):

Official Reporter

Heritage Reporting Corporation

RESOLUTION OF GSI B-56
DIESEL GENERATOR RELIABILITY

PRESENTATION TO THE
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

October 6,1989

A.W. SERKIZ RES/RPSIB
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RESOLUTION OF B-56

1. Resolution is provided in RG 1.9, Rev. 3
2. Resolution is related to Station Blackout.
3. RG 1.155 identified need for EDG reliability program to monitor and maintain EDG reliability levels selected for SBO.
4. RG 1.9, Rev. 3 provides guidance for monitoring reliability levels, defines an EDG reliability program and updates previous guidance issued.

OVERVIEW
RG 1.9, REV. 3

1. Has been revised in response to comments received and discussions with NUMARC's B-56 working group.

2. Integrates into a single RG guidance previously addressed in RG 1.9, Rev. 2, RG 1.108 and Generic Letter 84-15.

3. Defines reliability program and supplements guidance provided in RG 1.155.

4. Better defines testing reqmts, eliminates cold fast starts and minimizes accelerated testing.

5. Defines alert levels, remedial actions and reporting reqmts.

6. Incorporates proven industry practices and is consistent with NUMARC's revised NUMARC 8700, Appendix D.

7. Utilizes INPO's Industry-wide Performance Indicator Program (PPIP) surveillance definitions for consistency.

TABLE 1

CROSS-REFERENCE BETWEEN REGULATORY GUIDE 1.9, REV. 3
AND NUMARC-8700, APPENDIX D

RG 1.9,REV 3 SECTION	NUMARC-8700 APPENDIX D
Section A, Introduction	(Use RG 1.9,Rev.3)
Section B, Discussion	(Use RG 1.9,Rev.3)
Section C, Regulatory Positions	
C.1, Design Considerations	(Use RG 1.9,Rev.3)
C.2, Diesel Generator Testing	
C.2.1, Definitions	D.1
C.2.2, Test Descriptions	(Use RG 1.9,Rev.3)
C.2.3, Preoperational and Surveillance Testing	(Use RG 1.9,Rev.3)
C.3., EDG Reliability Goals and Calculations	
C.3.1, Reliability Goals for SBO	D.2
C.3.2, Design Basis Accident Assesment	(Use RG 1.9,Rev.3)
C.3.3, Diesel Generator Reliability Calculations	D.2.2
C.3.4, EDG Reliability Program Monitoring	D.2.3,D.2.4
C.3.5, Recovery From A Strong Alert	D.2.4.4
C.4, Record Keeping Guidance	D.2.1
C.5, Reporting Criteria	D.2.5
C.6, EDG Reliability Program	D.3
C.6.1, Diesel Generator Reliability Target	D.2.3
C.6.2, Diesel Generator Surveillance Plan	D.3.1
C.6.3, EDG Performance Monitoring	D.3.2
C.6.4, EDG Maintenance Program	D.3.4
C.6.5, EDG Failure Analysis and Root Cause Investigation	D.3.5
C.6.6, Problem Close-out	D.3.6
C.6.7, Data CApture & Utilization	D.3.3
C.6.8, Assigned Responsibilities and Management Oversight	(Use RG 1.9,Rev.3)
Section D, Implementation	Introduction

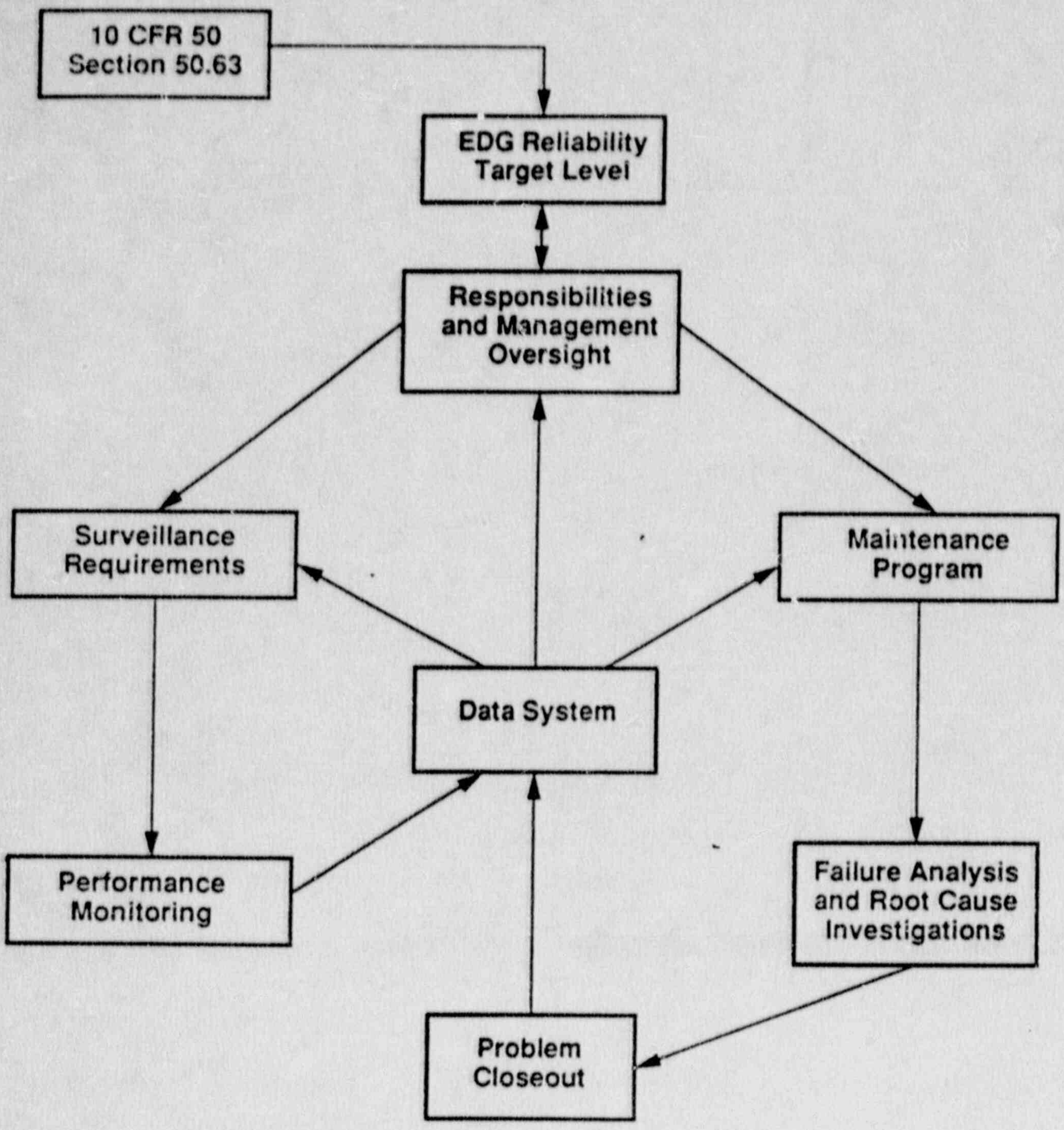


Figure 2 - Interaction of EDG Reliability Program Elements

Figure 1 Graded Response to Degrading EDG Reliability (10-5-89 Draft)

NORMAL ACTION STATE

- o Continue surveillance and condition monitoring according to approved reliability program plan.
- o Repair failures as they occur.

MILD ACTION STATE

Review failures in last 20, 50&100 demands to determine if there are patterns in the failure modes or causes

PATTERN:

- Devise corrective action for observed failure pattern
- Implement a program close-out procedure for the above corrective action

NO PATTERN:

- Increase or improve surveillance and/or condition monitoring for most likely failure modes
- Implement a problem close-out procedure for augmented surveillance/condition monitoring

Notify the NRC on-site inspector of adjustments to the EDG reliability program

STRONG ACTION STATE

1. Notify the NRC of the alert.
2. Ascertain the nature of the reliability problem. Assessment actions should include one or more of the following:
   - o root cause analysis
   - o analysis for patterns in failure modes and causes (last 100 demands)
   - o Assessment of other plants failure information
   - o Exploratory surveillance
   - o Exploratory condition monitoring
   - o Reliability diagnotic analysis (FEMA, fault tree, tracking and trending, etc.)
   - o Design/operational changes
3. Document and implement corrective actions plan.
4. Revise reliability program.
5. Demonstrate effectness of actions (*) taken.

* These recovery actions are discussed in Regulatory Positions C.3.5 and C.2.3.3.

EDG RELIABILITY
MONITORING & ACTIONS

- Based on monthly surveillance testing.
- Nuclear unit monitoring for SBO
- Utilizes reliability program and establishes action states vs. targets.

Target	Action State	Failure Combinations (All EDGs)
.95	Mild	3/20 or 5/50 or 8/100
.95	Strong	4/50 a[illegible] 8/100
.975	Mild	3/20 [illegible]50 or 5/100
.975	Strong	4/5[illegible] 5/100

- Individual EDG:

3/20 ---> Mild Action State (Fig. 1)

4/25 ---> Strong Action State (Fig. 1)

Verification Testing
Reg. Pos. C.2.3.3 -
7 consecutive failure
free tests

5/25 ---> Declare EDG inoperable, determine level of overhaul required.

RG 1.9, Rev. 3
Implementation

1. Apply to all plants for purposes of monitoring EDG reliability levels and reviewing EDG reliablility programs with respect to meeting the SBO rule.

2. Activities related to Design Considerations and Preoperational Testing will not have to be repeated by licensees or applicants where such activities have already been completed.

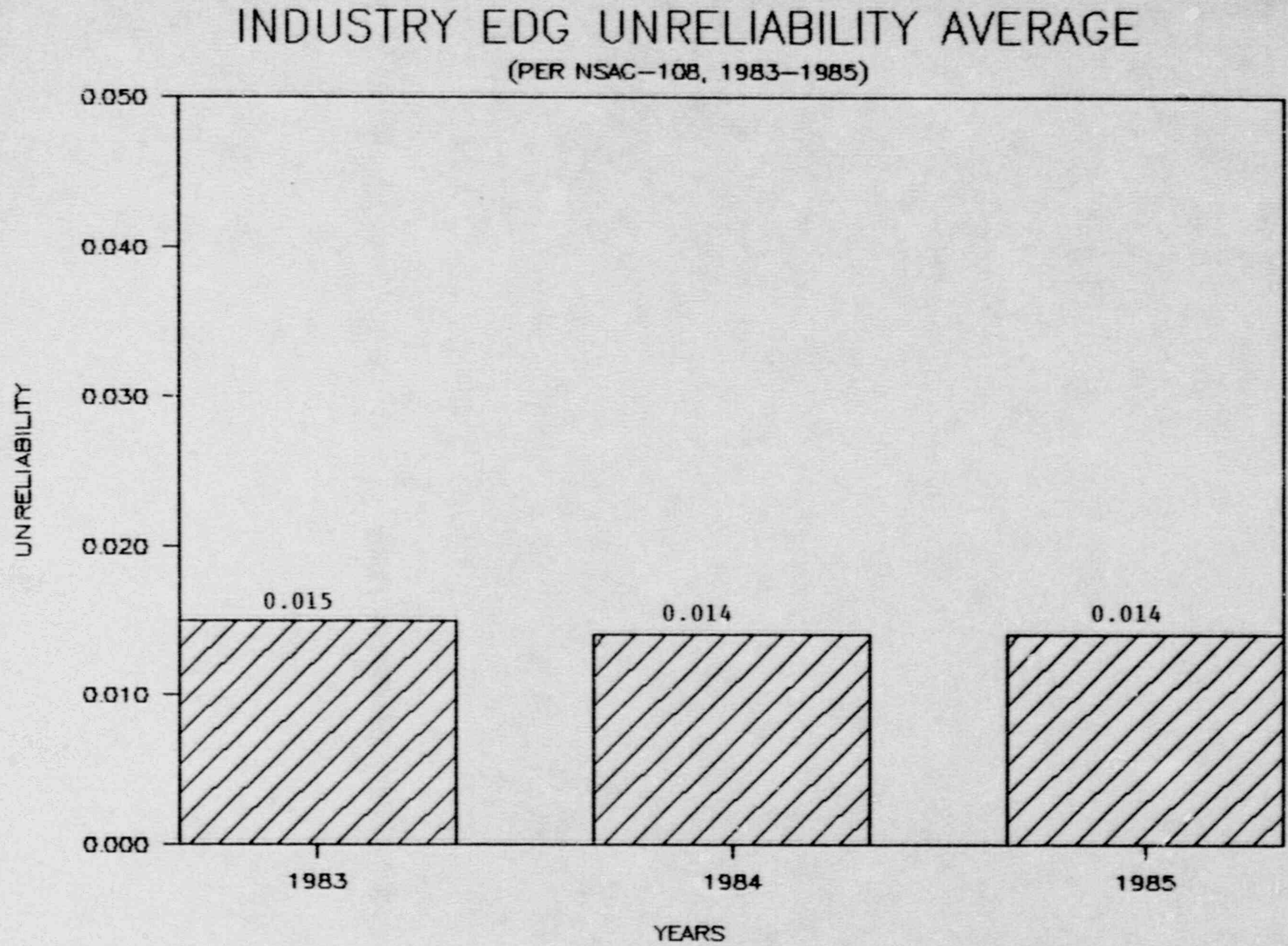
3. Applies to CPs and OLs docketed 6 months after issuance of RG.

4. Applies to ORs 9 months after issuance of RG.

B-56 RESOLUTION

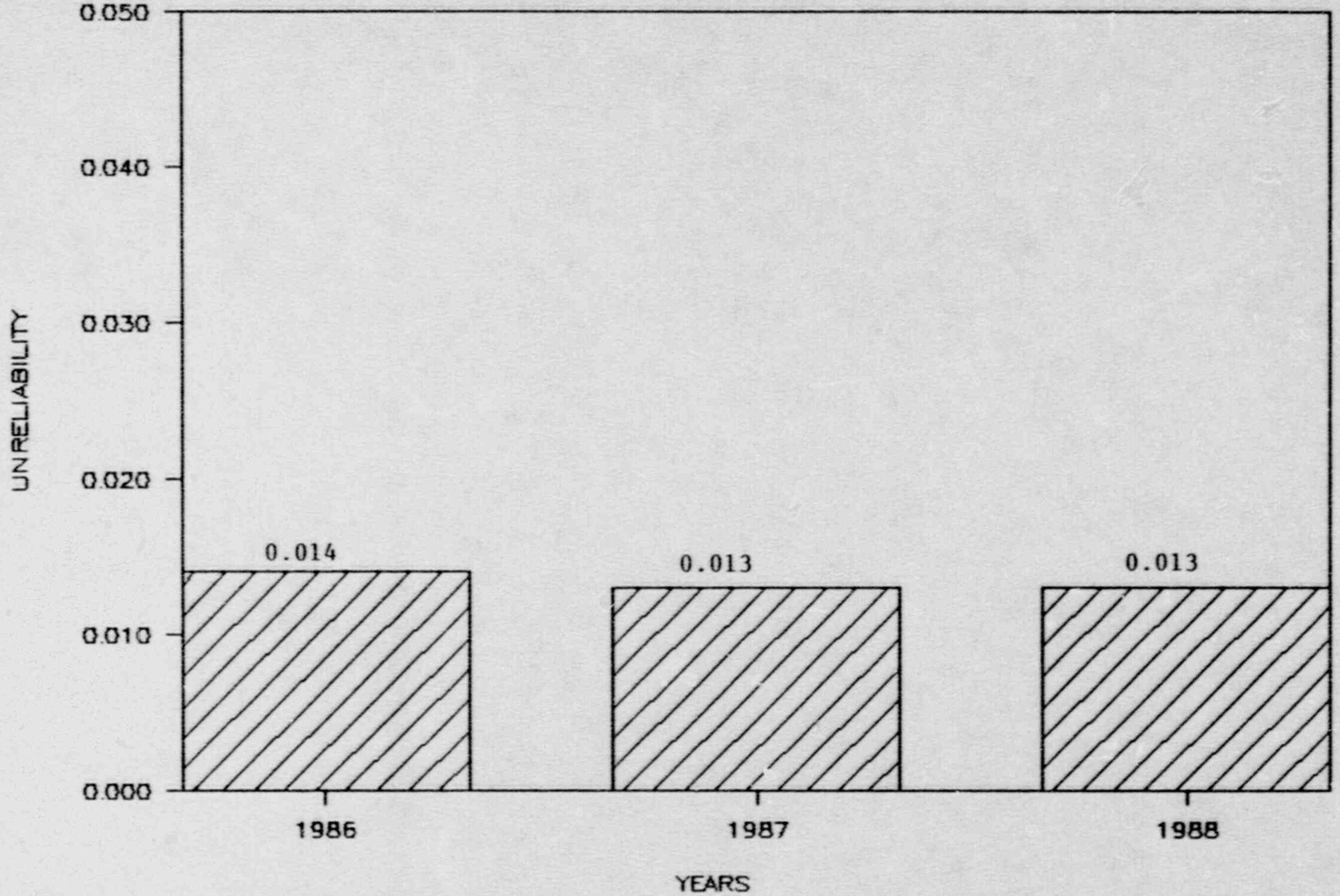
- RES will issue RG 1.9, Rev. 3.
- NRR will integrate findings into Tech Spec upgrades.
- NRR will develop inspection module for evaluating EDG reliability programs.
- NRR has revised pertinent SRP sections and reviewed with CRGR (CRGR Mtg 164,6/89).

~~where~~
#2


INDUSTRY EDG UNRELIABILITY AVERAGE
(PER NSAC-108, 1983-1985)
UNRELIABILITY
0.050
0.040
0.030
0.020
0.010
0.000
0.015
0.014
0.014
1983
1984
1985
YEARS

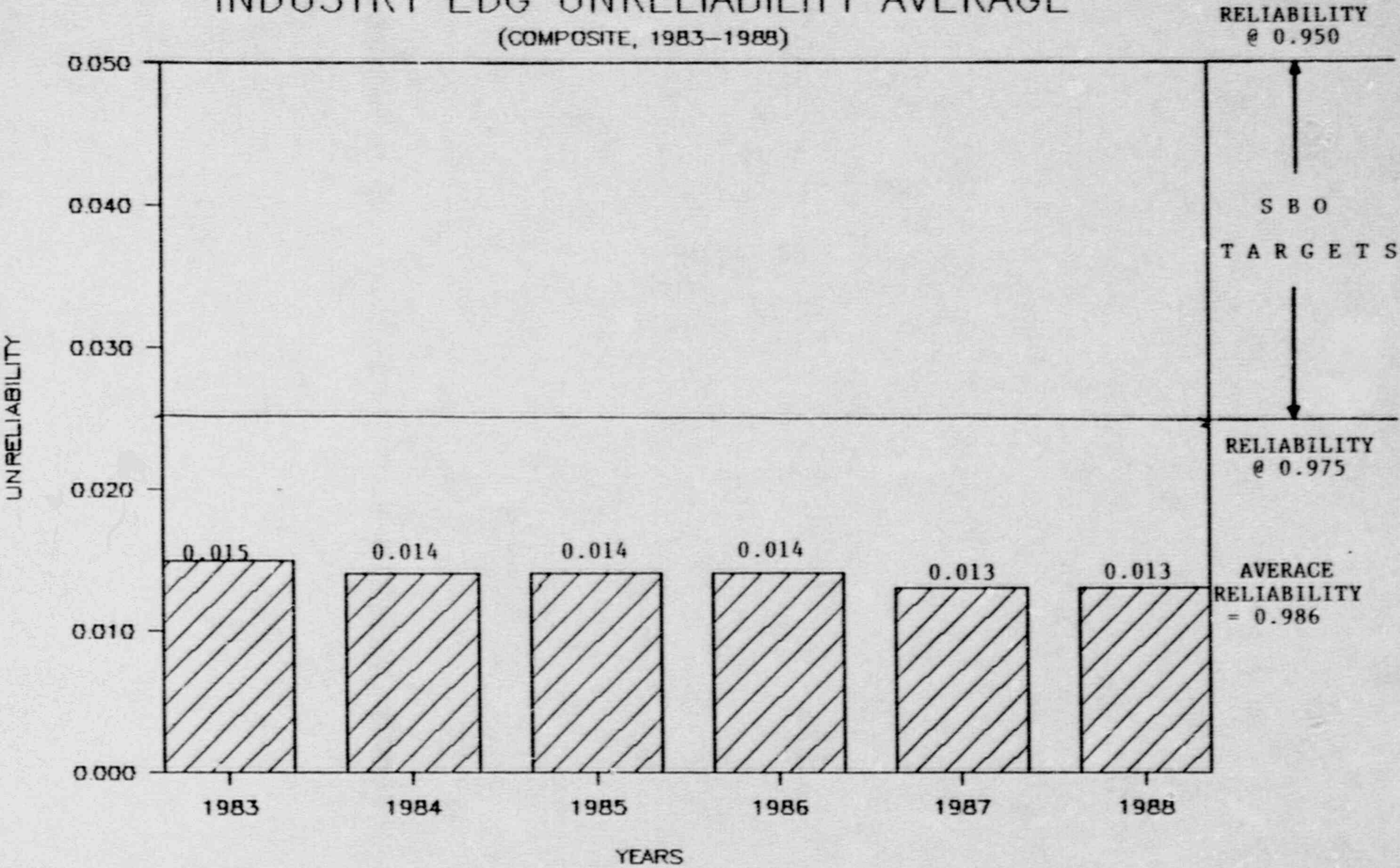
INDUSTRY EDG UNRELIABILITY AVERAGE

(PER INPO PPIP DATA, 1986–1988)



INDUSTRY EDG UNRELIABILITY AVERAGE

(COMPOSITE, 1983–1988)



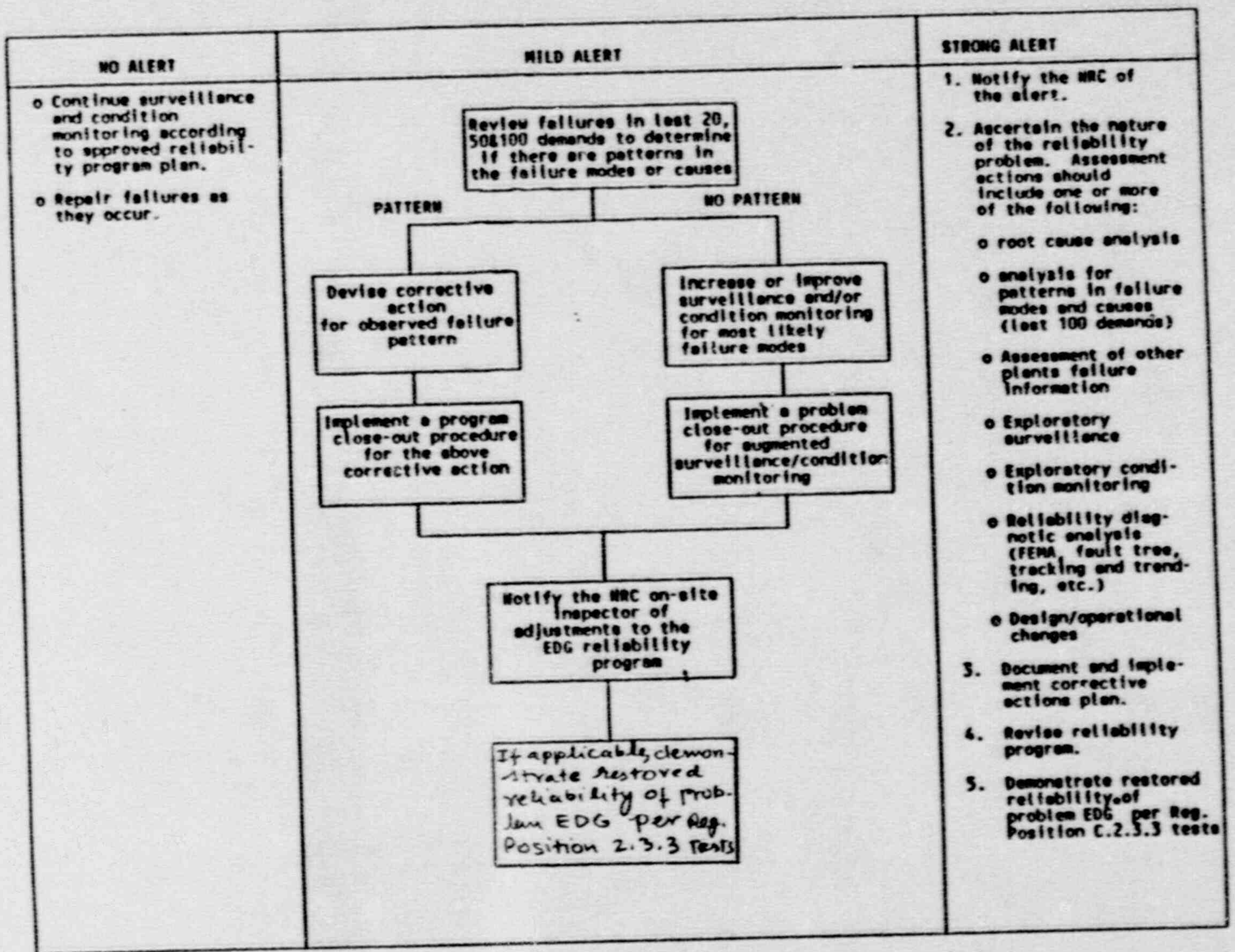
insert 3

2.3.3 Corrective Action Testing: Following the occurrence of a degrading situation as defined in Regulatory Position 3.5 for a problem EDG, the surveillance testing interval for that EDG should be reduced to no more than 7 days, but no less than 24 hours. This test frequency should be maintained until seven consecutive failure-free start and load-run tests have been performed to demonstrate the effectiveness of corrective actions taken and recovery of reliability levels. At that time, monthly surveillance testing can be resumed. However, if subsequent to the seven failure-free tests, one or more additional failures occur such that there are again three or more failures in the last 20 tests, the testing interval should again be reduced as noted above and maintained until seven consecutive failure-free tests have been performed or until the number of failures in the last 20 tests is less than three. The EDG undergoing corrective action testing should be considered "operable" unless other license requirements necessitate declaring the EDG inoperable.

3.5 Problem EDG

If any individual EDG experiences three or more failures in the last 20 demands, then a Mild Alert is declared and actions in Figure 1 are undertaken including the corrective action testing per Regulatory Position 2.3.3. If during the corrective action testing, the EDG experiences additional failures, so that the number of failures in the last 20 demands is five or more (including the previous three failures), consideration should be given to undertaking a major overhaul in accordance with the manufacturer's recommendations for such failures. If the overhaul necessitates the tear-down and overhaul of the diesel engine, then prior to returning the EDG to service, a series of 14 consecutive failure free start and load-run tests (per Regulatory Position 2.2.3) should be conducted. Regular EDG surveillance testing should then commence. Also, any failures which occurred prior to the 14 consecutive successful tests should not be counted for any subsequent determination of the 3/20 failures criterion of this position.

Figure 1 Graded Response to Degrading EDG Reliability



* This remedial action is discussed in Reg. Positions C.3.4 and C.3.5.

CANDU 300

Technical Outline

Insert

CANDU 300
Technical Outline

Document PPS-74-01010-003 Published by

Atomic Energy of Canada Limited — L'Énergie Atomique du Canada, Limitée

CANDU Operations — Opérations CANDU

Sheridan Park Research Community
Mississauga, Ontario L5K 1B2
Canada

ABSTRACT

This report identifies and discusses the major design features of the CANDU 300 which is the latest and also the smallest version of the modern CANDU line of Pressurized Heavy Water Reactor (PHWR) systems developed in Canada. Extensive experience together with updated technology now provides a highly flexible plant configuration capable of adaptation to a wide range of different user requirements and sites.

Design considerations are discussed together with brief descriptions of key items and systems. The CANDU safety philosophy is outlined and the functions of the shut-down systems are identified.

TABLE OF CONTENTS

CANDU 300 Technical Outline

1.0 INTRODUCTION

1.1 Background

The CANDU 300, is the latest and smallest version of the very successful CANDU Pressurized Heavy Water Reactor (PHWR) system developed in Canada. With a net electrical output in the range of 450 MW(e), the CANDU 300 complements the established mid-size CANDU 600, and makes nuclear electric generating capacity available for a relatively modest capital investment, while simplifying the problems of financing and matching construction programs to uncertain load growth. At the same time the cost and environmental advantages of nuclear power over fossil-fired plants are maintained. A schematic of the CANDU steam supply system is shown in Figure 1.1-1.

Proven technology is used throughout the CANDU 300, updated with relevant features resulting from ongoing Canadian research and development. The CANDU 300 is a highly flexible plant readily adaptable to the individual requirements of different utilities.

The CANDU 300 design has paid particular attention to the protection and security of the owner's investment. This includes the minimization of capital cost, the provision of a short and secure construction schedule, the assurance of high capacity factor through the use of highly reliable and easily maintained systems and components, the maximization of component life and the provision for the fast and easy replacement of any component at the end of life. The latter ensures minimal economic impact of any premature component degradation and permits the economic extension of plant operating life.

A very high level of standardization has always been a feature of CANDU reactors. This theme is emphasized in the CANDU 300; all key components (steam generators, coolant pumps, pressure tubes, fuelling machines, etc.) are identical to those proven in service on operating CANDU power stations.

In common with existing CANDU designs, the relatively simple and modular nature of the CANDU 300 permits many reactor and other station components to be manufactured by any country with a basic technical capability; hence, a wide range of countries can participate in the building of the power station.

CANDU reactors in operation or under construction are listed in Table 1.

TABLE 1
CANDU REACTORS IN OPERATION OR UNDER CONSTRUCTION

Name	Location	Capacity MWe net	In-service date
Pickering 1	Canada	515	1971
Pickering 2	Canada	515	1971
Pickering 3	Canada	515	1972
Pickering 4	Canada	515	1973
KANUPP	Pakistan	125	1971
RAPP 1	India	203	1972
RAPP 2	India	203	1980
Bruce 1	Canada	825*	1977
Bruce 2	Canada	825*	1977
Bruce 3	Canada	825*	1978
Bruce 4	Canada	825*	1979
Point Lepreau	Canada	633	1983
Gentilly-2	Canada	638	1983
Wolsung-1	Korea	638	1983
Embalse	Argentina	600	1984
Pickering 5	Canada	516	1983
Pickering 6	Canada	516	1984
Pickering 7	Canada	516	1984
Pickering 8	Canada	516	1986
Bruce 5	Canada	825	1985
Bruce 6	Canada	825	1984
Bruce 7	Canada	825	1986
Bruce 8	Canada	825	1987
Cernavoda 1-5	Romania	665 x 5	1990/98
Darlington 1	Canada	881	1989
Darlington 2	Canada	881	1988
Darlington 3	Canada	881	1991
Darlington 4	Canada	881	1992
	TOTAL	20 613	

*Electrical equivalent (electricity plus process steam)

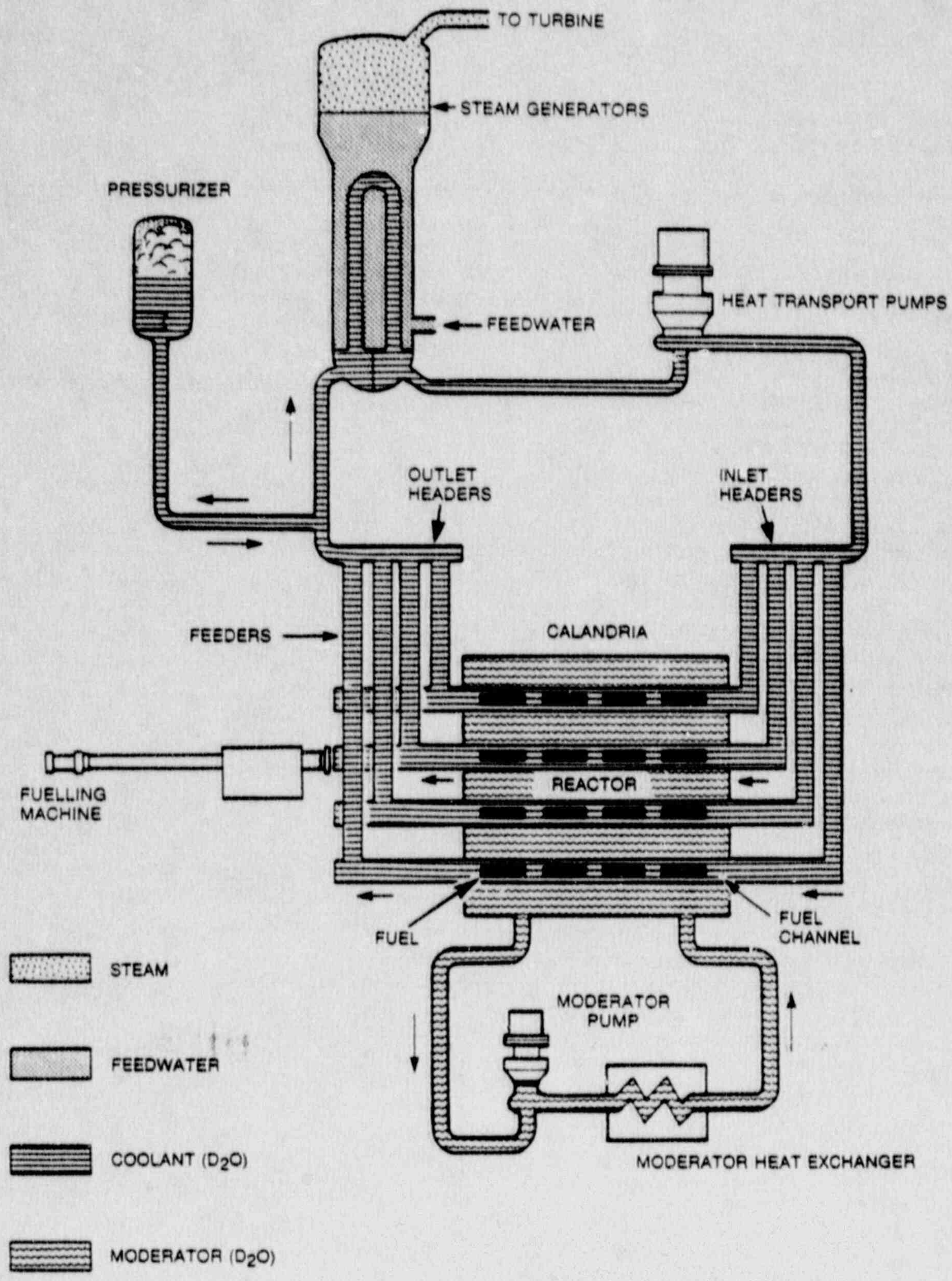


Figure 1.1-1 Steam supply system

1.2 CANDU 300 Unit Data

Key unit data is presented in Table 2.

1.3 Unit Operating Characteristics

The following is a summary of significant operating characteristics:

- The unit is capable of sustained operation at any net electrical output of up to 100 per cent of rated full power output.
- The overall plant control is normally of the reactor-following-turbine type.
- For power increases, the Nuclear Steam Plant (NSP) portion of the plant, is capable of manoeuvering at a rate of 4 per cent of present power per second in the range zero to 25 per cent full power, and at 1 per cent of full power per second in the range 25 per cent to 80 per cent of full power, and at 0.15 per cent of full power per second in the range 80 per cent to 100 per cent of full power.

  The power manoeuvering rate in the lower range is a function of turbine design and is typically 5 to 10 per cent of full power per minute. The unit's manoeuvering rate between 80 and 100 per cent full power is usually limited by the NSP restrictions.
- During normal plant operation with 4-adjuster rods, assuming an initial power of 100 per cent with equilibrium fuel in the reactor, the xenon load at a steady level, and normal flux shape, the reactor power may be reduced to 75 per cent full power at rates of up to 10 per cent of full power per minute. The comparable figure for the 12-adjuster option is 55 per cent. The power may be held at the new lower level indefinitely. Return to high power can be accomplished within 60 minutes or less depending on the degree and duration of the power reduction.

TABLE 2
CANDU 300 UNIT DATA

Reactor	
Type	Horizontal pressure tube
Coolant	Pressurized heavy water
Moderator	Heavy water
Number of fuel channels	232
Fuel	
Fuel	Compacted and sintered natural UO_2 pellets
Form	Fuel bundle assembly of 37 elements
Length of bundle	495 mm
Outside diameter	102.4 mm
Bundle weight	23.5 kg (includes 18.4 kg U)
Bundles per fuel channel	12
Heat Transport System	
Number of steam generators	2
Steam generator type	Vertical U-tube with integral steam drum and preheater
Number of heat transport pumps	2
Heat transport pump type	Vertical, centrifugal, single suction, double discharge
Reactor outlet header pressure	10 MPa(a)
Reactor outlet temperature	310°C
Reactor coolant flow rate (total)	5210 kg/s
Steam temperature (nominal)	260°C
Steam quality (minimum)	99.75%
Steam pressure	4.7 MPa(a)
Total heat transferred to system generators	1379 MW(th)
Net electrical output (nominal)	450 MW(e)*

*Typical for a cold water site; net electrical output is dependent upon cooling water temperature, and turbine-generator and condenser design.

- In the event of a temporary or extended loss of line(s) to the grid, the unit can continue to run and supply its own power requirements.

  The turbine bypass system to the condenser is capable of accepting the entire steam flow during a reactor power setback following loss of line or turbine trip. The steam flow is initially 100 per cent, but decreases to a steady state value in the range of 75 per cent after several minutes.

- The unit is capable of reaching 100 per cent net electrical output, from a cold shutdown in about ten hours. If the pressurizer is at its normal operating temperature and pressure and the xenon level in the fuel is low, the unit is capable of reaching 100 per cent electrical output from a cold shutdown within three hours. These time intervals are for the nuclear steam supply system and may be extended by the turbine generator requirements depending on the turbine design.

- The reactor and turbine are controlled by computer from zero to 100 per cent of full power.

- Following a shutdown from sustained full power operation with equilibrium fuel, the reactor can be restarted within 10 minutes and returned to full power operation. (In the optional 12-adjuster design a reactor restart within 30 minutes is possible, allowing an additional 20 minutes for review and decision making.) If restart within 10 minutes is not possible, xenon buildup and subsequent decay time will result in an outage interval of approximately 40 hours.

1.4 Black Start Capability

Traditionally, nuclear (and other major thermal) plants need external power to start up since the on-site standby generators cannot handle the total station service load.

An optional feature of the CANDU 300 is the capability of plant start-up without power from the external network.

The "Black-Start" process utilizes a low level of reactor power to produce sufficient steam to roll the turbine-generator. The generator output then provides a low frequency power supply to the heat transport pump motors. All other service loads are supplied from the Class III buses energized from the standby generators.

The turbine-generator and the heat transport pump motors accelerate in tandem to full speed.

At full speed, which corresponds to 50 or 60 Hz frequency, the turbine-generator and the standby generators are synchronized, and all service loads are transferred to the turbine-generator from the standby generators.

The "Black-Start" process takes about 10 minutes following turbine warm-up.

1.5 Site Data

The CANDU 300 is compatible with most potential sites, world wide. The Recirculated Cooling Water System is used for all nuclear steam supply system cooling requirements to accommodate salt or contaminated water sites. The recirculated cooling water is cooled by the Raw Service Water (RSW) System. Once-through raw water cooling is utilized in the turbine condenser.

Representative site data is provided in Table 3.

TABLE 3
CANDU 300 SITE DATA

Water temperatures	
Maximum condenser cooling water temperature (full power)	28°C (warm site)
Maximum raw service water inlet temperature	32°C (warm site)
Design Basis Earthquake (DBE)	
Horizontal	0.3 g
Vertical	0.2 g

2.0 DESIGN SUMMARY

2.1 Station Layout

The CANDU 300 Nuclear Generating Station shown in Figures 2.1-1 and 2.1-2 consists of five principal structures (Reactor and Reactor Auxiliary Building, Turbine Building, Group 1 Service Building, Group 2 Service Building, and Maintenance Building) and auxiliary structures. The distribution of equipment and services among the buildings is primarily by function.

To the maximum extent possible, the principal structures are self-contained units with a minimum number of connections to the other structures.

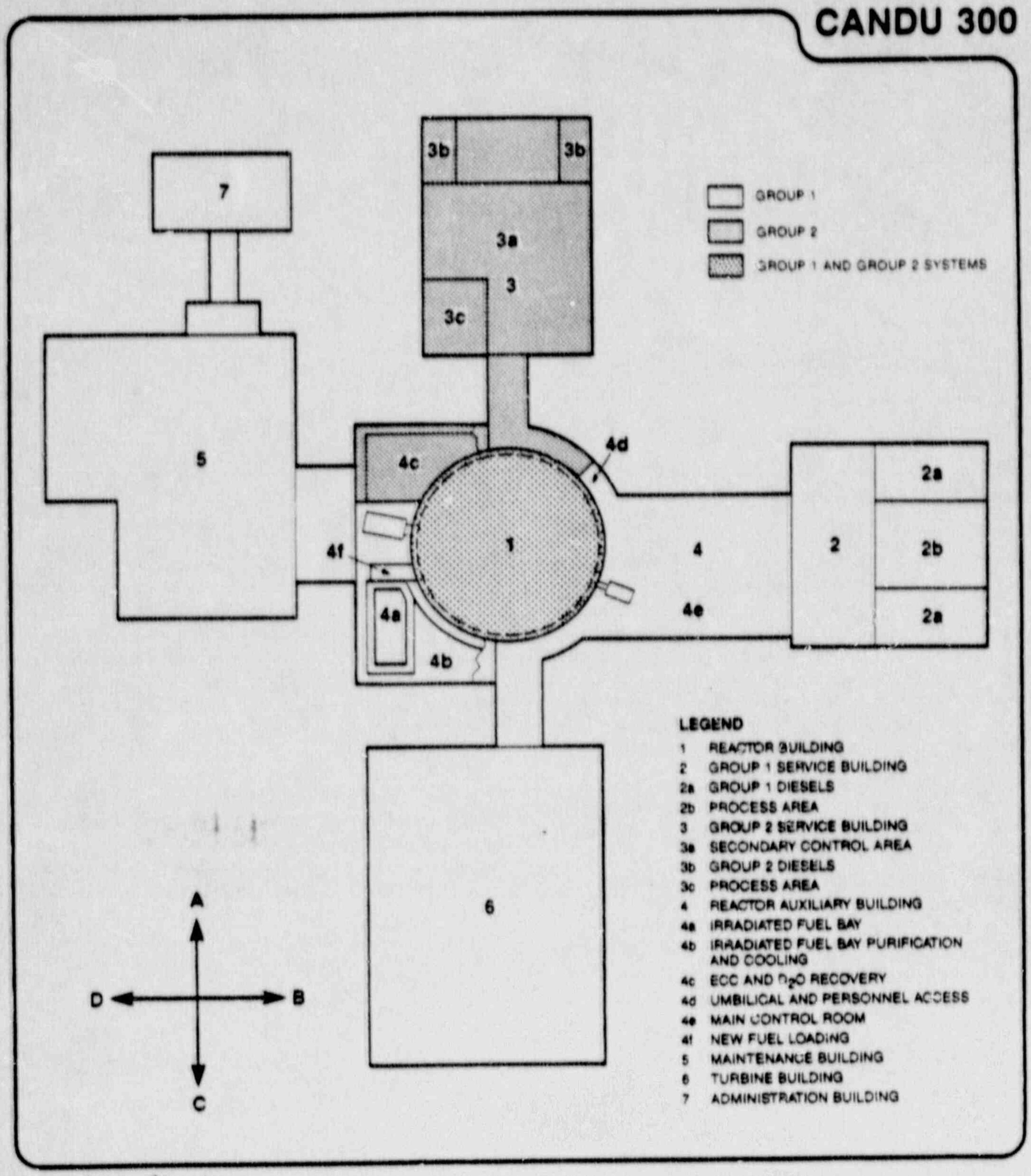


Figure 2.1-1 Site plan

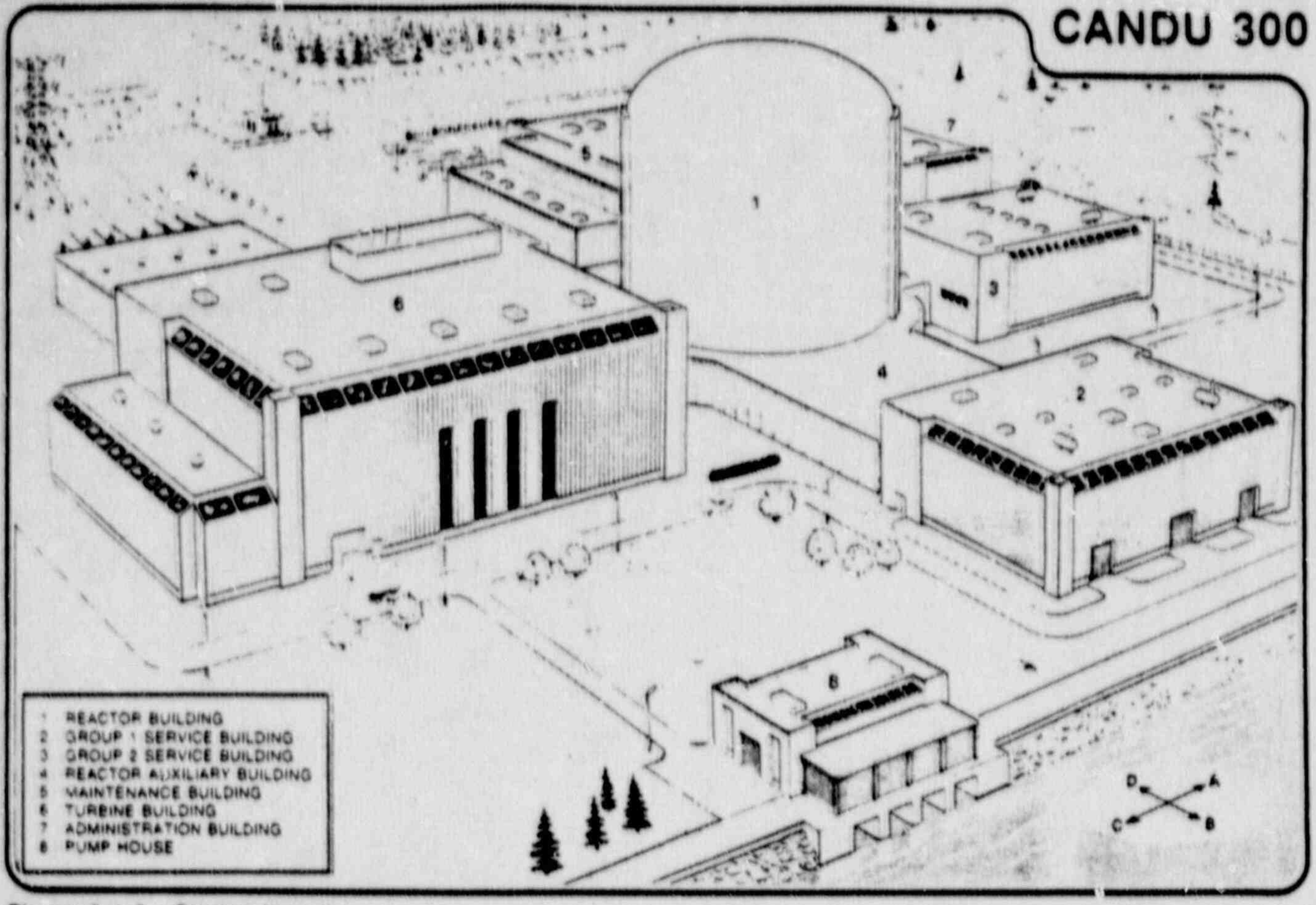


Figure 2.1-2 Station layout

The Turbine Building, for example, contains the services necessary to sustain operation of the turbine-generator unit including the water treatment plant. Significant connections from the Turbine Building to the other principal buildings are therefore limited to the steam and feedwater lines (to the Reactor Building), the electrical supply lines (to the Group 1 Service Building), and the turbine-generator control system wiring (to the Main Control Room).

The Main Control Room is located in the Reactor Auxiliary Building between the Reactor Building and Group 1 Service Building. The Secondary Control Room is located in the Group 2 Service Building. The enclosed passageway around the Reactor Building, which occupies the lower section of the Reactor Auxiliary Building provides personnel access around the Reactor Building, and connects to the Turbine Building, Maintenance Building, and Service Buildings as shown in Figure 2.1-1.

Implementation of the Two-Group approach (Refer to Section 2.2.7) is readily apparent in the CANDU 300 site plan illustrated in Figure 2.1-1. All Group 2 services, except for the Group 2 Raw Service Water System, are totally accommodated within the Group 2 Service Building and the Group 2 portion of the Reactor Auxiliary Building. These structures and all equipment within them are seismically and environmentally qualified. All Group 1 services are provided from the Group 1 Service Building or other Group 1 areas of the station. Except for the Main Control Room (which is seismically qualified to a sufficient extent to assure operator survival), and the irradiated fuel bay, the Group 1 areas are not seismically or environmentally qualified beyond local building code requirements.

This layout results from detailed study and review of station safety, constructability, maintainability, and operability. Specifically, the station layout maximizes safety and facilitates the CANDU Two-Group approach. This layout also shortens the construction schedule by simplifying, minimizing and localizing interfaces, by accommodating many contractors without interference, by eliminating construction congestion, by providing direct access to all areas, by providing flexible equipment installation sequences and by minimizing material handling requirements.

The layout also facilitates station operation and maintenance, and accommodates client, contractual and licensing requirements without significant design modifications.

2.2 Nuclear Design

2.2.1 General

The design of the CANDU 300 reactor core closely follows that of the larger CANDU reactors. The core design incorporates the standard geometrical arrangement of horizontal fuel channels in a square lattice, which leads to a very well thermalized neutron spectrum. The neutronic characteristics of the CANDU 300 and CANDU 600 reactors are similar.

Standard CANDU fuel, shown in Figure 2.2-1, is used. It consists of 37 elements of uranium dioxide sheathed in Zircaloy and held together as a bundle by end-plates. There are 12 fuel bundles in each channel. Fuel and coolant are enclosed in standard zirconium-niobium pressure tubes which in turn are surrounded by Zircaloy calandria tubes (Refer to Section 3.3.3). High purity heavy water contained in the calandria vessel at low pressure and low temperature serves as the moderator. Pressurized heavy water is used in the heat transport circuit.

CANDU reactor core design using natural UO_2 fuel and D_2O moderator, is dedicated to maximum neutron economy and fuel utilization. As the fuel burns and U-235 is depleted, the buildup of plutonium provides additional reactivity and eventually contributes substantially to the energy production. Saturated and unsaturated fission products account for a very small fraction of the total neutron absorption. The heavy water moderator and high purity Zircaloy used for fuel sheath and structural components within the core also have very low neutron absorption. On-power fuelling obviates the need to carry excess reactivity to compensate for fuel depletion and permits continuous overall flux shaping to give an optimum power distribution. Therefore, the energy produced per U-235 nucleus introduced into the reactor is very high compared to reactors using ordinary water as moderator and coolant.

2.2.2 Core Configuration

The CANDU 300 employs the standard CANDU lat-

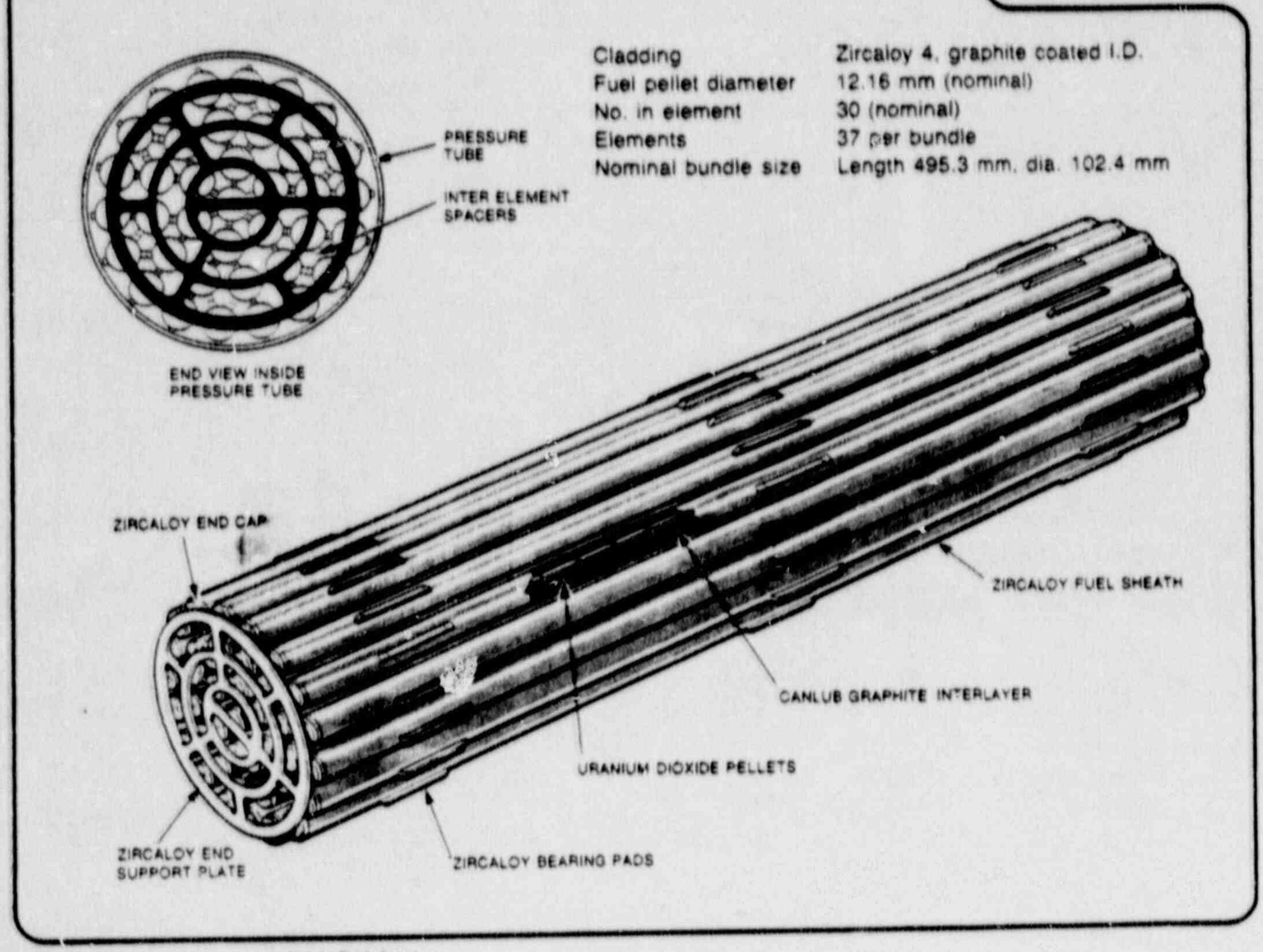


Figure 2.2-1 Standard CANDU fuel

tice design and fuel channel arrangement, with 232 fuel channels. Control and shutdown devices and in-core instrumentation are located in tubes perpendicular to the fuel channels and function in the low pressure, low temperature moderator portion of the core. Additional information on the reactor core is provided in Section 3.3; the key reactor core parameters are summarized in Table 4.

TABLE 4
CANDU 300 REACTOR CORE PARAMETERS

Number of fuel channels	232
Lattice pitch	28.6 cm
Reflector thickness	70 cm
Radial power form factor	0.82

2.2.3 Heat Transport System

The CANDU 300 Heat Transport System is essentially the same as one loop of the two loop CANDU 600 Heat Transport System. The major Heat Transport System equipment, including 2 steam generators and 2 heat transport pumps is of identical design to equipment now in service in CANDU stations. The equipment arrangement however, has been modified such that the steam generators and pumps are at opposite ends of the reactor core.

The optimum design parameters established for the CANDU 600 Heat Transport System are applicable to the CANDU 300. As a result, Heat Transport System conditions are similar. These are summarized in Table 5 and compared with those of other CANDU nuclear power stations in Figure 2.2-2. This figure also illustrates the modular nature of the CANDU reactor concept, and shows the systematic evolution of CANDU Heat Transport System design.

TABLE 5
CANDU 300
HEAT TRANSPORT SYSTEM CONDITIONS

Outlet header pressure	10 MPa(a)
Outlet header quality	4%
Nominal maximum channel flow	27 kg/s
Nominal maximum channel power	7.3 MW

2.2.4 Refuelling

Refuelling operations are carried out routinely on a semi-continuous basis with the reactor at power. The number of fresh fuel bundles introduced into a channel is variable and the bundle shuffling pattern along a channel is flexible. By adjusting the fuelling rate in various regions of the core the power distri-

CANDU 300

		Fuel channels		Heat Transport System conditions				HT pumps			Steam generators			
	Net power output (MWe)	Number of fuel channels	Number of elements in fuel bundle	Number of loops	Outlet header pressure MPa(a)	Max channel flow kg/s	Outlet header quality %	Total	Operating	Motor rating per pump (kW)	Number	Area (m^2) per S/G	Integral preheater	Steam pressure MPa(a)
CURRENT STATIONS														
Pickering A	515	390	28	2	8.8	23	0	16	12	1420	12	1850	Yes	4
Pickering B	516	380	28	2	8.8	23	0	16	12	1420	12	1850	Yes	4
Bruce A* and B*	825	480	37	1	9.2	24	0.7	4	4	8200	8	2415	No	4.4
CANDU 600	665	380	37	2	10	24	4	4	4	6700	4	3200	Yes	4.7
Darlington	881	480	37	2	10	25.2	2	4	4	9400	4	4760	Yes	4.9
NEW STATIONS														
CANDU 300	450	232	37	1	10	27	4	2	2	9100	2	3900	Yes	4.7
CANDU 600	750	388	37	2	10	30	4	4	4	8500	4	3700	Yes	4.7

* Electrical equivalent

Figure 2.2-2 CANDU Heat Transport System comparison

3.0 NUCLEAR STEAM PLANT

3.1 Introduction

This section covers the Nuclear Steam Plant (NSP) which includes the Reactor Building, Reactor Auxiliary Building and the systems and equipment within these structures. The principal components, systems, equipment and controls associated with the reactor assembly, the moderator, the heat transport and auxiliary systems are described.

This section also describes the design aspects of the NSP related to fuel handling, steam and feedwater systems, electrical power systems, station instrumentation, control and safety systems.

The NSP services are described in Section 4.0.

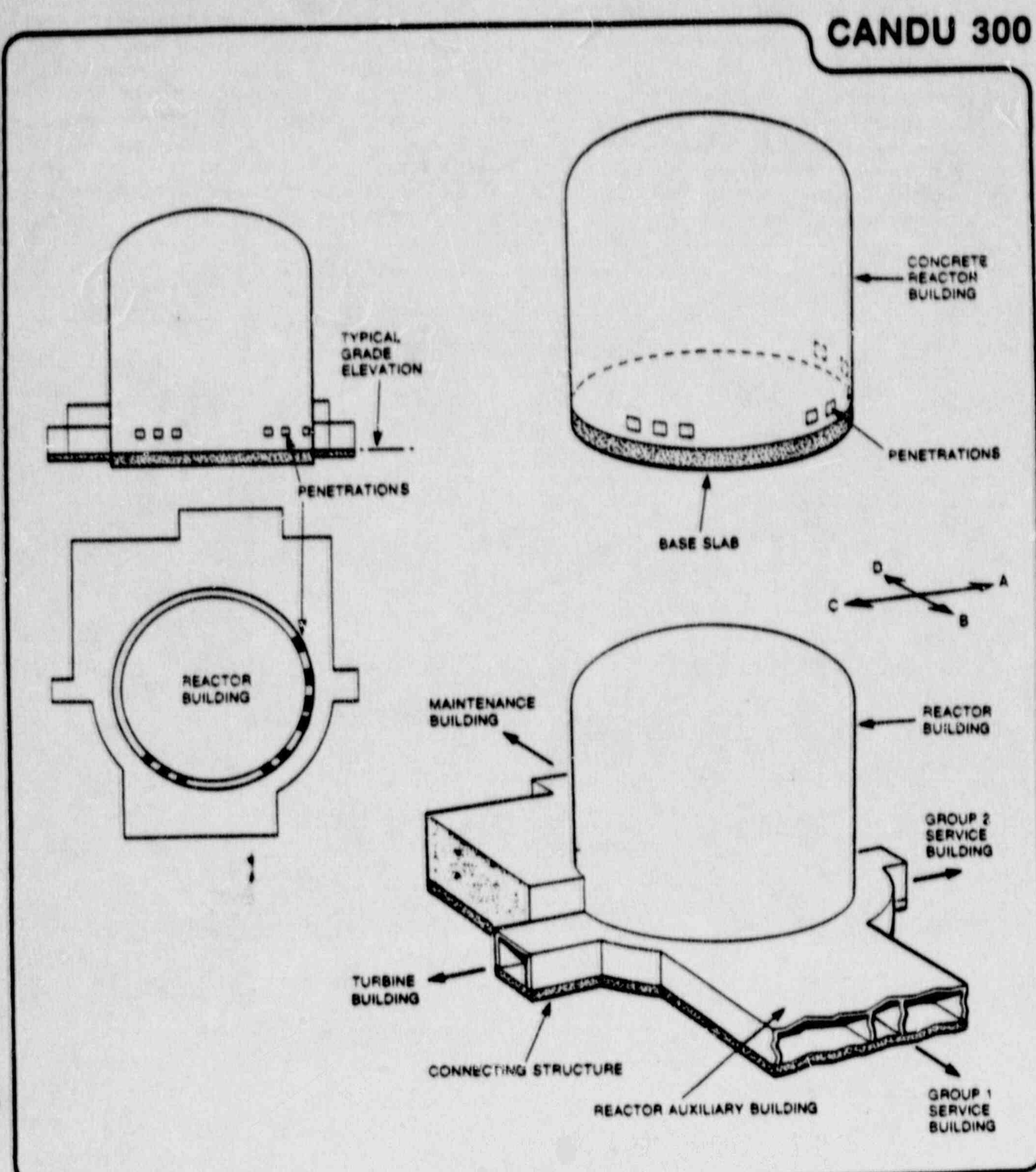


Figure 3.2-1 Reactor and Reactor Auxiliary Buildings

3.2 Buildings and Structures

3.2.1 Reactor Building

The Reactor Building is a reinforced concrete structure that features a full steel liner.

All umbilicals from the Reactor Building, ranging from cooling water lines to instrumentation and electrical supplies are grouped into a small number of large penetrations located near the base slab, thereby minimizing the number of penetrations in the containment structure.

The Reactor Building is encompassed by an annular structure, shown in Figure 3.2-1, referred to as the Reactor Auxiliary Building. The Reactor Auxiliary Building interfaces with the other principal buildings via enclosed passageways which accommodate the connecting umbilicals.

The Reactor Building, which provides an environmental boundary, a post-LOCA pressure boundary, and biological shielding, is a principal component of the containment system.

The Reactor Building perimeter walls are separate from the building internal structures. This provides flexibility in the building construction and eliminates any interdependence between the containment wall and the internal structures. The internal structures include the reactor vault walls, steam generator enclosure walls and heat transport pump support walls, the reactivity mechanism floor and intermediate floors. The internal structure and perimeter walls are shown in Figures 3.2-2, 3.2-3 and 3.2-4. Most floors provided for support of process equipment are of steel with steel grating.

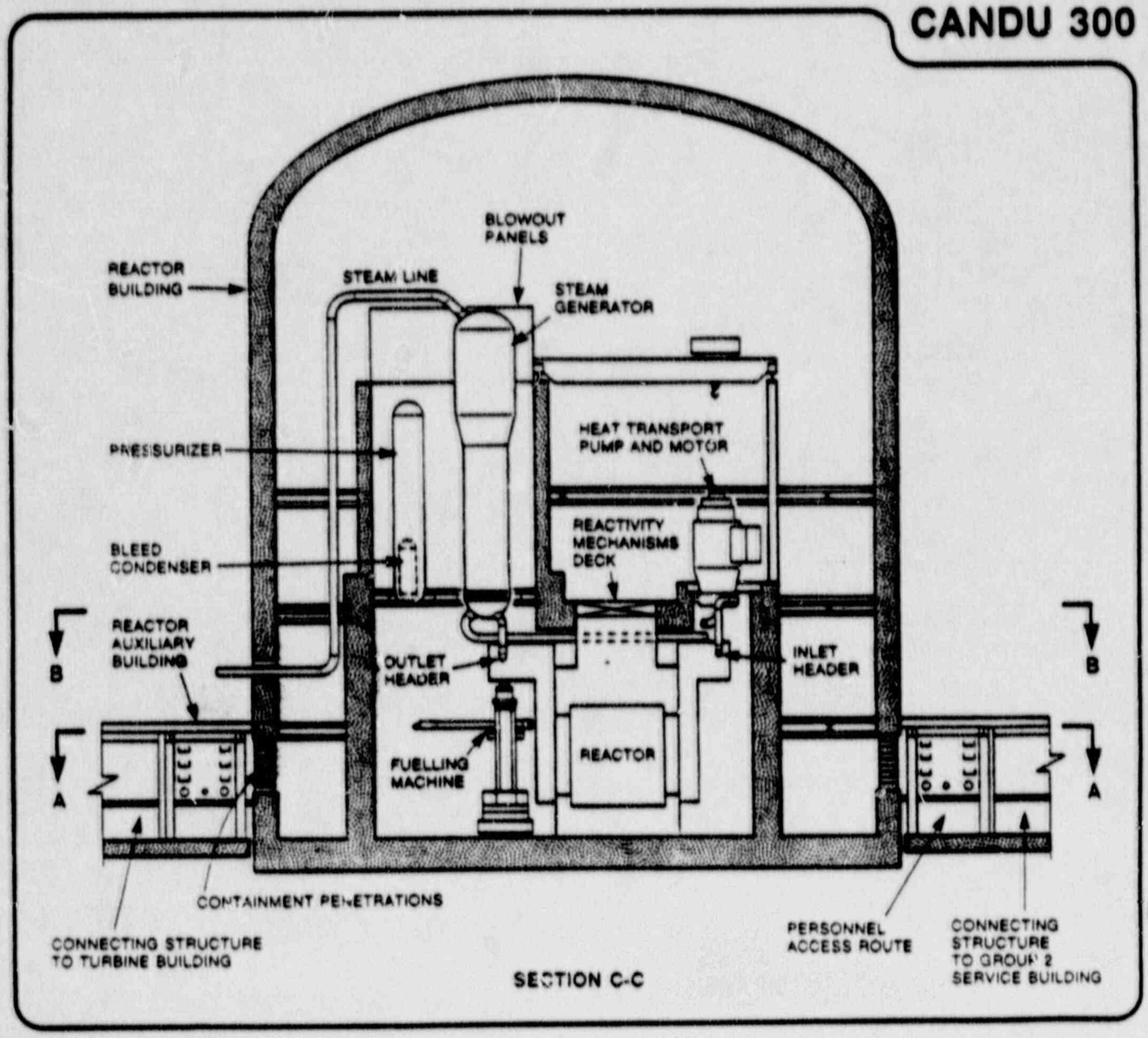


Figure 3.2-2 Reactor Building section

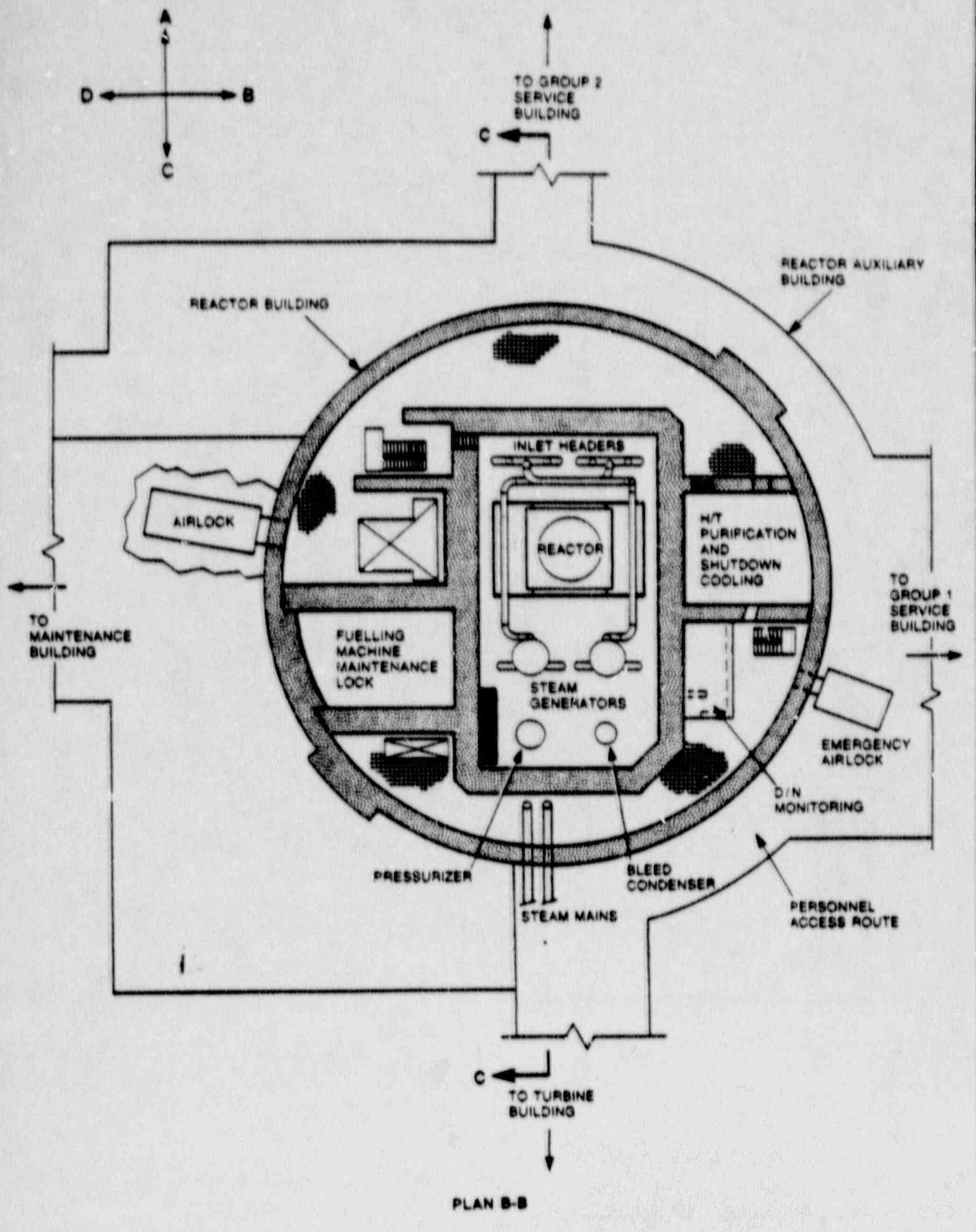


Figure 3.2-3 Reactor Building plan at elevation 115 m

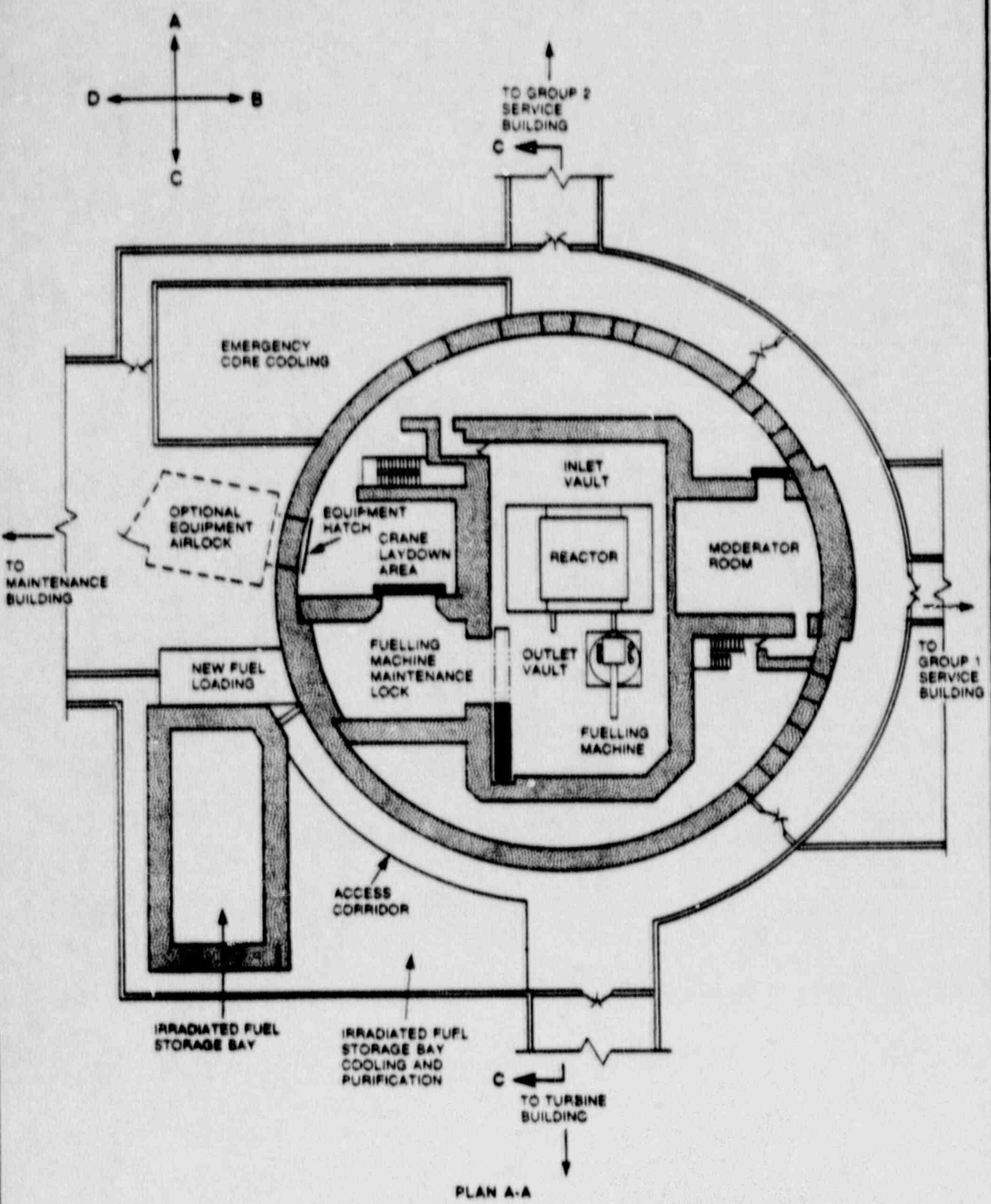


Figure 3.2-4 Reactor and Reactor Auxiliary Buildings at elevation 100 m

A large portion of the Reactor Building is accessible when the reactor is operating, facilitating on-power maintenance, inspection, and testing. Shielding for personnel from steam generator radiation fields is provided by the concrete steam generator enclosure walls. Similarly, the reactivity mechanism floor, a concrete slab extending over the shield tank assembly, provides a shielded working floor for maintenance personnel.

Entry into the containment structure is via an airlock, sized to accommodate all equipment and components required for routine maintenance. Major equipment enters or leaves the containment building via the larger equipment hatch or via the (optional) equipment airlock which replaces the equipment hatch.

Shielding doors located within the Reactor Building separate the accessible area and the reactor vault from the fuelling machine maintenance area.

Rooms containing potential heavy water leakage sources such as the fuelling machine and certain moderator system components, have controlled atmospheres. Doors within the Reactor Building have face seals as required to maintain isolation between the different reactor building atmospheres.

The Reactor Building crane, augmented by monorails and hoists, facilitates maintenance of equipment in the Reactor Building.

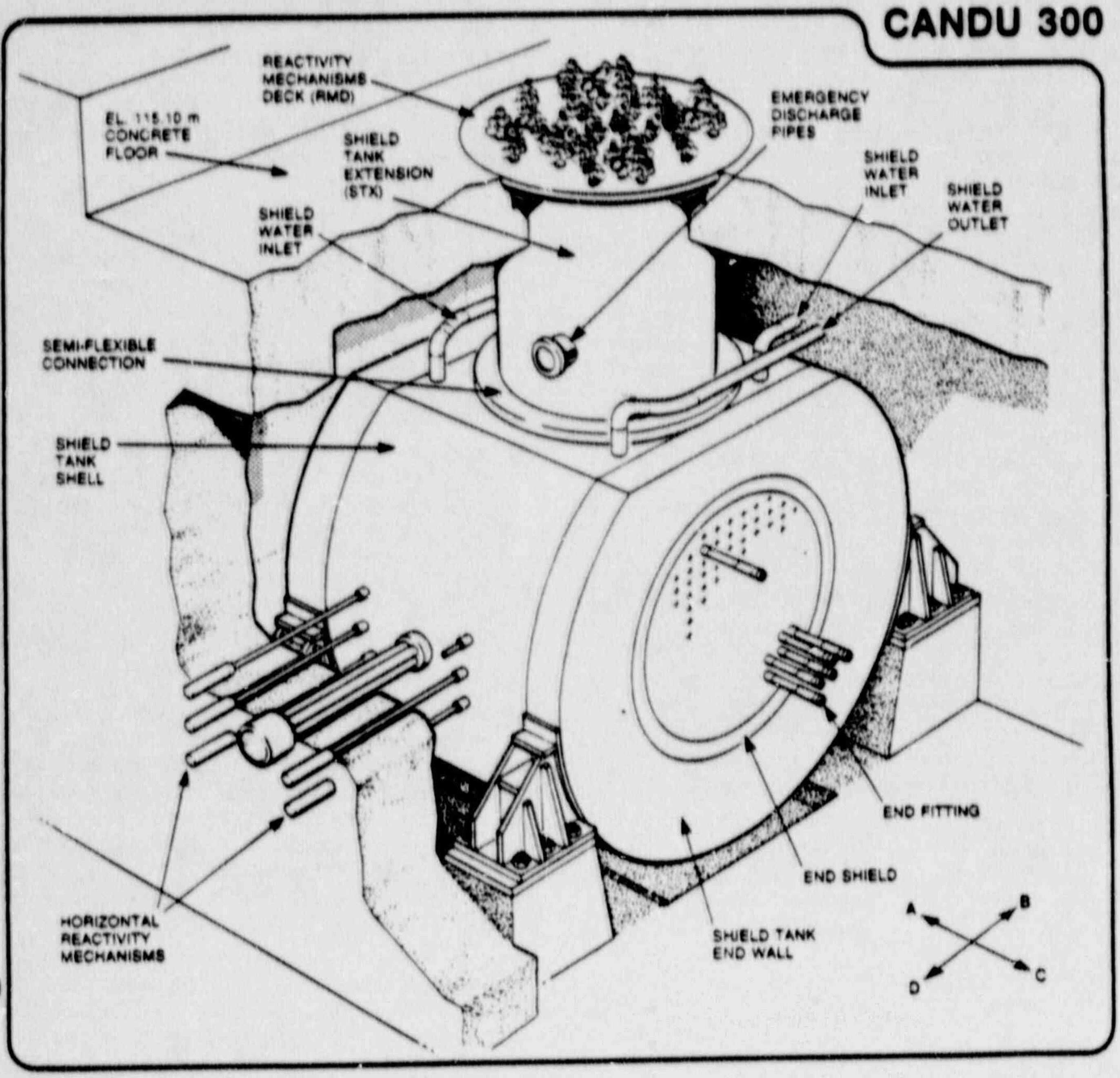


Figure 3.3-1 Reactor structures assembly

3.2.2 Reactor Auxiliary Building

The Reactor Auxiliary Building surrounds the Reactor Building and accommodates the umbilicals which run between the principal structures, the Main Control Room, and the irradiated fuel bay and associated fuel handling facilities. The Reactor Auxiliary Building layout is shown in Figure 3.2-4.

3.3 Reactor

3.3.1 General

The reactor assembly comprises the calandria vessel, end shields, fuel channel assemblies, the reactivity control units and the shield tank. The CANDU 300 calandria is shown in Figure 3.3-1.

All components of the reactor assembly except the pressure tube of the fuel channel assembly, including the reactivity mechanisms, function under low stress and at low temperatures. These factors, together with strict moderator chemistry control, preclude stress-corrosion cracking and all forms of erosion-corrosion.

3.3.2 Calandria and Shield Tank Assembly

The calandria is a horizontal cylindrical vessel consisting of the calandria shell, end shields and calandria tube/calandria tube extension portion of the fuel channels. The calandria contains the low pressure, low temperature, heavy water used to moderate the fast neutrons produced by fission.

The calandria vessel, in conjunction with the integral end shields, supports the horizontal fuel channel assemblies and the vertical and horizontal in-core reactivity control unit components. This entire assembly is integrally supported by the end walls of the shield tank.

The shield tank is a horizontal cylindrical structure, surmounted by the shield tank extension, which includes the reactivity mechanisms deck. The reactivity mechanisms deck is a box structure penetrated by fittings to accommodate the reactivity control unit components. It supports the upper ends of the vertical reactivity control units and is flexibly secured to the top of the shield tank.

The shield tank and end shields protect adjacent areas against radiation from the reactor. As a result, nuclear heat is generated within these shields. Additional heat is transferred to the end shields from the Heat Transport System by conduction from the fuel channels and from the calandria shell into the shield tank. The shield tank and end shields are cooled by the Recirculated Cooling Water System.

The reactivity mechanisms deck structure and the internal structure provide shielding to permit controlled access to external reactivity control unit actuators and connections during reactor operation.

3.3.3 Fuel Channel Assemblies

Each of the 232 fuel channel assemblies, illustrated in Figure 3.3-2, supports and locates 12 fuel bundles within the reactor core. Connected at each end to a feeder pipe, the fuel channel forms an integral part of the pressure boundary of the Heat Transport System. The Heat Transport System heavy water coolant flows around and through the fuel bundles in the fuel channel and removes the nuclear-generated heat.

The fuel channel assembly includes a pressure tube, a calandria tube, calandria tube extensions at each end of the calandria tube, end fittings at each end of the pressure tube and 4 garter springs which maintain the annular separation of the calandria tube and pressure tube. Each pressure tube is thermally insulated from the low-temperature, low-pressure moderator by the CO_2 filled gas annulus formed between the pressure tube and the calandria tube. The fuel channel assemblies are installed and replaced as a single factory assembled unit. The fuel channel inlet and outlet end fittings connect respectively to Heat Transport System inlet and outlet feeders. Reactor coolant flow through the reactor core is uni-directional. All outlet end fittings are therefore located at one end of the reactor, and all inlet end fittings at the opposite end of the reactor. The fuelling machine gains access to the fuel channel by removing the closure plug, latched spacer plug, and shield plug which are located in the outlet end fitting (Refer to Section 3.8.4).

The pressure tubes have a nominal diameter of 100 mm and a wall thickness of 4.2 mm. They are made of a cold-worked, zirconium-2.5% niobium alloy, which offers high strength, low neutron absorption and high corrosion resistance. The pressure tubes are the only components in a CANDU reactor subjected to a combination of high radiation, high stress, and high temperature.

3.3.4 Reactivity Control Units

The reactivity control units form the reactor sensor and actuator portions of the Reactor Regulating and Reactor Shutdown Systems. These comprise reactor power measuring devices, neutron absorbing reactivity control and shutdown devices, and the liquid injection nozzles of Shutdown System 2 (SDS2). The shutdown systems are independent of the regulating system, and of each other. An example of a reactivity control unit is shown in Figure 3.3-3.

The vertical reactivity control units for both the Reactor Regulating System and Shutdown System No. 1 (SDS1) are installed through the reactivity mechanisms deck. The horizontal reactivity control units are installed through the shield tank side wall and are primarily dedicated to Shutdown System No. 2.

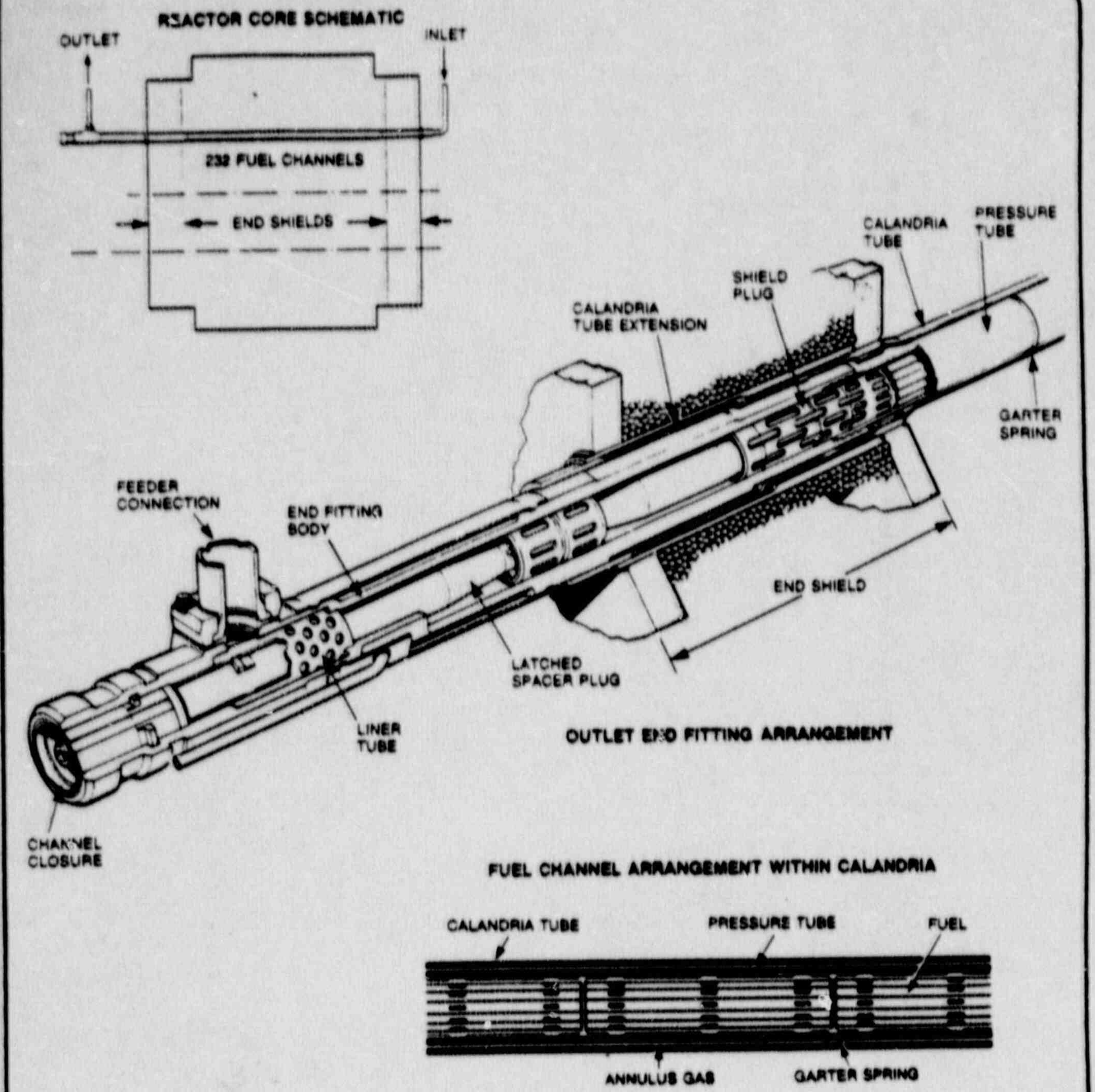


Figure 3.3-2 Fuel channel assemblies

Generally, reactivity control unit sensors or absorbers are housed within Zircaloy tubes that span the calandria between the rows of fuel channels, both horizontally and vertically.

Zone control units are located in vertical assemblies for reactivity adjustment during normal reactor operation (Refer to Section 3.11.4).

3.4 Moderator System

3.4.1 General

Neutrons produced by nuclear fission are moderated by the heavy water in the calandria. The heavy water is circulated through the moderator system for cooling, purification and control of the concentration of soluble poisons used for reactivity adjustment.

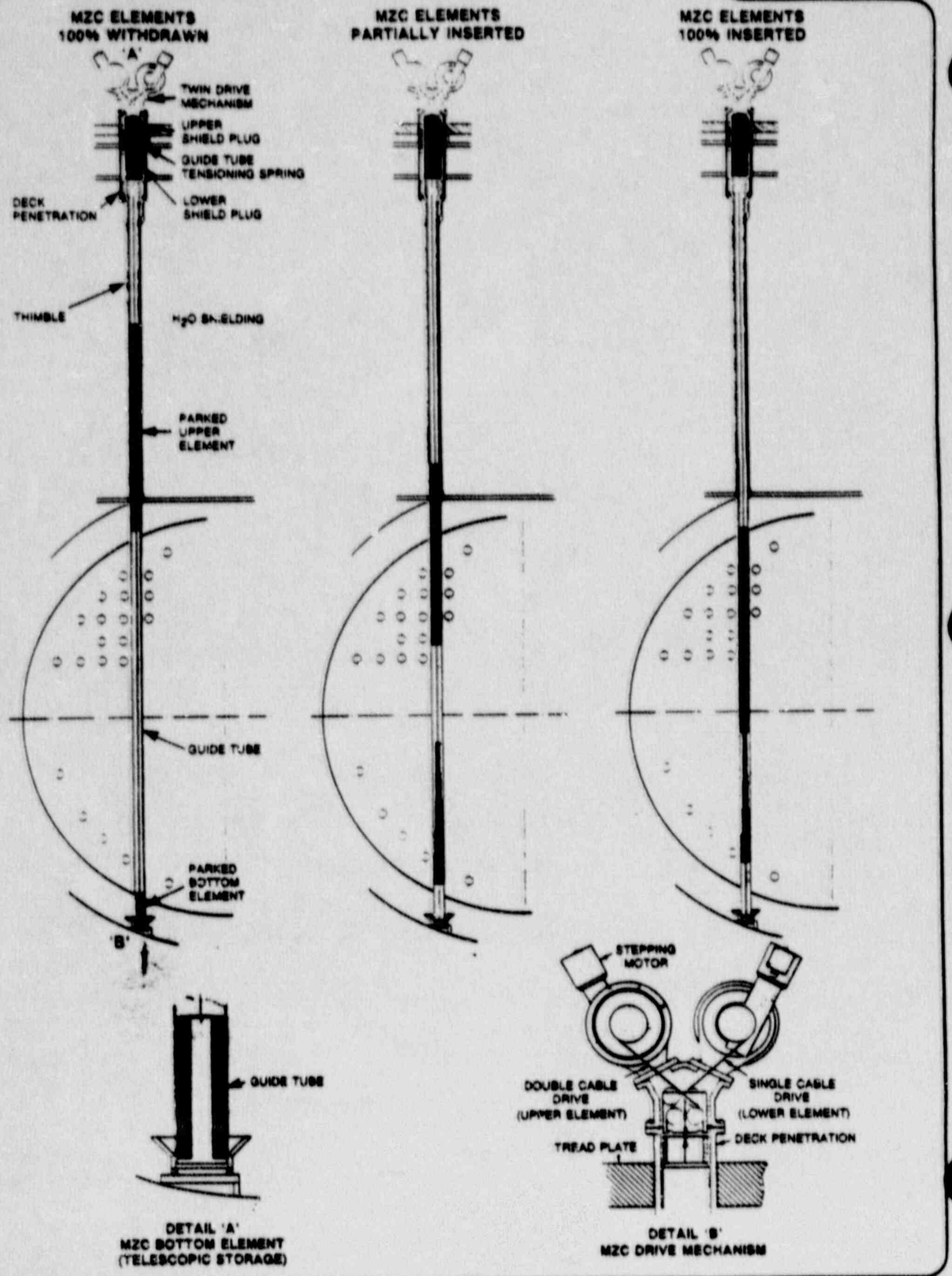


Figure 3.3-3 Reactivity control units

3.4.2 Moderator System

The Moderator System, illustrated in Figure 3.4-1, is fully independent from the Heat Transport System. The Moderator System includes 2 x 50% pumps and 2 x 50% plate type heat exchangers. The Moderator System is connected to the Moderator Purification System (Refer to Section 3.4.3) and the D_2O sampling system, and to the D_2O Supply System to enable moderator D_2O to be transferred to and from D_2O Supply System. The calandria extension maintains the moderator level in the calandria within the required range by accommodating moderator swell and shrink resulting from temperature fluctuations.

Potential heavy water leak sources (such as valve stem seals and mechanical joints) are kept to a minimum by using welded construction, seal welding, and bellows seals wherever practicable. All potential leak sources in the Moderator System are connected to the D_2O Collection System.

The reliability of the Moderator System is assured by careful design and appropriate component, instrument and power supply redundancies.

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss of Heat Transport System coolant accident coincident with failure of emergency core cooling. The capacity of the heat sink is assured by controlling the heavy water temperature in the calandria at a low value.

The materials in contact with the heavy water moderator are summarized in Table 7.

TABLE 7
MODERATOR SYSTEM MATERIALS

Calandria	Stainless steel
Calandria tubes	Zircaloy 2
Moderator heat exchangers	Stainless steel
Moderator pumps	Stainless steel
Moderator piping and valves	Stainless steel

3.4.3 Moderator Auxiliary Systems

The Moderator is complemented by several subsystems, described below, which are designed to maintain operating parameters within optimum range.

- The Moderator Purification System maintains the purity of the heavy water and minimizes corrosion of components and crud activation by controlling the pD (pH) and by removing impurities present in the D_2O. The Moderator Purification System is also used to adjust the concentration of the soluble poisons boron and gadolinium (which have large neutron capture cross-sections) in response to reactivity demands and to remove the soluble poison, gadolinium, after initiation of the Shutdown System 2 (SDS2). The system consists of a filter and ion exchangers.

- The Moderator Cover Gas System controls the concentration of deuterium gas by catalytically recombining the deuterium and oxygen gases resulting from the radiolysis of the heavy water moderator to reform heavy water. Helium, which is chemically inert and not activated by neutron irradiation, is used as the cover gas for the Moderator System.
- The Liquid Poison System adds negative reactivity to the moderator when required for reactivity adjustments, and also provides neutron poison in the moderator to preclude criticality during certain reactor shutdown conditions. The liquid poisons employed are boron as boric anhydride, and gadolinium as gadolinium nitrate, dissolved in D_2O.
- The Moderator D_2O Collection System collects any heavy water leakage from the moderator and associated systems and pumps it into the heavy water management systems for cleanup and upgrading.

3.5 Heat Transport System

3.5.1 General

The Heat Transport System circulates pressurized heavy water through the reactor fuel channels to remove heat produced by fission of natural uranium fuel. The heat is carried by the reactor coolant to the steam generators where it is transferred to ordinary water to produce steam. This steam subsequently drives the turbine generator or, alternatively, is provided to process users.

The principal performance features for the Heat Transport System and associated systems are as follows:

- Circulation of the reactor coolant (heavy water) through the reactor fuel channels, is maintained at all times during reactor operation, shutdown and maintenance.
- Heat Transport System pressure is controlled for all normal modes of operation by the Pressure and Inventory Control System.
- The Heat Transport System is protected from overpressure by instrumented relief valves and reactor regulating and/or safety system action.
- Heat Transport System coolant inventory is controlled for all normal modes of operation by the Pressure and Inventory Control System.

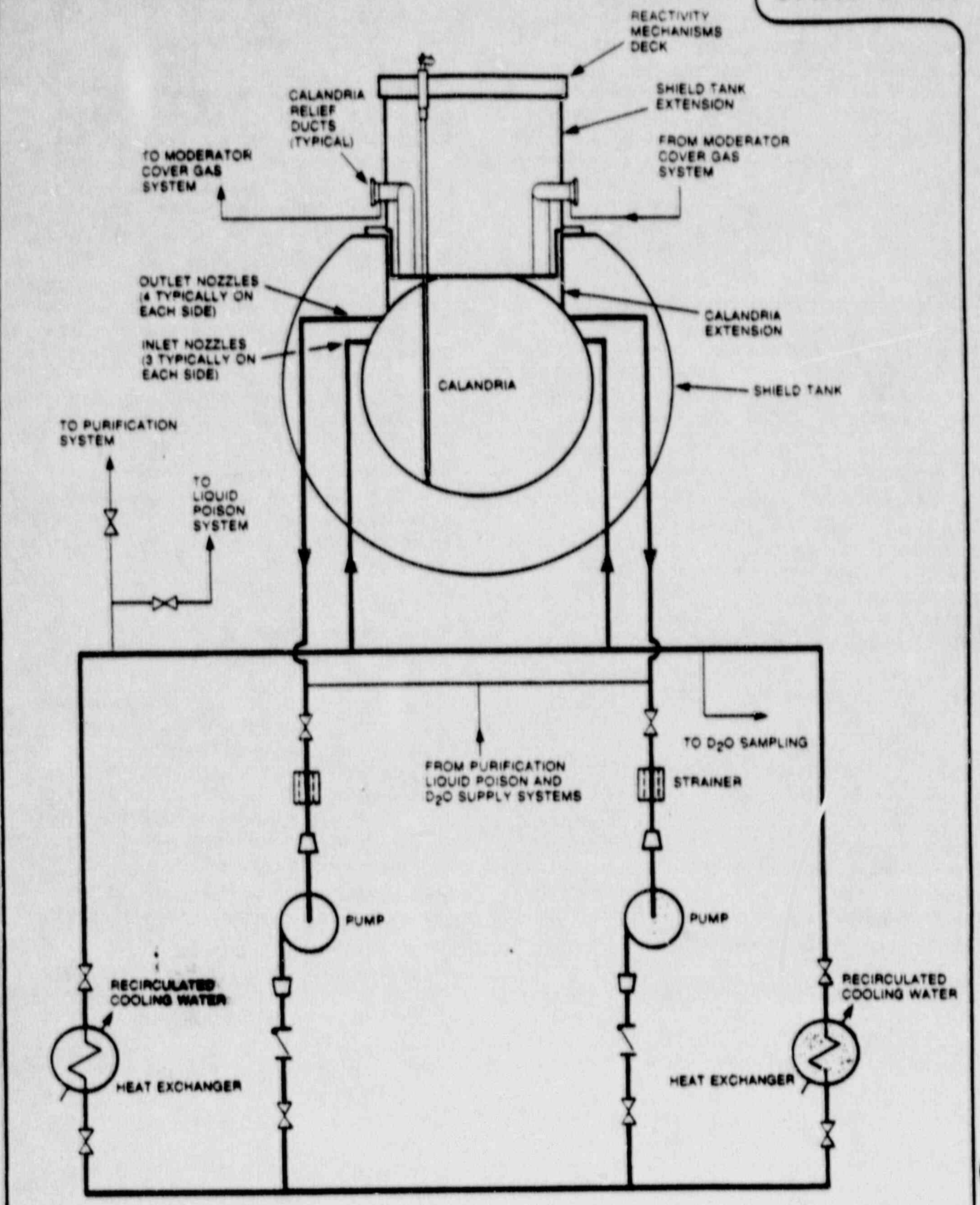


Figure 3.4-1 Moderator System

- The Shutdown Cooling System, capable of operation at full Heat Transport System temperature and pressure can be used to remove reactor shutdown heat. This system also permits the draining of pumps and steam generators in the Heat Transport System for maintenance.
- Purification by filtering, ion exchange and degassing, and chemical addition maintains the chemistry and purity of the reactor coolant.
- The Emergency Core Coolant System supplies light water to the Heat Transport System in the unlikely event that reactor coolant is lost from the Heat Transport system due to a pipe rupture.
- Heavy water leak sources are minimized by using welded construction and bellows-sealed valves where practicable. Potential leak sources are connected to collection and recovery systems.

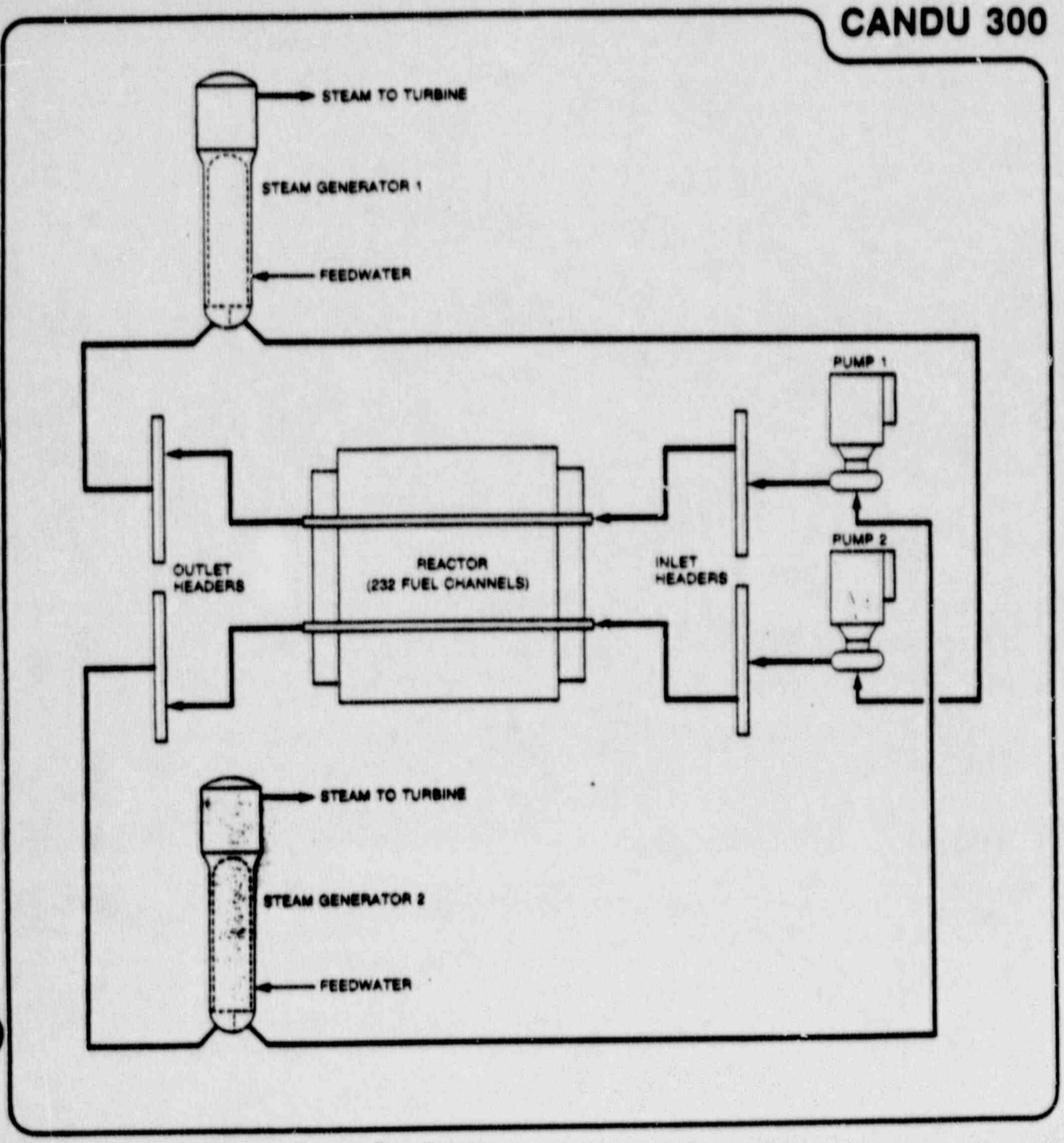


Figure 3.5-1 Heat Transport System flow diagram

3.5.2 Heat Transport System

The major components of the CANDU 300 Heat Transport System, illustrated in Figure 3.5-1, are the reactor fuel channel pressure tubes, two steam generators, two electrically driven pumps, reactor inlet and outlet headers, and the interconnecting piping. Heavy water is fed to the fuel channels from the inlet headers at one end of the reactor and is returned to the outlet headers at the opposite end of the reactor.

Each steam generator is connected to a heat transport pump on the opposite end of the core by a single pump suction line. Each heat transport pump has two discharge pipes each of which connects to a reactor inlet header. Similarly, each steam generator has two inlet pipes, each of which connects to a reactor outlet header.

The coolant flow in the CANDU 300 Heat Transport System is in the "figure of 8" pattern employed in all CANDU reactors, with the heat transport pumps in series and the coolant making two core passes per cycle. The equipment arrangement however, with the steam generators and pumps at opposite ends of the reactor, results in uni-directional coolant flow through the core. This facilitates fuelling from one end of the ruactor with a single fuelling machine. The header arrangement results in reactor fuel channels in each vertical half of the core belonging to the same core pass.

The arrangement of Heat Transport System equip-

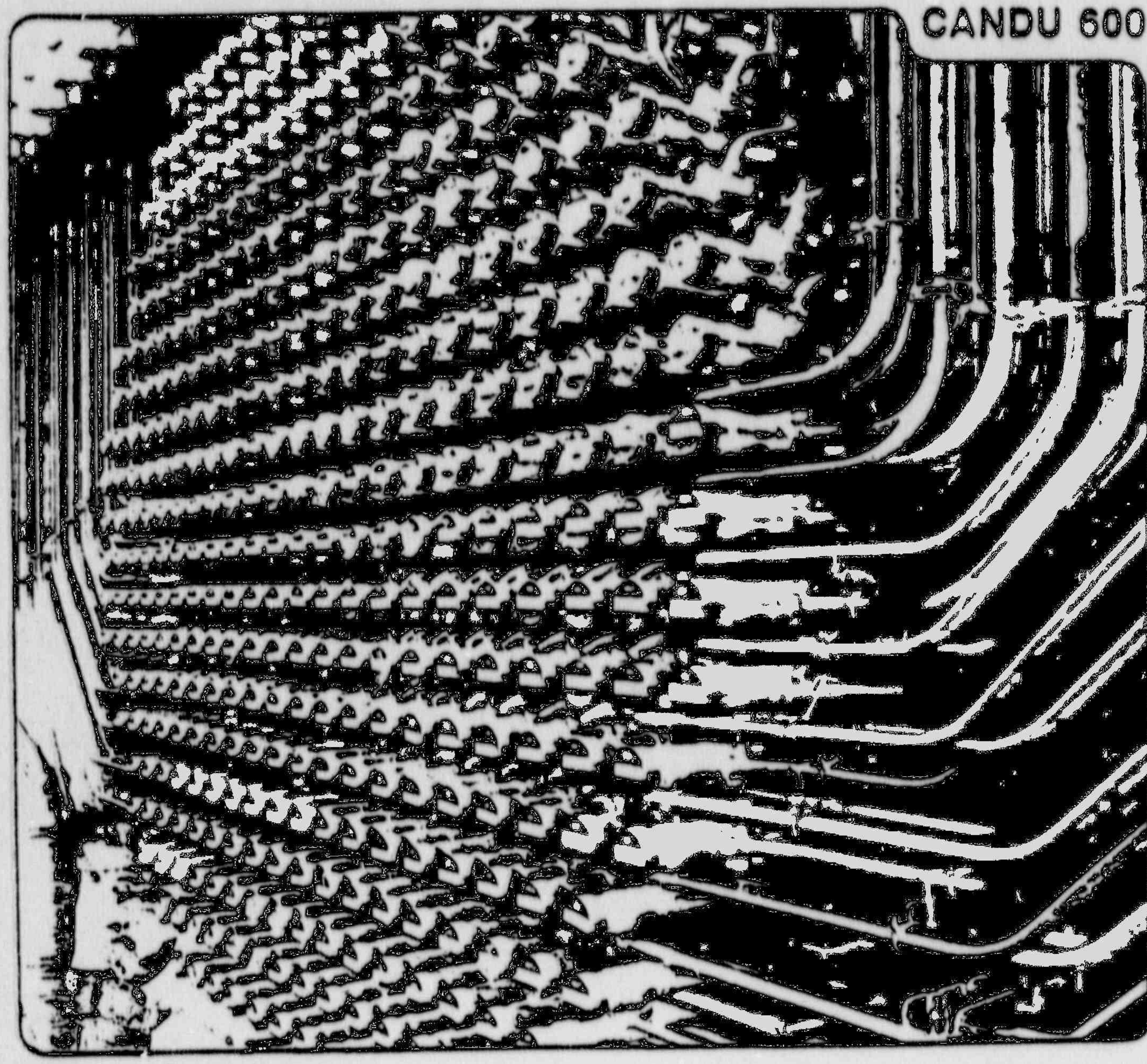


Figure 3.5-2 Reactor face

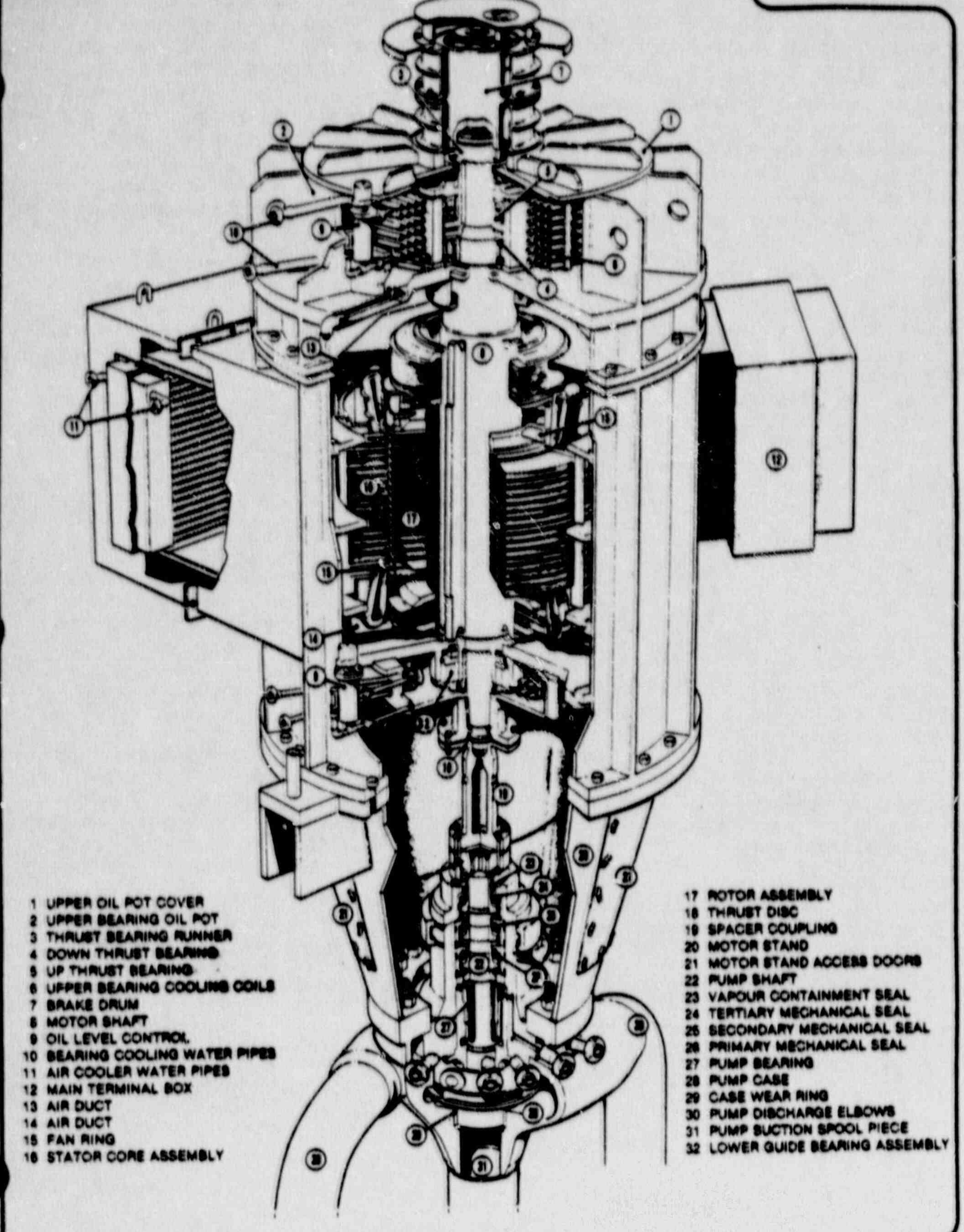


Figure 3.5-3 Heat Transport Pump and motor assemblies

ment within the Reactor Building is shown in Figures 3.2-2 and 3.2-3. A photograph of a CANDU 600 reactor face showing the outlet end-fittings and feeder arrangements is presented in Figure 3.5-2.

The headers, steam generators and pumps are located above the reactor. This ensures natural coolant circulation on loss of power to the heat transport pumps. This configuration also allows the pumps and steam generators to be drained for inspection and maintenance while reactor shutdown heat removal is maintained by the Shutdown Cooling System.

Key heat transport parameters are presented in Tables 2 and 5.

The CANDU 300 heat transport pump/motor assemblies, shown in Figure 3.5-3 are of the same design as those of the CANDU 600. Each pump is driven by a vertical, totally enclosed, air-water cooled squirrel cage induction motor. A removable shaft coupling connects the motor to the pump. Removal of the coupling allows space for the pump seals and bearings to be removed without removing the motor.

The shaft sealing arrangement consists of three mechanical seals and one back-up seal in series. Each mechanical seal is designed to withstand the full differential pressure. All three seals are housed in a removable cartridge located in the upper part of the stuffing box.

The Gland Seal System supplies cooled and filtered D_2O for lubricating and cooling the mechanical seals of the heat transport pumps. The cool D_2O is supplied by the heavy water feed pumps in the Pressure and Inventory Control System (Refer to Section 3.5.3). A back-up supply is provided from within the pump casing in the event that the D_2O feed pumps are unavailable.

Shielding is installed between the pump casing and the pump motor to protect personnel engaged in pump and motor maintenance and other maintenance tasks above the pump casing.

Two steam generators transfer heat from the reactor coolant (D_2O), contained on the steam generator primary side, to raise the temperature of and evaporate light water (H_2O) on the secondary side. The CANDU 300 steam generators, which are of the same design as those of the CANDU 600 consist of an inverted vertical U-tube bundle in a cylindrical shell. Steam separating equipment is provided in the steam drum at the upper part of the shell. A typical CANDU steam generator is illustrated in Figure 3.5-4. A photograph of a steam generator in shipment is shown in Figure 3.5-5.

Feedwater enters the baffled preheater section of the steam generator, and flows over the D_2O outlet end of the U-tube bundle. Water at saturated temperature from the preheater section mixes with recirculating water flowing over the hot leg section of the tube bundle.

The steam-water mixture rising from the upper end of the U-tube bundle passes through cyclone steam separators. The separated water recirculates to the tube bundle through a shrouded annulus, and the steam, with less than 0.25% moisture by weight, leaves the steam generator through the steam outlet nozzle.

High recirculation ratios and relatively low heat flux, in combination with comprehensive chemistry control, material specifications and detailed attention to design have assured long life and low maintenance requirements for CANDU steam generators. CANDU steam generator tube failure rates are two orders of magnitude lower than the industry average.

CANDU Heat Transport Systems make extensive use of carbon steel, which is relatively ductile, easy to weld, and easy to inspect. Specific requirements are imposed on Heat Transport System materials to limit corrosion and reactivity transport (low cobalt content for example).

The materials utilized in CANDU Heat Transport Systems are listed in Table 8. Similar materials are utilized in Heat Transport Auxiliary Systems.

TABLE 8
HEAT TRANSPORT SYSTEM MATERIALS

Steam generator	• Head	Carbon steel
	• Tubing	Incoloy 800
	• Tube sheet	Inconel clad carbon steel
Pump	• Casing	Carbon steel
	• Impeller	Stainless steel
	• Shaft	Stainless steel
Piping	• Feeders	Carbon steel
	• Headers	Carbon steel
	• Other	Carbon steel
Fuel channel	• End fittings	Stainless steel
	• Pressure tube	Zircaloy 2½% Niobium

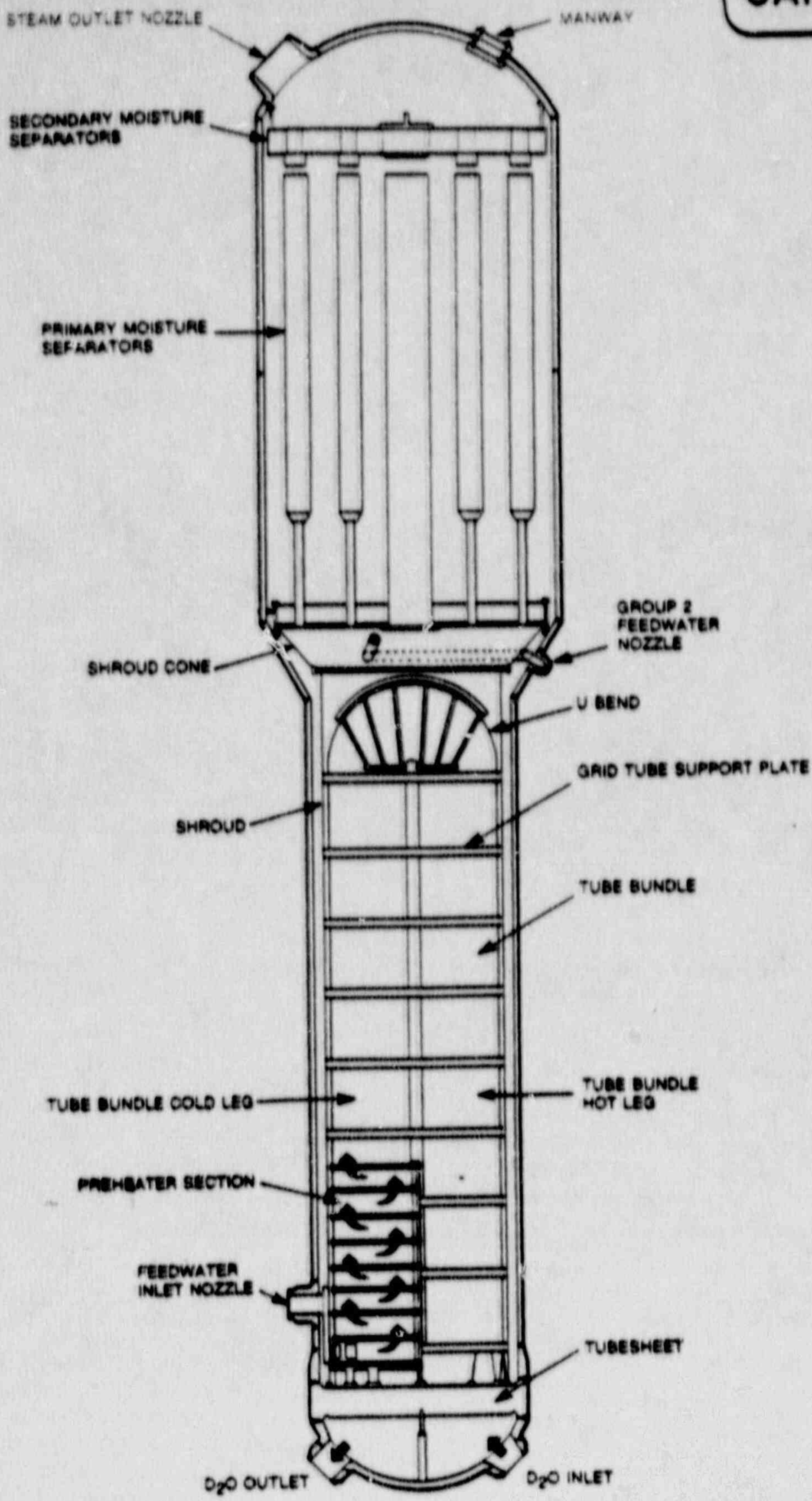


Figure 3.5-4 Steam Generator

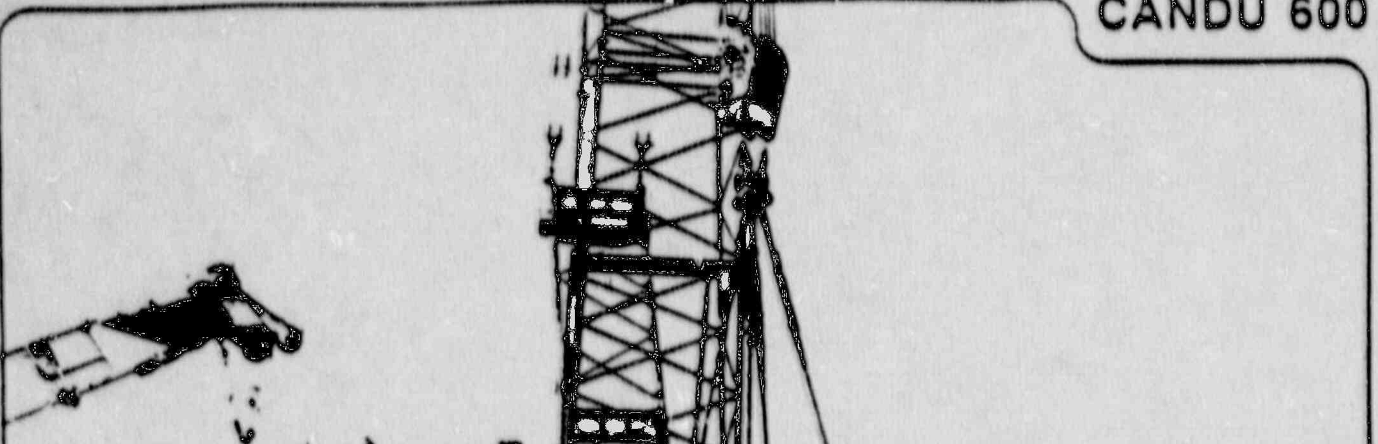


Figure 3.5-5 Steam Generator during shipment

3.5.3 Heat Transport Auxiliary Systems

The Heat Transport System is complemented by several auxiliary systems which support its operation and maintain variable parameters within optimum ranges to suit the various system functions. These systems are discussed below.

The Pressure and Inventory Control System provides pressure and inventory control and overpressure protection for the Heat Transport System.

The principal system functions are to:

- Control Heat Transport System pressure over the full range of Heat Transport System and reactor operating modes.
- Limit Heat Transport System pressure increases and/or decreases caused by transients to acceptable values.
- Accommodate Heat Transport System coolant thermal expansion and contraction associated with warm-up, cooldown, and power manoeuvring.
- Control Heat Transport System inventory over the full range of Heat Transport System and reactor operating modes.
- Provide a means of degassing the Heat Transport System coolant.

The Pressure and Inventory Control System, illustrated in Figure 3.5-6, includes a pressurizer, a bleed condenser, feed pumps and control valves.

The pressurizer, which controls the pressure in the Heat Transport System during operation at power, is a cylindrical pressure vessel installed vertically and connected to one of the reactor outlet headers. Pressure in the pressurizer is controlled by adding heat with electrical immersion heaters, or by condensing steam in the vapour space by D_2O spray flow. The feed pumps provide heavy water via the feed valves to maintain inventory in the Heat Transport System.

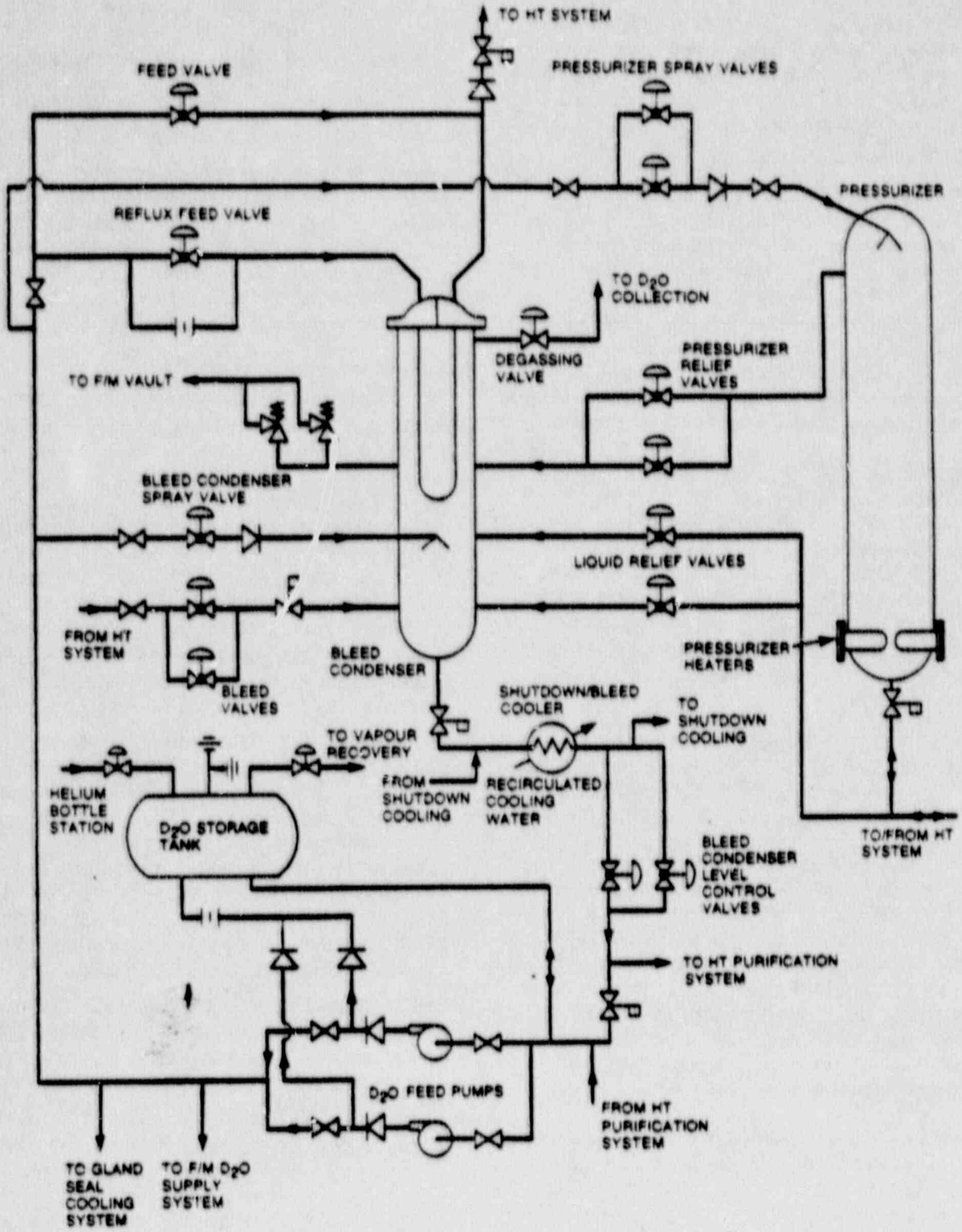


Figure 3.5-6 Pressure and Inventory Control System

instrumented liquid relief valves, in conjunction with the reactor shutdown systems provide overpressure protection of the Heat Transport System. The liquid relief valves are connected to a reactor outlet header and discharge into the bleed condenser. The bleed condenser is equipped with spring-loaded safety relief valves. These valves are designed in conjunction with the heat transport relief valves to assure that Heat Transport System overpressure protection requirements are met even when the two sets of valves operate in series.

The relief pressure for the bleed condenser relief valves is set above the normal operating pressure of the reactor outlet headers. The Heat Transport System relief valves, and all other valves that permit outflow of coolant from the Heat Transport System discharge into the bleed condenser; therefore, in the event that any valve permitting discharge from the Heat Transport System fails in the open position, heavy water outflow from the Heat Transport System is limited, and retained in the bleed condenser.

The bleed flow from the Heat Transport System is discharged into the bleed condenser as two-phase steam and liquid. The steam is condensed by a tube bundle located inside the bleed condenser using the feed flow for cooling. The bleed flow is further cooled by the shutdown/bleed cooler before passing through the Heat Transport Purification System. The shutdown/bleed cooler operates as a shutdown cooler for decay heat removal during reactor shutdown.

D_2O inventory is transferred into and out of the Heat Transport System via the D_2O storage tank which is connected to the D_2O supply system. This tank also serves as a head tank to the D_2O feed pumps.

The Heat Transport Purification System limits the activity and corrosion product buildup in the reactor coolant by removing soluble and insoluble impurities and by maintaining the pD (pH) of the D_2O at the required value. Coolant conductivity is also controlled. This minimizes the contribution made by activated corrosion products to radiation fields and hence radiation dose to operating personnel. Sampling points at various locations enable the system performance to be evaluated.

The system includes on-line filters and ion exchange columns containing appropriate ion exchange resins. Hydrogen is added via the system to suppress oxygen generated from radiolysis of D_2O.

Fission product releases to the Heat Transport System are very low due to the excellent performance of CANDU fuel and the ability to detect and remove any failed fuel utilizing the on-power refuelling equipment.

The Heat Transport Purification System does not have a reactivity control function. Therefore, no chemicals are added to the Heat Transport System for reactivity control.

The D_2O Collection System is a closed system which collects heavy water leakage from mechanical components, and receives heavy water drained from equipment prior to maintenance. The collected heavy water is returned to the heat transport heavy water storage tank by means of a small pump.

The Heavy Water Recovery system recovers heavy water from small Heat Transport System leaks and returns it to the Heat Transport System. This avoids the economic penalty of downgrading the Heat Transport System heavy water by the light water coolant injection from the Emergency Core Cooling System for small heat transport leaks.

3.5.4 Shutdown Cooling System

The Shutdown Cooling System removes decay heat following a reactor shutdown and cools the Heat Transport System to a temperature suitable for maintenance of the heat transport and auxiliary systems components. It is capable of cooling the Heat Transport System from full system pressure and temperature and of maintaining it at a low temperature for an indefinite period of time. It is also capable of providing reactor core cooling with the Heat Transport System coolant drained to the header level, to facilitate maintenance and inspection of steam generator and/or heat transport pump internals.

The Shutdown Cooling System consists of pumps and heat exchangers connected between the reactor inlet and outlet headers as shown in Figure 3.5-7. The pumps take coolant from the reactor outlet headers and return it via the heat exchangers to the reactor inlet headers. The pumps and heat exchangers are located at an elevation below the reactor headers so that net positive suction head (NPSH) is available for the pumps when the Heat Transport System is drained to the headers.

The Shutdown Cooling System is normally cold and depressurized, and isolated from the Heat Transport System by valves during reactor operation. The shutdown cooling pumps are powered by Group 2 power supply.

During normal operation, the shutdown/bleed cooler is cooled by the Recirculated Cooling Water System, and the shutdown cooler is cooled by the Group 2 Recirculated Cooling Water System.

3.6 Reactor Auxiliary Systems

3.6.1 Annulus Gas

Carbon dioxide gas is supplied at low pressure to the annuli between the pressure tubes and calan-

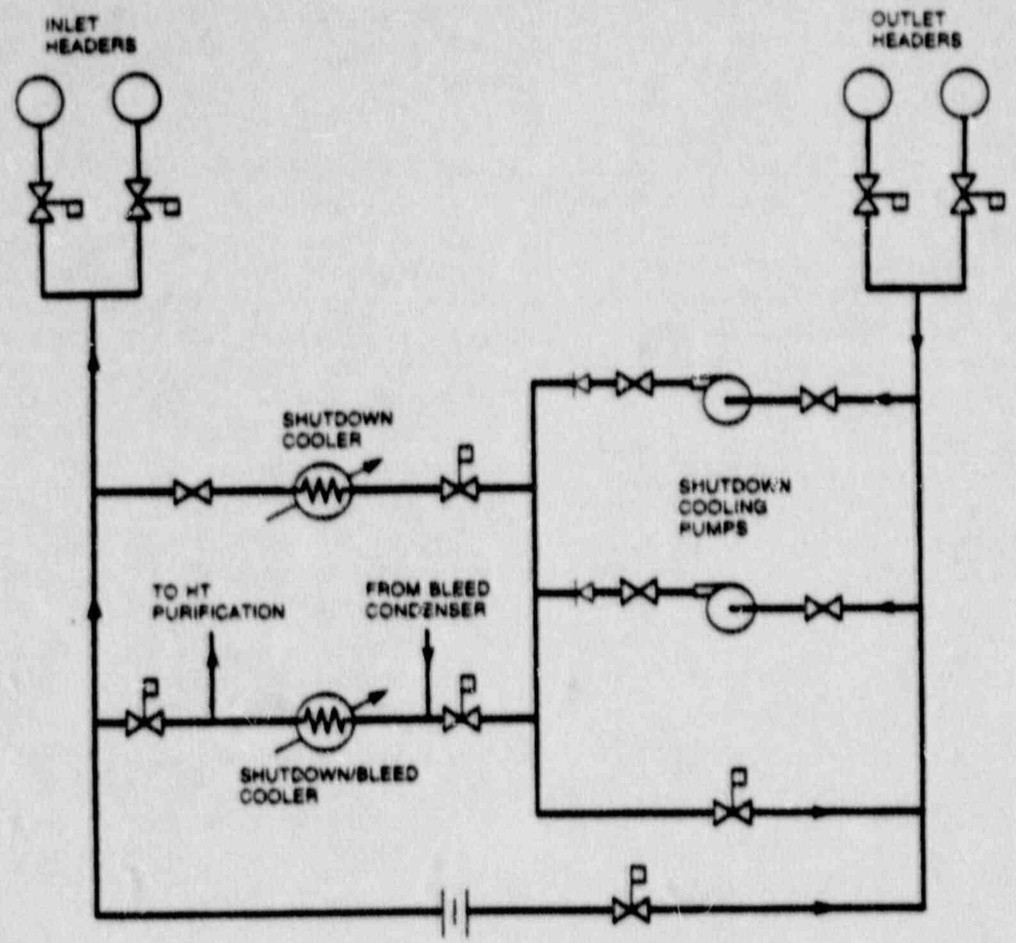


Figure 3.5-7 Shutdown Cooling System

dria tubes. The dry, gas-filled annulus prevents corrosion of fuel channel components and reduces heat transfer from the fuel channel pressure tube to the moderator.

Provision is made to promptly detect leakage from the Moderator or Heat Transport Systems into the annulus of any fuel channel by continuously monitoring the moisture content of the gas.

3.6.2 Failed Fuel Detection and Location

Very few fuel failures have occurred in operating CANDU stations. However, if a fuel failure does occur, it is detected and located by the delayed-neutron (DN) system. When used in the monitoring mode the DN system monitors a bulk sample from the Heat Transport System. In the location mode coolant samples from different channels are successively monitored until the channel containing the defected fuel is located.

The refuelling equipment is then used to remove the failed fuel. This essentially removes fission products as a source term for major component activation, and ensures that the activity of the CANDU Heat Transport System remains at very low levels.

3.7 Chemistry Control

3.7.1 General

The main objective of the various system chemistry control programs is to minimize corrosion, thereby preserving system integrity and, in the case of the reactor systems, maintain low radiation fields. The chemistry control requirements are established based on system materials and operating conditions. Chemistry control specifications are based on extensive research and over 150 reactor years of CANDU operating experience.

Chemistry control is provided for the Heat Transport System, the Moderator System, the steam generator secondary side, feedwater, and for auxiliary process systems such as recirculating cooling water, emergency core cooling water, moderator cover gas, main condenser cooling water and the irradiated fuel storage bay water.

3.7.2 Heat Transport System Chemistry

The Heat Transport System, as indicated in Table 8, uses zirconium alloys, AISI 400 series steel, carbon steel, and nickel alloys. The two major Heat Transport System coolant chemistry control requirements are the maintenance of low concentrations of dissolved oxygen, to ensure low rates of zirconium alloy corrosion and carbon steel pitting, and appropriate alkalinity to ensure acceptably low rates of carbon steel corrosion.

3.7.3 Moderator System Chemistry

The main moderator circuit materials as indicated in Table 7 are stainless steel and zirconium alloys. These materials are compatible with low temperature, slightly acidic water chemistry conditions. The primary emphasis of the moderator chemistry control program is to maintain the moderator water in a high state of purity thereby minimizing net radiolytic decomposition.

3.7.4 Secondary Side Chemistry

The prevention of steam generator tube failures is important in all indirect-cycle nuclear power plants. Thus, careful consideration is given to secondary side chemistry and the control of inadvertant additions of undesirable chemical species through system leakage. CANDU secondary side systems use All Volatile Treatment (AVT) and high quality make-up water, exclude alloys containing copper as a major constituent, have leak-tight condensers and optimize operating chemistry to minimize the transfer of corrosion products into the steam generators.

3.8 Fuel Handling System

3.8.1 General

The fuel handling facilities include equipment for storage of new fuel, for fuel changing and for storage of irradiated fuel. Reactor fuel is changed on a routine basis with the reactor operating at full power. Space and lifting facilities are provided in the irradiated fuel bay area for shipping irradiated fuel.

The fuelling machine that refuels the fuel channels at the reactor face, and the fuel transfer system that transfers new fuel into the Reactor Building and irradiated fuel out of the Reactor Building, are fully automated and are operated from the Main Control Room. The fuel handling equipment can be serviced with the reactor at power while the plant is in a non-refuelling mode of operation. Fuelling machine decontamination and service rooms are located in the Maintenance Building.

3.8.2 Fuelling Machine

A typical CANDU fuelling machine is shown in Figure 3.8-1. Heavy water, electrical power and control signals are supplied to the fuelling machine through a flexible catenary assembly which connects to stationary auxiliary systems. The latter are located outside the reactor vault and are accessible during refuelling operations.

3.8.3 Fuel Handling Control System

The Fuel Handling System uses dedicated Programmable Multiplexer Controllers communicating with each other via a serial data-highway. The controllers are distributed in the field at convenient locations close to the actuators/sensors they interface with. The controllers are capable of decision control logic and of implementing the control loops required by the various process requirements of the fuel handling system.

Computer driven colour CRT displays in the main control room enhance the man/machine interface and provide the capability for extensive presentation of information to the operator when required.

Extensive use is made of electric drives with state-of-the-art intelligent motion controllers for the control of mechanisms throughout the Fuel Handling System.

The automatic defuelling/refuelling sequences are controlled from a dedicated computer located in the Main Control Room area, with operator interfaces via keyboard and CRT display units.

3.8.4 Refuelling Procedure

The fuelling machine, by the combination of horizontal and vertical drives, on the fuelling machine carriage is aligned with the fuel channel to be refuelled. After the fuelling machine is aligned and clamped to a channel, the pressures in the machine and channel are equalized. The fuelling machine then removes the channel closure, the latched spacer plug, the downstream shield plug, and the irradiated fuel bundles from the fuel channel. The new fuel bundles and the desired irradiated fuel bundles are then installed in the fuel channel in the desired sequence followed by the shield plug, the spacer, and the channel closure. The refuelling sequence is shown in Figure 3.8-2.

After unclamping from the fuel channel, the fuelling machine rotates 90°, and is moved to the fuel transfer port by the fuelling machine carriage, where the irradiated fuel is discharged to the irradiated fuel storage bay. New fuel bundles are picked up at the new fuel port and the fuelling machine is returned to the reactor face to refuel the next channel.

3.8.5 Irradiated Fuel Storage Bay

The irradiated fuel storage bay located in the Reactor Auxiliary Building provides for the interim storage of irradiated fuel. Equipment is provided for moving fuel bundles from the fuel transfer system and storing them under water in modules. The water cover ensures adequate attenuation of radiation fields.

Criticality of stored irradiated CANDU fuel bundles is impossible, regardless of storage configuration because of the natural uranium fuel used in CANDU reactors.

Provision is made in the irradiated fuel bay for underwater handling of various irradiated assemblies and hardware as well as for the discharge of irradiated fuel into dry storage canisters, an auxiliary wet storage bay, or into shipping flasks.

The Irradiated Fuel Bay Cooling and Purification System removes decay heat from the irradiated fuel removed from the reactor both in the storage bay and in the fuel handling bay. The system ensures water clarity and maintains the activity of the water at a low level. Provision is made to maintain adequate water level in the bays to ensure shielding during all phases of fuel handling and storage.

3.8.6 Maintenance and Servicing

Most fuel handling equipment, including the control equipment is accessible for maintenance with the reactor at full power. Routine maintenance of the fuelling machine is done within the Reactor Building, in the accessible fuelling machine service room. For major maintenance, the fuelling machine head may be removed from the carriage and transferred into the fuelling machine maintenance room in the Maintenance Building. Full decontamination facilities are provided in the Maintenance Building.

3.9 Steam and Feedwater Systems

Steam is produced in the two steam generators and fed by two separate steam mains, which pass through containment penetrations to the turbine building where they connect directly to the turbine steam chest.

The steam pressure is normally controlled at a constant value by varying reactor power to suit the turbine-generator demand. The turbine bypass system is sized to permit 100% of full power steam flow to the condenser for a short period, and a continuous steam flow to the condenser of up to

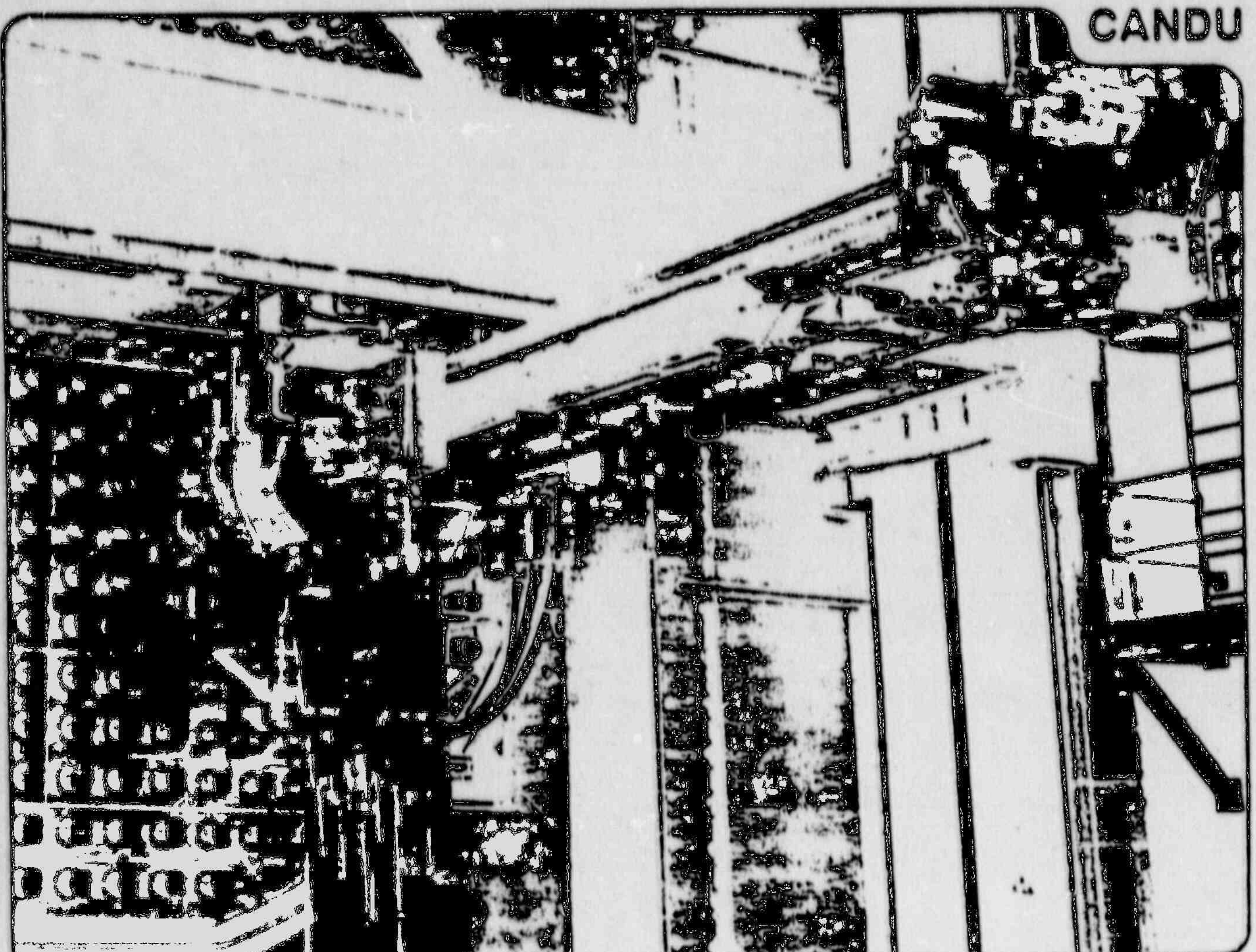


Figure 3.8-1 CANDU Fuelling Machine

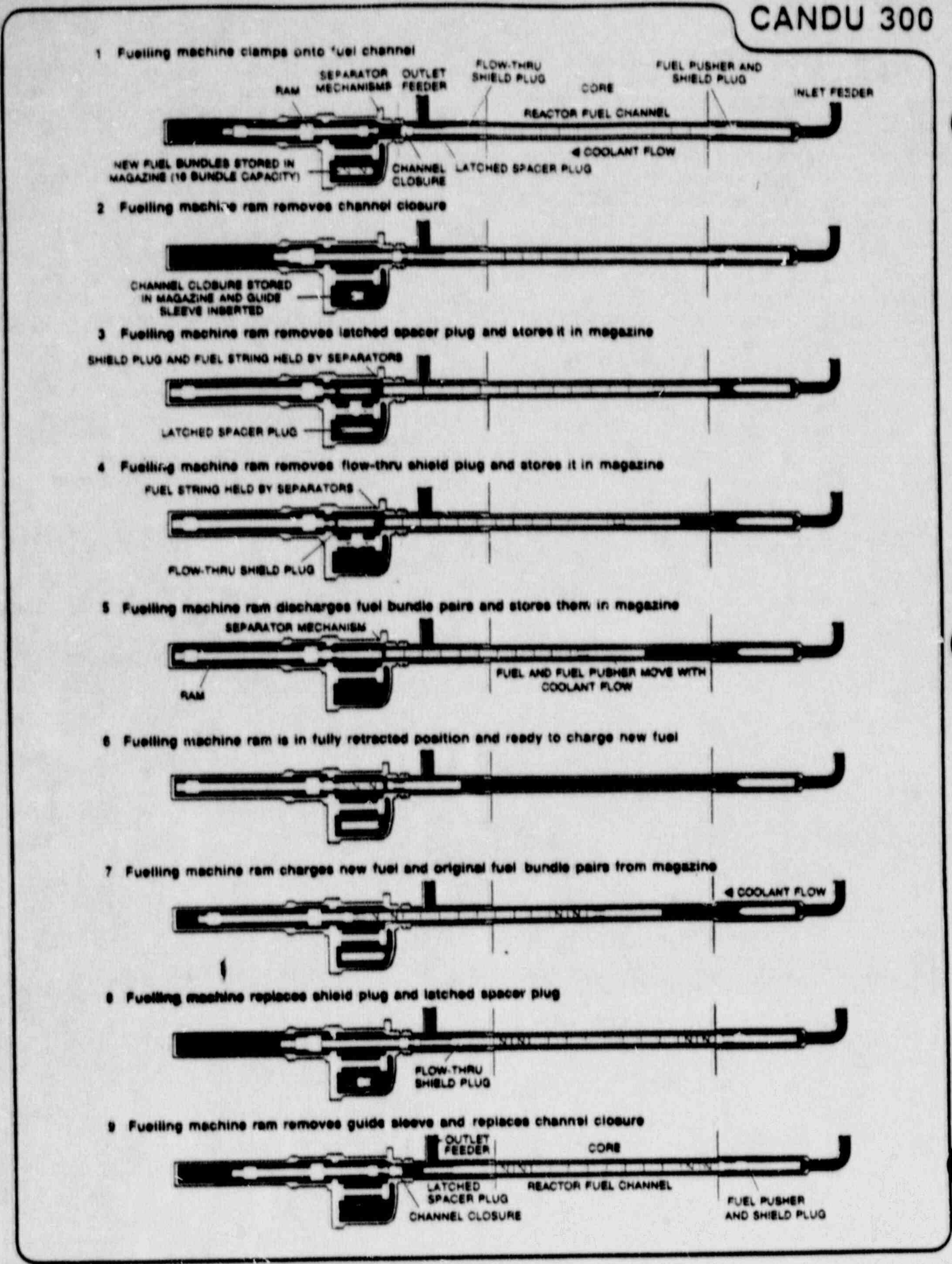


Figure 3.8-2 Fuel movement sequence for single ended refuelling

75% of full power steam flow. The condenser steam discharge valves have sufficient capacity to avoid lifting the main steam safety valves following a loss of line or a turbine trip and, hence, permit continuation of reactor operation.

Main steam safety valves (MSSVs) are provided in each steam main to protect the Steam System and steam generators from overpressure.

The Feedwater System comprises the main feedwater pumps on Class IV power, an auxiliary feedwater pump on Class III power, and a steam turbine driven auxiliary feedwater pump. The feedwater is demineralized and preheated light water. Feedwater from the regenerative feed heating system is supplied to the steam generators through two separate feedwater mains.

A Blowdown System is provided to limit impurities in the steam generators. To accomplish this, provision is made for a continuous blowdown from the steam generators secondary side to the cooling water outfall.

Facilities to sample and measure flow in each blowdown line are provided.

To protect the integrity of the steam generator tubes, strict control of the secondary side water chemistry is required (Refer to Section 3.7.4).

3.10 Electrical Power System

3.10.1 General

This section covers the Electrical Power System for both the Nuclear Steam Plant (NSP) and the Balance of Plant (BOP). The complete description is provided here for clarity.

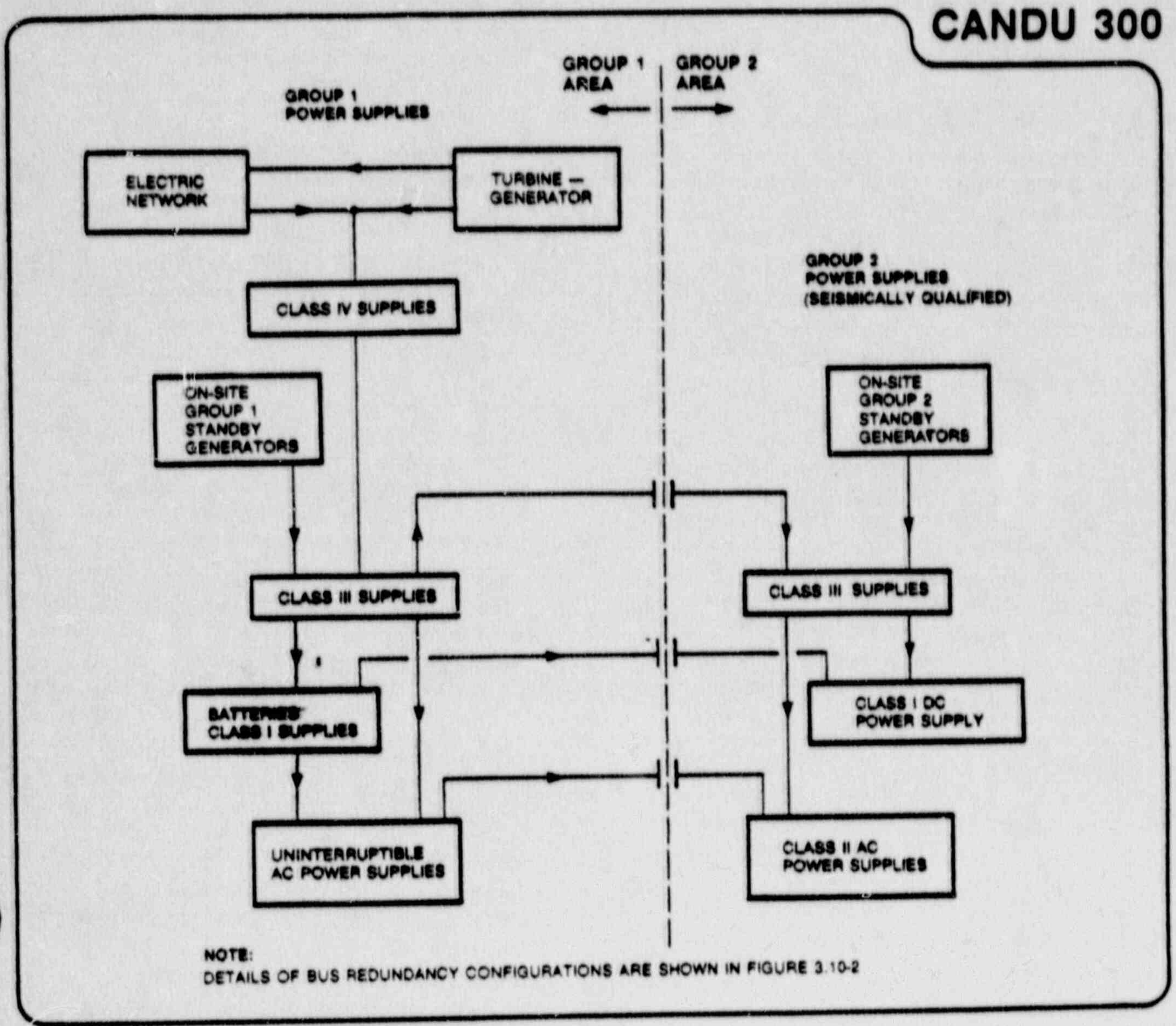


Figure 3.10-1 Two group electrical power supply concept

The station Electrical Power System consists of the Main Power Output System and the Station Service Power Distribution System. The Main Power Output System transfers power produced by the turbine-generator to the off-site electrical network. The system includes the turbine-generator with static excitation system, generator neutral grounding, generator metering and protection system, a generator breaker, isolated phase busduct, a unit service transformer (UST), a system service transformer (SST) and the main output transformer (MOT).

The main output transformer steps up the generator output voltage to the same level as the switchyard transmission voltage.

The Station Service Distribution System supplies process and instrumentation loads within the plant. The two group electrical supply concept is illustrated in Figure 3.10-1, and a simplified single line diagram is illustrated in Figure 3.10-2. Group 1 supplies the systems of the plant dedicated to normal power production, while Group 2 supplies safety related loads.

3.10.2 Electric Power System Station Services

The station service power supplies are classified in order of their level of reliability requirements. Four classes that range from uninterruptible power to that which can be interrupted with limited and acceptable consequences are provided as follows:

- Class IV Power Supply: alternating current (AC) power to auxiliaries and equipment that can tolerate long duration interruptions without endangering personnel or station equipment. Class IV power is available to Group 1 systems only.
- Class III Power Supply: alternating current (AC) supplies to auxiliaries that are necessary for the safe shut down of the reactor and turbine and can tolerate short interruptions (in the order of one to three minutes) in their power supplies. Class III power is available to both Group 1 and Group 2 systems and is normally supplied from the Class IV electric power sources. On-site standby generators located in both the Group 1 and Group 2 Service Buildings provide an alternative power source to the Class III system. Connections between the Group 1 and Group 2 Class III supplies are via fully qualified isolation devices. The Group 2 Class III system, including its standby generators is seismically qualified to be operational after an earthquake.
- Class II Power Supply: uninterruptible, alternating current (AC) supplies for essential auxiliaries, channelized to match the redundancy requirements of station instrumentation and control systems. Class II power is available to Group 1 and Group 2 systems. The Group 2 Class II distribution system is seismically qualified to be operational after an earthquake.
- Class I Power Supply: uninterruptible, direct current (DC) supplies essential auxiliaries, triplicated and channelized to match the redundancy requirements of control logic and reactor safety circuits. Class I power is available to Group 1 and Group 2 systems. The Group 2 Class I distribution system is seismically qualified to be operational after an earthquake.

3.10.3 Normal Power Sources

Power for the Station Service Distribution System during normal or shutdown conditions is supplied from two sources; the off-site network and/or the main generator.

On start-up, the Station Service Distribution System is fed by the system service transformer (SST) which is supplied from the switchyard. During normal operation, the Station Service Distribution System is equally shared between the system service transformer and the unit service transformer. However, both the transformers are sized for the total station service load and thus either one can supply the total service load in the event of failure of one supply.

Transfer schemes are provided to obtain power from either the main generator through the unit service transformer or from the grid through the system service transformer. This requirement arises from the necessity to minimize interruptions in supply to the reactor's Heat Transport System. Three schemes are provided:

- A parallel transfer scheme, which is manually initiated but automatically supervised and executed, to allow for a transfer between the two sources at the end of the station start-up phase or before shutdown and as needed during normal operation of the plant, e.g. mechanical trips on the turbine, reactor trips and generator stator cooling trips (both via turbine trip);
- A fast open transfer, which is automatically initiated, supervised and executed, on electrical faults on the main generator or any of the main transformers; this is a "dead bus" transfer allowed to proceed only when both of the incoming breakers are in the open position during the transfer;
- A residual voltage transfer, which is used as a backup to the fast open transfer, activated on failure of the fast open transfer to be completed within the prescribed time or when the angular displacement of the two voltage vectors exceeds the allowable limits, the latter being dependent on the capabilities of the distribution equipment and loads.

The unit service transformer has three (3) windings (one primary and two secondary) so that a suitable margin in the standardized switchgear short circuit

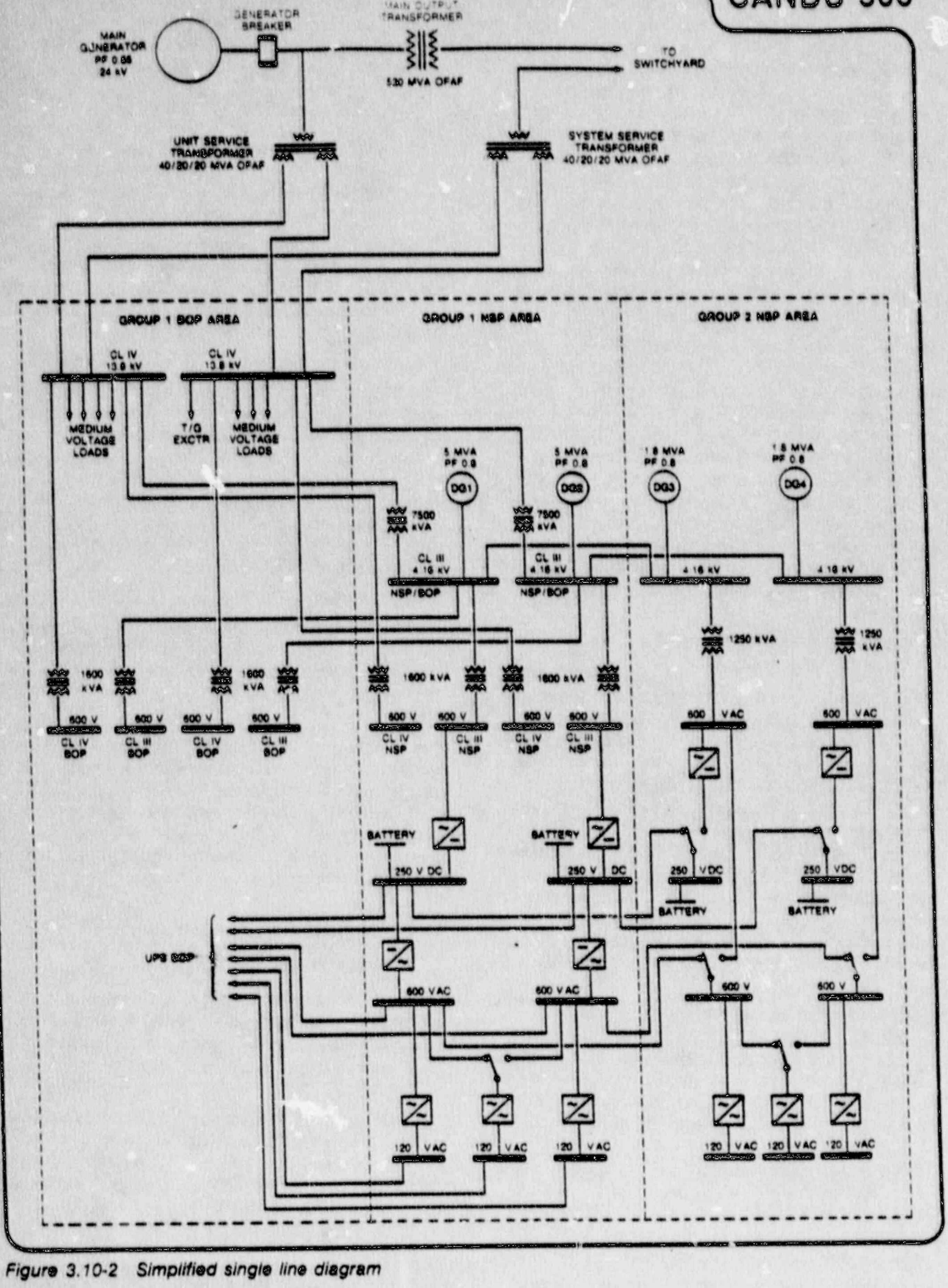


Figure 3.10-2 Simplified single line diagram

level can be provided. The low voltage side of the unit service transformer is resistance grounded. The system service transformer is a 3 phase 3 winding unit and the HV winding is star connected, solidly grounded. The LV windings are star connected and grounded through a resistor. An additional unloaded delta connected winding, approximately 15% of transformer capacity is provided for third harmonic currents. The transformer is provided with an on-load tap changer, for automatic operation and maintenance of the voltage on the LV side. The medium voltage switchgear (13.8 kV and 4.16 kV typical) is of the indoor metal clad type with drawout magnetic air circuit breakers with a nominal short circuit rating of 750 MVA and 250 MVA respectively.

The low voltage buses are supplied through step-down transformers. The size and impedance of the transformers are selected to permit starting of the low voltage motors without excessive voltage drop and to limit the fault level to less than 50 MVA. This allows the use of commercially available switchgear. The neutral path of the low voltage system is solidly grounded. The low voltage switchgear is metal enclosed with drawout air circuit breakers. The low voltage loads are limited to 300A. Loads up to 100A are supplied from motor control centres and between 100A and 300A from the low voltage switchgear.

3.10.4 Standby Power Sources

- **Standby Generator Sets**

On-site standby generators located in both Group 1 and 2 Service Buildings provide an alternative power source to specific Group 1 and Group 2 station loads connected to the Class III system. The Class III shutdown loads are duplicated, one complete system being fed from each diesel generator. There are two diesel generator sets in the Group 1 Service Building, each sized to supply the total safe shutdown load of the unit following loss of Class IV power supply. There are two additional diesel generator sets in the Group 2 Service Building, each sized to supply the total safe shutdown load of the unit following a combination of worst events including an earthquake.

In the event of failure of the normal power sources, all standby generators start automatically. If the Group 1 diesels have successfully run, both Group 1 and 2 Class III electrical systems will be fed by the Group 1 diesels. In case of unsuccessful Group 1 diesel generators' operation or an earthquake, the Group 2 diesel generators will power the Group 2 Class III buses. The standby sets are designed to accept key loads within 35 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days.

- **Station Batteries**

The reactor protective and safety systems, control logic, instrumentation, computers, critical motor loads, essential and emergency lighting and switchgear operation are supplied with uninterruptible power. The uninterruptible power supplies provide AC and DC power as required to the Class II and Class I systems.

The Group 1 batteries are sized to support all the loads connected for up to 60 minutes following an interruption to the normal (Class III) source. AC power is obtained through static inverters.

The Group 2 batteries, located in the Group 2 Service Building, are seismically qualified to be operational after an earthquake. They are utilized for control logic of standby generators and switchgear operation.

3.10.5 Cabling System

The selection of insulation type is dictated by voltage level, maximum operating temperature, expected radiation exposure, environmental conditions, chemical stability and flame retardancy. Shielding is utilized where necessary. The cable routing and separation rules for safety-related systems follow well established CANDU safety and reliability practices. The cable containment penetrations meet IAEA testability requirements.

3.10.6 Grounding

There are two grounding networks. The main network consists of a bare copper cable net in the ground and grounding rods. This serves as the neutral and protective ground for the power distribution systems and switchyard.

The second grounding network is used for the station instrumentation. It consists of a network of insulated copper conductors. To avoid ground loop currents which could affect instrumentation, this network is connected to ground only at one point, as close as possible to the geographical centre of the main grounding system.

3.11 Instrumentation and Control

3.11.1 Distributed Control System

Most Group 1 control functions are implemented by a Distributed Control System (DCS) which uses data-highways for signal transmission, and programmable micro-processors to implement the control logic.

The DCS consists of a number of channelized local stations distributed throughout the plant outside the reactor building, linked by channelized dual-redundant co-axial cable data highways. Sensors and actuators are connected to input and output modules in the local stations by relatively short cables.

Figure 3.11 1 shows the general configuration

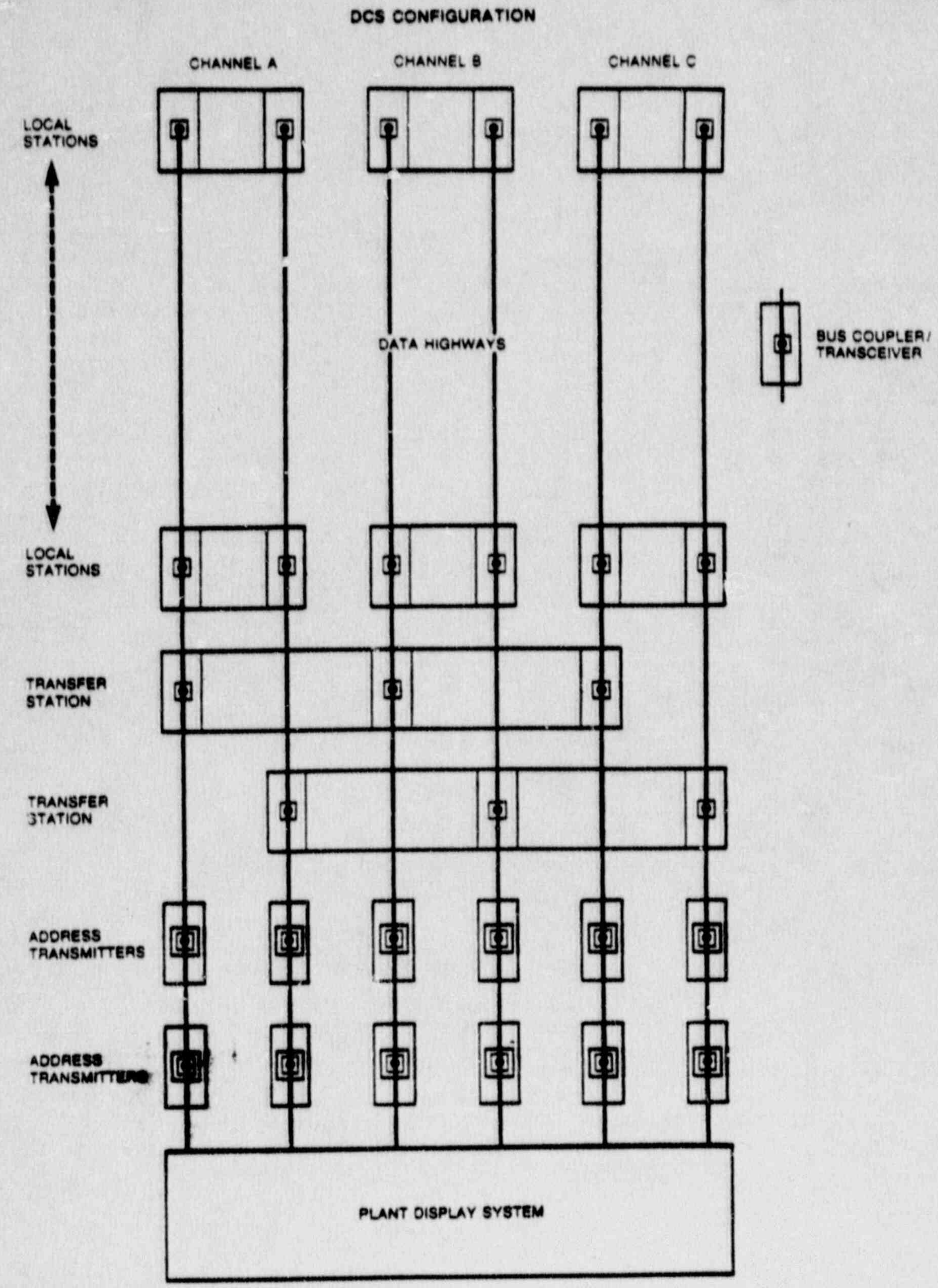


Figure 3.11-1 General configuration of DCS

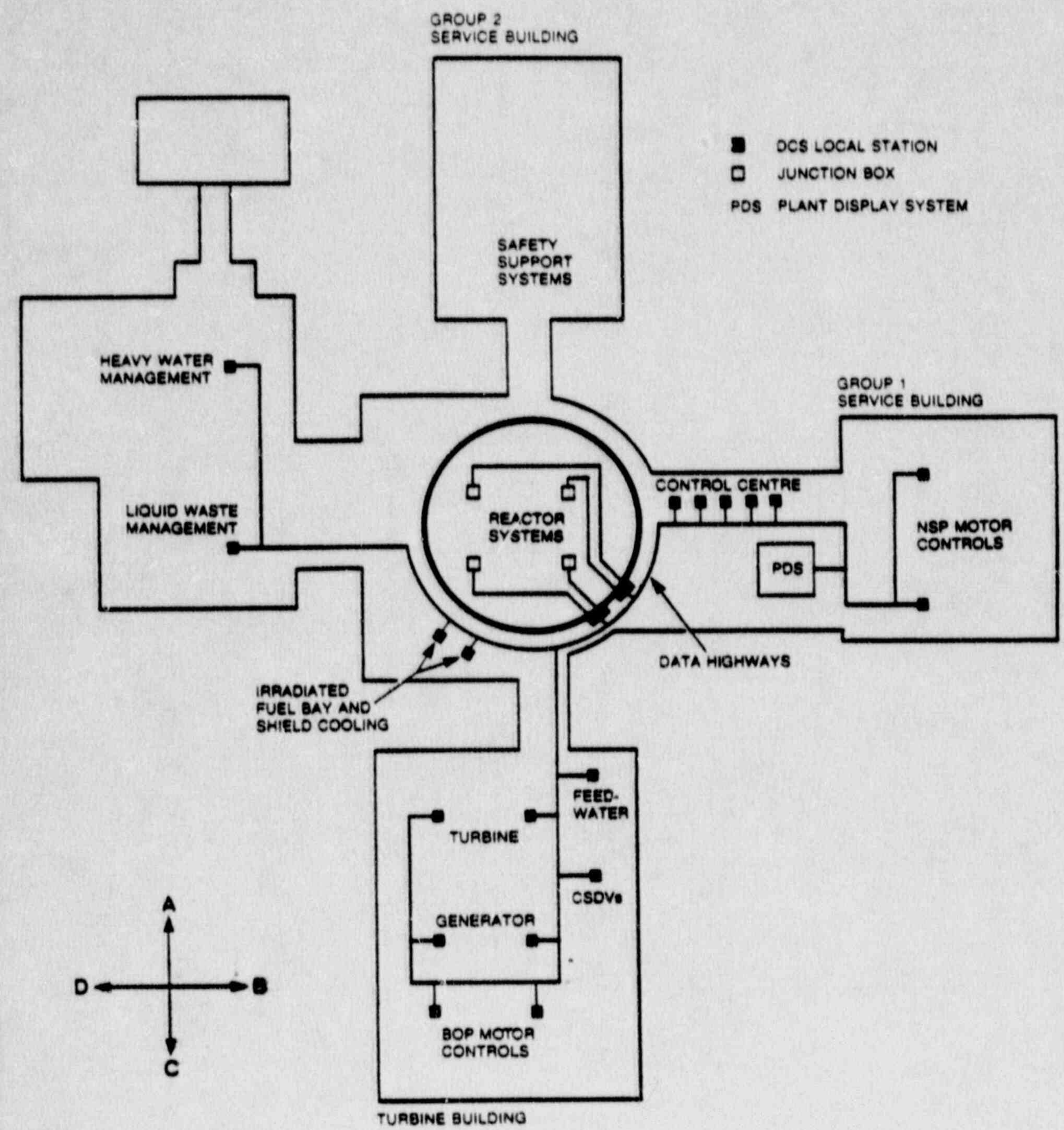


Figure 3.11-2 Representative locations of DCS local stations — (only one channel is shown)

of the DCS. Figure 3.11-2 shows representative locations of the DCS local stations.

The DCS replaces the control trunk cabling, control distribution frame, relay logic, analog controllers, and comparators used in previous CANDU control systems, as well as the central computers used for signal multiplexing and digital control. The major control functions (reactor regulation, heat transport pressure and inventory control, steam generator pressure and level control, etc.) implemented in the central station computers of previous CANDU power stations, as well as most of the control loops previously not computerized, are distributed among a number of small, powerful DCS micro-processors. However, the system configuration uses the same dual-redundant fault-tolerant concepts used in previous CANDU computerized control systems to assure a high level of system reliability.

The DCS provides a high rate of data transmission, high system reliability, data security and comprehensive system fault detection. Automatic failure detection and recovery features protect against cable or device failure. If a module fails, the automatic failure detection features transfer control to the backup module. The system permits replacement of a failed module on-line without affecting the functions performed by the system.

A separate control system is used for control and monitoring of the fuelling machine and related systems (Refer to Section 3.8.3).

3.11.2 Plant Display System

The Plant Display System (PDS) provides alarm message display, graphic data display, operator-machine interface and data logging. Most process

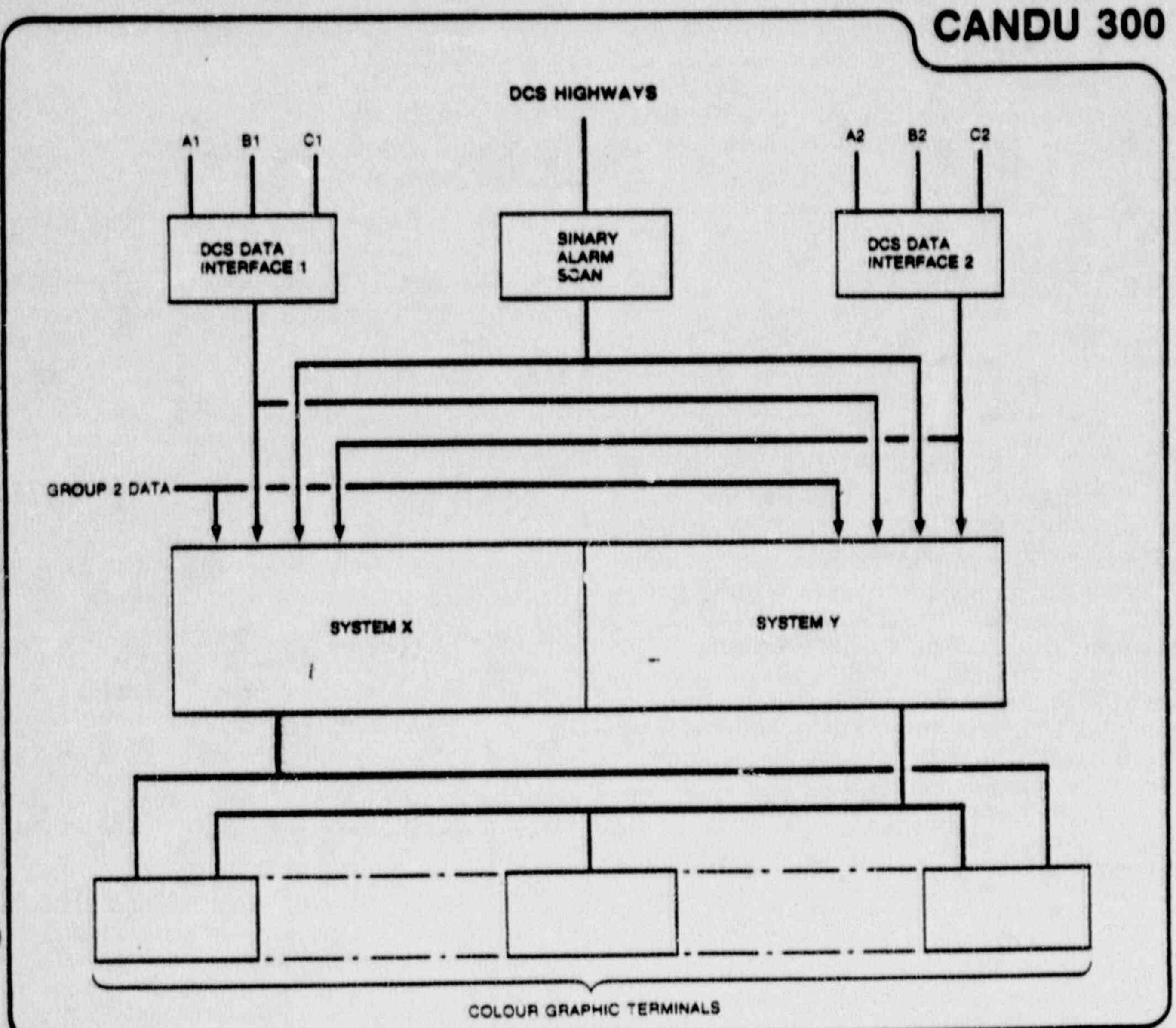


Figure 3.11-3 PDS configuration

information from the plant is fed to the PDS via the Distributed Control System (DCS). Operator input to the DCS, such as setpoint adjustment for control functions, is communicated via the PDS.

In addition, the PDS also receives information from systems not connected to the DCS, such as safety systems, meteorological monitoring, station clock, etc., to provide a complete signal data base accessible to the operator.

The PDS, shown in Figure 3.11-3, is dual redundant so as to maintain function availability in excess of 99 per cent. With its modular architecture, it is possible to expand the system to meet future requirements.

The major subsystems of the PDS are the:

- Graphic Operator Stations which form the interface between the operator and the plant utilizing colour graphic CRT displays.
- Annunciation System which reports alarms on the dedicated CRTs and records the alarms for retrieval and analysis by the operator. Long-term archiving of the alarm information is performed by the File Server.
- File Server which maintains the plant database for use by the other systems within the PDS and provides archiving and retrieval for long-term historical information such as trends and alarms.
- Log Generator, which generates the necessary records of station operating data.

3.11.3 Control Centre

- **General**

The Control Centre is located in the Reactor Auxiliary Building between the Group 1 Service Building and the Reactor Building, as shown in Figure 3.11-4.

The Main Control Room (MCR), shift supervisor's office, and the Work Control Office are located in close proximity to one another. The computer equipment driving the MCR CRT's is located next to the MCR. Control equipment is located adjacent to the MCR for items requiring fast operator access, with the remainder of the control equipment across the hall.

- **Main Control Room**

The Main Control Room contains the following:

- One main control console, containing a number of CRT displays and functional keyboards.
- A console for safety systems monitoring and testing.
- Vertical panels containing alarm annunciation (windows plus dual CRTs), indicators and controls required for post-accident monitoring, along with a small number of infrequently used manual controls and overrides.
- A console for operation of the fuel handling systems.
- Other facilities, such as hard copy output devices, drawing laydown area, etc..

The control room design provides the resources and environment to support the operator in the safe and efficient control of the plant. Plant control at low power levels uses controls at the vertical panels. At higher power levels, plant control is based on automated control supervised at the main control console.

The main control console is arranged to allow the operator an unobstructed view of the alarm annunciation (CRTs and windows) on the vertical panels. The control console has a number of functional keyboards to input control and display commands. A standard CRT interface methodology provides a uniform operator interface.

- **Alarm Annunciation**

Main Control Room annunciation includes the following functions:

- A constantly updated CRT indication of the last forty alarm and clearance messages.
- Lighted window messages for a number of important alarm conditions, including safety system annunciation.
- An audible signal to attract the operator.
- A permanent record of the times of all alarms and clearances occurred.

Alarm annunciation analysis facilities at the control consoles provide the operator with an enhanced capability to process alarm information during plant upset conditions. These capabilities include alarm summaries and extended use of alarm conditioning.

Approximately 300 alarm windows are provided. They are located in the upper portion of the vertical panels as a supplement to the CRT displayed messages, but are completely independent from them and the PDS. The windows are colour coded to indicate the classification of the alarm.

- **Post Accident Monitoring**

Control and monitoring of the plant in the post accident mode of operation shall be based in the MCR. If the MCR is rendered uninhabitable at any time, them the capability for shutting down the reactor and performing the same long term control and monitoring in the post accident mode (as that available in the MCR) shall be available in the Secondary Control Area (SCA).

The SCA is located in the Group 2 area of the plant, so that its location, instrumentation and support facilities are independent of the MCR. The MCR is in the Group 1 area of the plant.

3.11.4 Overall Plant Control

The Overall Plant Control System uses digital processors to perform all major control functions.

The plant operates in one of two modes: in NORMAL mode the turbine is controlled to a setpoint specified by the operator and the reactor follows to maintain constant steam generator secondary side pressure; in ALTERNATE mode the reactor is controlled to a setpoint that is either set by the operator or by the state of the plant, and the turbine follows to keep steam generator secondary side pressure constant. In NORMAL mode the plant is inherently responsive to grid frequency changes — a frequency drop opens the turbine governor valves causing a drop in steam generator secondary side pressure which in turn increases reactor power.

Figure 3.11-5 shows the main elements of the Overall Plant Control System.

The Nuclear Steam Plant (NSP) "loads" used to control the steam generator secondary side pressure are:

- The turbine-generator — normally controlled by the turbine load control program. In ALTERNATE mode it is controlled by the steam pressure control program. If required, in NORMAL mode it can also be controlled manually by the operator.

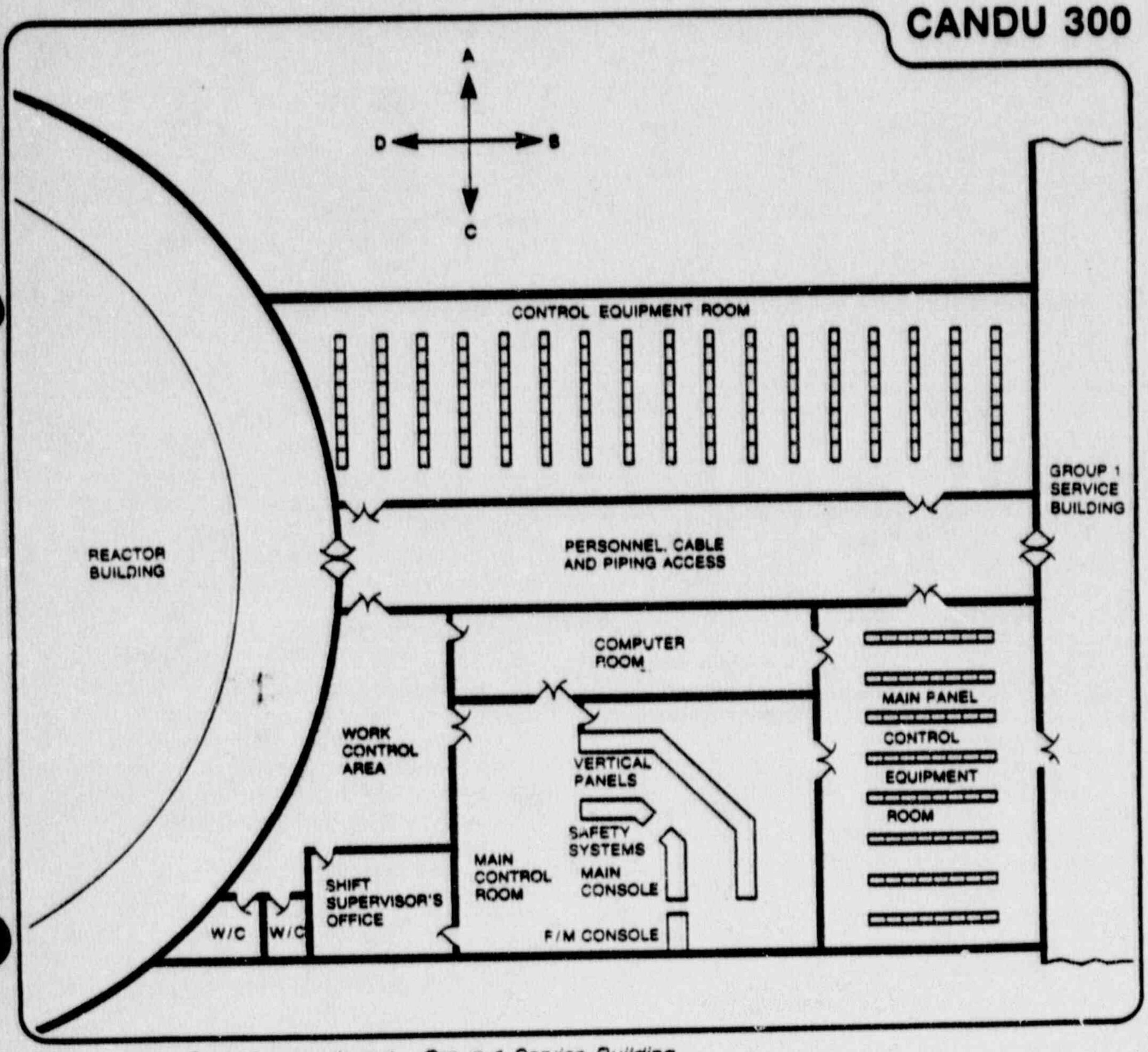


Figure 3.11-4 Control centre layout - Group 1 Service Building

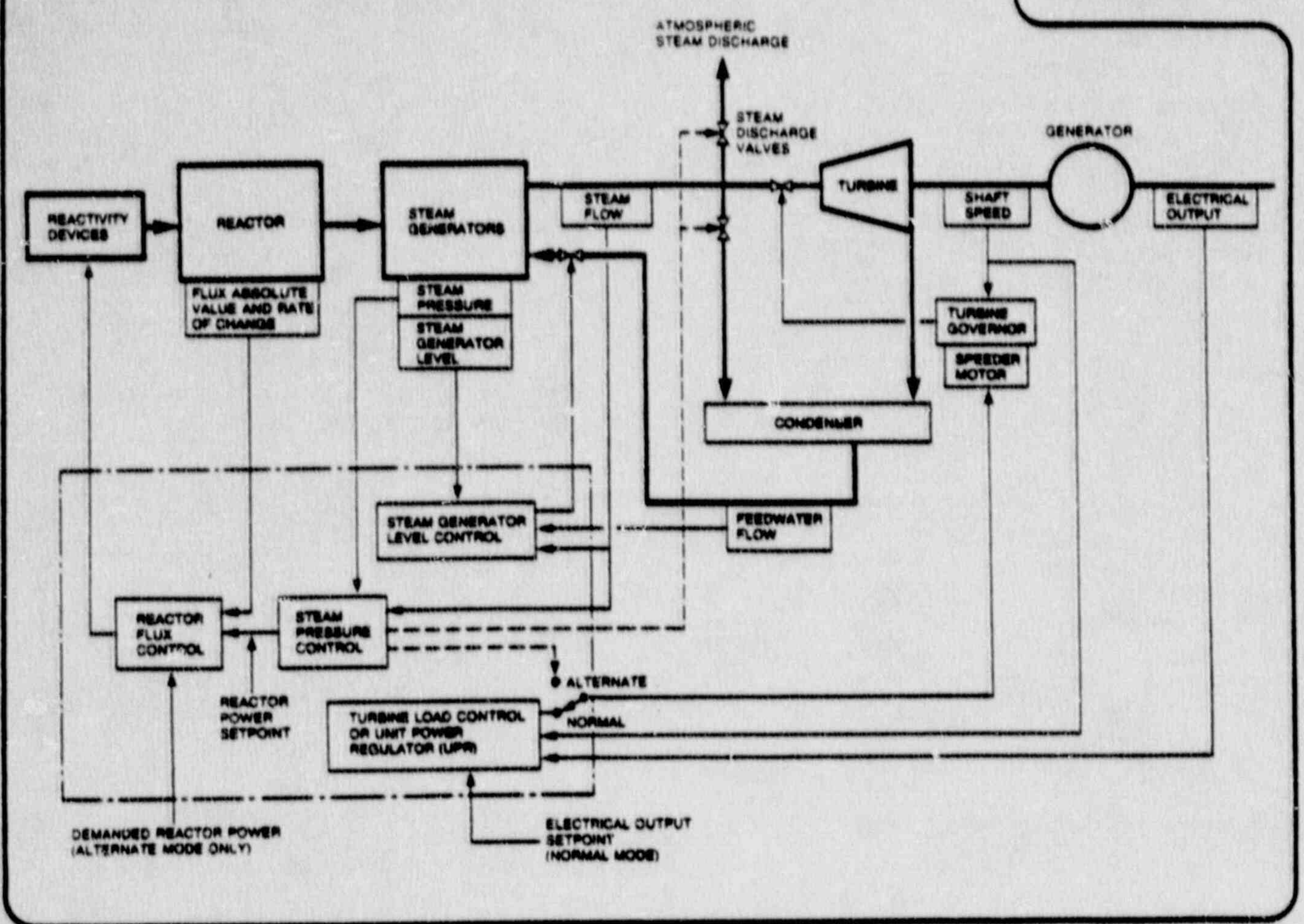


Figure 3.11-5 Main components of overall plant control

- The condenser steam discharge valves (CSDVs) normally actuated by the steam pressure control program but can also be positioned under manual control. The CSDVs are normally closed and open if steam generator secondary side pressure exceeds its setpoint by more than a specified offset. The CSDVs can continuously pass 75 per cent of full power steam flow to the condenser which is sufficient to allow reactor power to be held above the poison-prevent level if the turbine is unavailable.
- The atmospheric steam discharge valves (ASDVs) are of limited capacity (10 per cent of reactor full power steam) and normally controlled by the steam pressure control program, but can also be positioned manually. The ASDVs are normally closed but open if the steam pressure exceeds a setpoint by more than a preset offset. The ASDVs are used for trimming steam pressure transients or as a heat sink during warmup or when the condenser is unavailable.

These NSP loads are controlled by the steam pressure control program which controls steam generator secondary side pressure to a constant setpoint. This is accomplished by changing the reactor power setpoint or by adjusting the plant loads. The steam pressure control program also controls the Heat Transport System warmup and cooldown.

Other control programs are:

- The steam generator level control maintains steam generator level at the setpoint during normal operating conditions and within limits under upset conditions such as reactor fast run-up, reactor setback, turbine fast run-up, turbine fast run-back and step change of turbine power and turbine trip.
- The heat transport pressure control holds the Heat Transport System at a fixed pressure. This control system controls pressure over a narrow range near the normal operating pressure of the system. The reactor outlet header is the sensing point for pressure control.
- The heat transport inventory control maintains the combined mass of heavy water in the Heat

Transport System and the pressurizer at specific values for all Heat Transport System temperature and reactor powers. The inventory control algorithm in the processors calculate the mass of heavy water in the Heat Transport System based on various process measurements and establishes the desired level in the pressurizer to achieve the required combined inventory. This level is then used as the setpoint for the control of pressurizer level.

- Reactor control - bulk reactor power is raised or lowered by changing the position of the zone control units in unison. Spatial control of the reactor power is achieved by differentially changing the position of the zone control units illustrated in Figure 3.3-3.

  The reactor power is determined by flux detector measurements. The average flux measurement is automatically calibrated by comparison to the overall reactor thermal power.

3.12 Safety Systems

3.12.1 Nuclear Safety Principles

The nuclear safety principles applied to the design of the CANDU 300 reactor ensure that Canadian regulatory requirements are met. Rather than imposing a number of detailed regulatory requirements, the Canadian regulatory body (the Atomic Energy Control Board (AECB)) sets general criteria and requires the designer to develop detailed design requirements to comply with them.

The public dose limits are given in a probabilistic way. The AECB defines reference dose levels for two classes of accidents: a single failure in any of the process systems (those systems required for normal operation), and a single failure when one of the special safety systems is unavailable. These two types of accidents have quite different expected frequencies and the permissible dose limits reflect this. The AECB also requires that the special safety systems (Shutdown System No. 1, Shutdown System No. 2, Emergency Core Cooling, Containment) be separate from each other and from the process systems, and that they be testable to show an unavailability of less than 10^{-3}.

To guard against cross-linked and common mode events, all systems in the CANDU 300 are assigned to one of two groups (Group 1 and Group 2). The systems of each Group are capable of shutting down the reactor, maintaining cooling of the fuel and providing plant monitoring capability, in the event that the other group of systems is unavailable. The Group 2 systems have the additional role of mitigating the effects of any postulated accident. Group 1 systems are those primarily dedicated to normal plant power production. These systems are, except for certain nuclear class systems within the Reactor Building, not seismically nor environmentally qualified. Group 2 systems have safety or safety support functions, and are seismically and environmentally qualified. The Group 1 and Group 2 systems are located, to the greatest extent possible, in separate areas and use diverse principles of operation.

The design concepts which ensure that special safety systems and safety support systems perform their safety functions with a high degree of reliability, include the use of redundancy, diversity, separation and the application of quality assurance standards, and the use of stringent technical specifications including environmental qualification for accident conditions. Redundancy is the use of two or more components or systems which are each capable of performing the necessary function. Redundancy provides protection against independent equipment failures.

Diversity is the use of two physically or functionally different means of performing the same safety function. Diversity provides protection against certain types of common mode failures, such as those arising from design or maintenance errors.

Where practical, the special safety systems use diversity in performing the same safety function. For example, the two shutdown systems that shut the reactor down quickly after an accident use different methods of operation and are of a physically different design.

Separation refers to the use of barriers of distance to separate components or systems performing similar safety functions, so that a failure or localized event occurring in or near one system or component is unlikely to affect the other.

Separation provides protection against common mode or cross-linked effects, such as fires and missiles.

A series of Canadian technical standards has been developed to control various key aspects of the design and construction of CANDU nuclear power plants. These standards, known as National Standards of Canada, have been prepared through the Canadian Standards Association. These standards cover the design of pressure retaining components (using ASME Boiler and Pressure Vessel Code, Section III as a basis), the quality assurance programs applicable to the various project stages, the design and construction requirements for containment structures, periodic inspection requirements, seismic qualification methods, and shutdown system requirements. The use of these standards ensures a high system quality, which is essential to the real safety of the plant.

The completed plant design is analyzed to demonstrate that systems are capable of performing their assigned safety functions and that the radiation dose criteria can be satisfied. In addition, the frequency of accidents and the likelihood that the consequences of the accident can be mitigated by available systems is confirmed using fault tree reliability techniques. This form of analysis identifies the failures of components which can cause an accident condition, analyzes the effects on connected systems, and shows whether the operator is likely to be successful in taking corrective actions in the time available.

3.12.2 Shutdown Systems

The CANDU 300 reactor incorporates two diverse shutdown systems which are independent of each other and from the Reactor Regulating System. Both systems can render the reactor subcritical during normal reactor operation or under accident conditions.

Shutdown System No. 1 (SDS1) consists of solid shutoff absorber rods which drop into the core when a trip signal de-energizes clutches which hold them out of the core.

Shutdown System No. 2 (SDS2) uses injection of liquid poison into the low pressure moderator to quickly render the core subcritical. A concentrated solution of gadolinium nitrate is injected into the moderator through nozzles. The injection is initiated by opening fast acting valves to pressurize with helium the individual poison tanks associated with each nozzle.

Each system uses appropriate sensing parameters for each identified process system failure for which it is designed to detect. The instrumentation to measure each trip parameter is triplicated, and trips the reactor on a two-out-of-three logic basis. Each of the three logic channels for a specific parameter is separated from the other two.

Neutron and process measurements are used to trip both shutdown systems. Neutronic measurements are obtained from self-powered in-core flux detectors and ion chambers. Process measurements include heat transport pressure, heat transport flow, reactor building pressure, steam generator low level, pressurizer low level and low steam generator feedline pressure and high moderator level. Manual trips are provided in the Main Control Room and in the Secondary Control Area.

All information regarding trip parameters and the status and operation of the system is displayed on dedicated panels in the Main Control Room. Sufficient information for post-accident monitoring is also provided in the Secondary Control Area.

A computerized monitoring and test system provides the operator with indications of all shutdown system parameters and assists the operator in testing. The system prompts the operator, executes the testing, and records the test results.

For each shutdown system, trip parameter instrumentation and logic is tested in such a way that the complete system, from process variable sensing to final trip action is tested; e.g. shutoff rod tested by partial drop into core. The test frequency depends on the unavailability requirement and the equipment failure rate for each trip variable. A test is automatically terminated if another trip channel goes into a tripped state.

3.12.3 Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) removes residual and decay heat from the fuel following a failure of the Heat Transport System pressure boundary. Depending on break size and location, the Emergency Core Cooling System may operate in conjunction with the steam generator and feedwater systems in transferring the core heat to an ultimate heat sink.

The ECCS operation is divided into two parts: short term injection and long term recirculation. Short term injection consists of two stages: high pressure and low pressure injection. During the high pressure injection stage, water from the accumulator tanks is injected into the Heat Transport System by pressurized gas. After this water is depleted, low pressure injection takes over injecting water from a grade level tank via the ECC pumps. The water level in the grade level tank is continuously monitored via triplicated level measurements. A connection is provided to this tank for demineralized water makeup, and for initial filling.

When this water supply is depleted, the long term operation begins by recirculating a mixture of H_2O and D_2O from the reactor building floor back into the Heat Transport System via the heat exchangers and pumps.

The parameters used to detect a LOCA and initiate system operation consist of a low Heat Transport System pressure conditioned by either high reactor building pressure, low pressurizer level, or a high moderator level. The conditioning parameters are used to ensure that a LOCA has actually occurred, minimizing the possibility of spurious operation.

Information on the status of the system is displayed on a dedicated panel in the Main Control Room. Parameters and controls needed for long-term operation of the system after an accident are also provided in the Secondary Control Area.

A computerized monitoring and test system monitors appropriate system parameters, operating visible and audible alarms if an abnormal indication is detected. This system also guides the operator through the system test sequence and records the results of the test.

3.12.4 Containment System

The Containment System for the CANDU 300 consists of the containment envelope and the Containment Isolation System.

The containment envelope is a pressure retaining boundary consisting of the reactor building and metal extensions such as airlocks, essential piping systems (for core heat removal), and electrical penetration assemblies. The envelope is designed to withstand the peak pressure which would occur following the largest postulated loss of coolant accident. Piping systems passing through the envelope are equipped with isolation valves.

The Containment System automatically closes all penetrations open to the containment atmosphere when an increase in containment pressure or radioactivity level is detected. Redundant valves or ventilation dampers are provided in these through-containment piping or ducting open to the containment atmosphere. Measurements of containment pressure and radioactivity are triplicated and the system is actuated using a two-out-of-three logic system.

A containment atmosphere heat sink is provided by reactor building air coolers. These coolers are used to control temperatures within containment during normal plant operation and are environmentally qualified to continue operating after a loss-of-coolant accident.

3.12.5 Safety Support Systems

Safety support systems supply reliable services, such as cooling water, electrical power, and instrument air to support the operation of the Special Safety Systems described above. These systems are located in the Group 2 area, separate from the Group 1 systems. These systems are seismically qualified to remain functional following a Design Basis Earthquake. They can be monitored and controlled from the Main Control Room, via suitable buffering devices.

The Group 2 Feedwater System provides a supply of water to the steam generators independently of the normal Feedwater System.

The Electrical System supplies all electrical power needed to perform safety functions for any accident condition. It consists of the Group 2 Electrical Distribution System which is connected to the Group 1 Electrical Power System through suitable isolating devices and the Group 2 standby generators and batteries. The Group 2 Electrical System supplies power to all Special Safety Systems and Safety Support Systems, and to the Secondary Control Area.

The Group 2 Instrument Air System supplies compressed air to instruments and valve operators, as required, from air tanks located in the Group 2 Service Building. These tanks are replenished from the station Compressed Air System.

The Group 2 cooling water systems transfer heat from components performing a safety function to the ultimate heat sink (reservoir, lake, river, or sea).

A Secondary Control Area is provided in the Group 2 Service Building which contains sufficient monitoring and control capability to maintain the plant in a safe condition following an accident condition. The Secondary Control Area, which is seismically and environmentally qualified, provides a back-up in the event that the Main Control Room, located in the Reactor Auxiliary Building, becomes uninhabitable or inoperative. Services are supplied to maintain adequate environmental conditions for operating staff and for equipment.

3.13 Reactor Building and Reactor Auxiliary Building Ventilation

3.13.1 Reactor Building

Ventilation and cooling facilities, including local fan coil cooling units, are installed in the Reactor Building. All air entering the Reactor Building is dried and filtered to remove moisture and impurities. The Reactor Building is divided into atmospheric areas so that ventilation air flows to areas of increasing activity. All air leaving the Reactor Building is monitored for activity, and is filtered if necessary.

Local fan-coil units remove heat dissipated by operating equipment and are used to cool the individual Reactor Building areas. The reactor vault is cooled by externally located fans and heat exchangers.

D_2O Vapour Recovery System in the Reactor Building recovers heavy water from the Reactor Building atmosphere by passing Reactor Building air through molecular sieve desiccant dryers to remove any water vapour present. The dryers are reactivated by passing hot air through the dryers where the absorbed moisture is evaporated and carried out in the air stream to a cooling coil condenser for recovery.

3.13.2 Reactor Auxiliary Building

The Reactor Auxiliary Building is provided with ventilation and cooling facilities, including independent cooling and ventilation systems for the Main Control

Room and the irradiated fuel bay and the emergency core cooling room.

The air conditioning system for the Main Control Room uses two independent air conditioning units connected to a common ducting system serving the Main Control Room and associated areas. Outside air is filtered and cooled prior to distribution. The fresh air intakes for these ventilation systems are separated, from one another and from potential sources of radioactive, chemical or gas contamination. The Air Conditioning System remains operational after a loss of Class IV power, and ensures that the Main Control Room equipment remains functional.

The irradiated fuel storage bay and the enclosing Reactor Auxiliary Building environment may be subject to contamination due to gaseous fission product releases from defected fuel bundles. To contain this activity and to minimize releases to the environment, a "once-thru" ventilation system, with air supplied from the ventilation system serving the adjacent areas is provided to prevent leakage of air and contamination via shipping doors, etc.. The air exhaust system maintains the areas at a slightly negative pressure in the irradiated fuel bay area. The exhaust air is continuously monitored for activity and if necessary, is passed through a series of high efficiency particulate filters and specially treated charcoal radio-iodine absorbers prior to release to the atmosphere.

4.0 NUCLEAR STEAM PLANT SERVICES

4.1 Introduction

This section describes the Nuclear Steam Plant (NSP) services which are located in the Group 1 Service Building, Group 2 Service Building, Maintenance Building and the Group 2 Pumphouse. This section also describes heating, ventilation and air conditioning for the above buildings and the radioactive waste management systems.

The Group 1 Service Building and the Maintenance Building house Group 1 services, while the Group 2 Service Building accommodates Group 2 services.

4.2 Buildings and Structures

4.2.1 General

The arrangement of the Maintenance Building, Turbine Building, Group 1 Service Building, Group 2 Service Building and principal auxiliary buildings relative to the Reactor and Reactor Auxiliary Buildings is shown in Figures 2.1-1 and 2.2-2.

4.2.2 Group 1 Service Building

The Group 1 Service Building, located on the B side of the Reactor Building, houses equipment which serves the normal power production requirement of the nuclear steam supply system. This equipment includes the Recirculated Cooling Water System pumps and heat exchangers, the water chillers and Group 1 motor control centers and electrical supplies. This building connects to the Main Control Room located in the Reactor Auxiliary Building. In normal operation, systems in this area are not exposed to any radiation fields.

4.2.3 Group 2 Service Building

The Group 2 Service Building is located on the A side of the Reactor Building and houses Group 2 systems. These include the essential cooling water supplies, electrical power supplies and the Secondary Control Room. The Secondary Control Room provides control and monitoring of all the systems required for the safe and maintained shutdown of the plant. This building is connected to the Reactor Auxiliary Building via an enclosed passageway.

4.2.4. Maintenance Building

The Maintenance Building, on the D side of the Reactor Building, provides the facilities necessary for the maintenance of the plant. These include electrical instrumentation and mechanical shops, change rooms, plant stores, and health physics laboratories. Heavy water management and spent resin storage facilities are provided in the basement area of the Maintenance Building. Areas containing radioactive components have separate atmospheric control systems.

4.2.5 Group 2 Pumphouse

The Group 2 pumphouse, containing the Group 2 raw service water pumps, travelling screens and trash racks, is located remotely from the Group 1 pumphouse. It has a reinforced concrete substructure and a steel frame superstructure with suitable architectural cladding.

An intake duct is provided, the length of which depends on the distance between the Group 2 pumphouse and the water source of sufficient depth. A water pipe transports the Group 2 cooling water from the pumphouse to the Group 2 service building. A return pipe channels the water back to the outlet bay.

4.3 Nuclear Steam Plant Common Process and Services

4.3.1 Heavy Water Management

The heavy water management systems are housed within the Maintenance Building. The D_2O Supply System receives and stores D_2O from drums or tank trucks, and pumps it to the Moderator or Heat Transport Systems during initial filling. In addition, the D_2O Supply System is capable of containing the inventory of the Moderator or the Heat Transport System in the event that draining for maintenance is required. The tanks can also store high isotopic D_2O during normal reactor operation.

The D_2O Cleanup System removes dissolved, particulate and organic impurities from D_2O recovered from various process systems and produces a product suitable for upgrading.

4.3.2 Water Systems

- **General**

The water systems are divided into the following two independent groups:

- Group 1 systems supply cooling water to normal plant processes and provide demineralized water, domestic water and chilled water to station users.
- Group 2 systems, which are seismically and environmentally qualified, supply cooling water to safety and safety-related systems including one of the shutdown cooling heat exchangers. These consist of the Group 2 Raw Service Water, Group 2 Recirculated Cooling Water and Group 2 Feedwater Systems.

- **Raw Service Water**

The raw service water (RSW) systems are once-through raw water cooling systems. Two independent raw service water systems are provided. The Group 2 Raw Service Water System provides cooling to the Group 2 Recirculated Cooling Water (RCW) System through the Group 2 RCW heat exchangers.

The Raw Service Water System cools the Recirculated Cooling Water System through the RCW heat exchangers. The Raw Service Water System simplified flow sheet is shown in Figure 4.3-1; the Group 2 Raw Service Water System arrangement is similar.

• **Recirculated Cooling Water**

There are two independent recirculated cooling water systems. The Group 2 Recirculated Cooling Water System supplies safety related and special

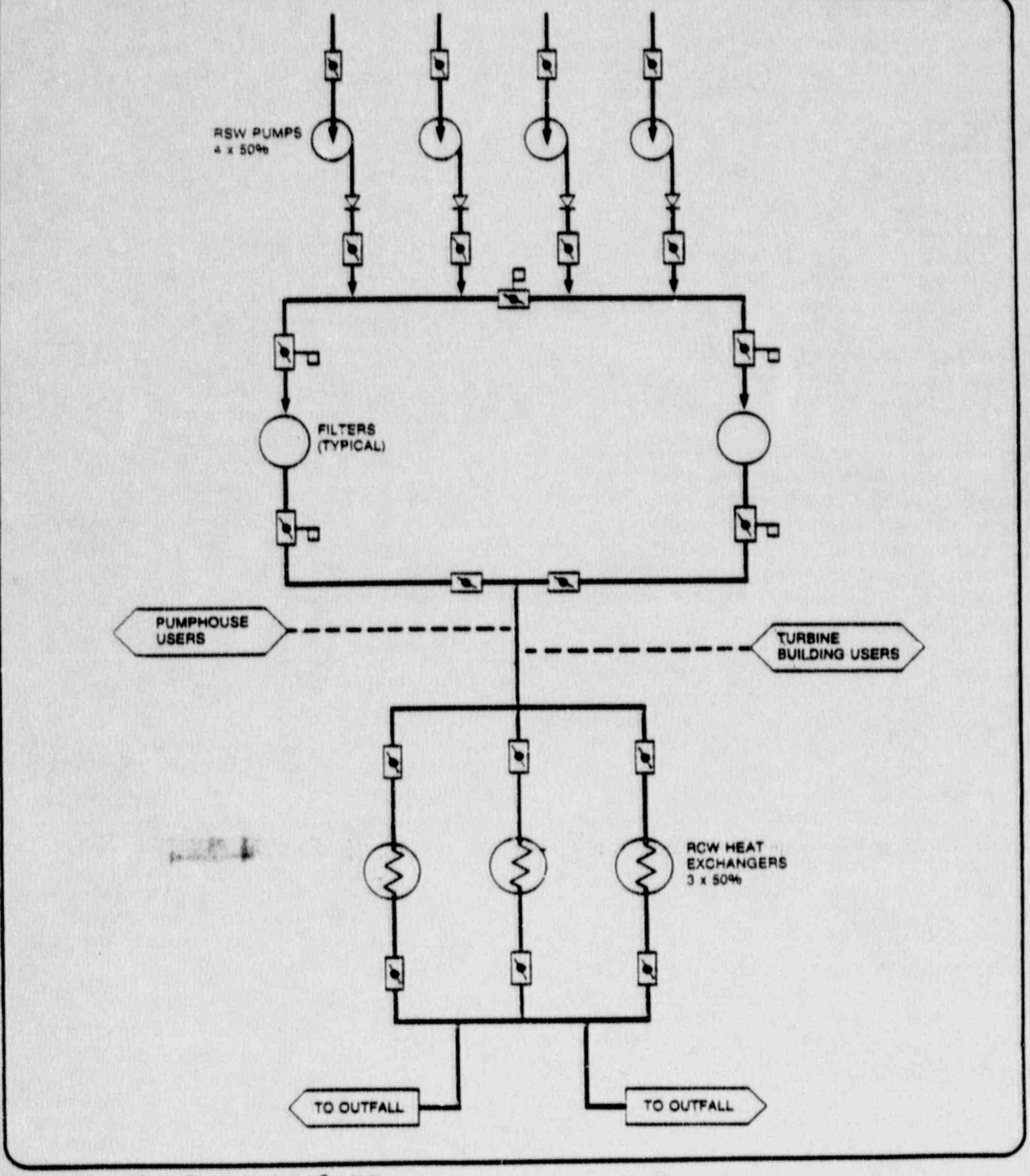


Figure 4.3-1 Raw Service Water System

safety systems, while the Recirculated Cooling Water System serves the Nuclear Steam Plant components used for power production plus the turbine-generators and auxiliaries. The Recirculated Cooling Water System simplified flow sheet is shown in Figure 4.3-2; the Group 2 Recirculated Cooling Water System arrangement is similar.

- **Chilled Water**

The Chilled Water System supplies chilled water at controlled temperatures to the various ventilation and air-conditioning systems in the plant, local air coolers in the Reactor and Service Buildings when required by site conditions and the coolers for the D_2O vapour recovery system, and the Main Control Room Cooling System. The chillers are located in the Group 1 Service Building.

- **Group 2 Feedwater System**

The Group 2 Feedwater System, located in the Group 2 Service Building supplies feedwater to the steam generators if the Feedwater System is unavailable.

The system is totally independent from the Group 1 Feedwater System, and is capable of supplying feedwater at rated pressure to the steam generators for decay heat removal.

4.3.3 Ventilation and Air Conditioning

- **General**

The ventilation and air conditioning systems for the buildings provide a controlled environment for personnel and equipment. The detailed design and capacities of these systems are dependent on site

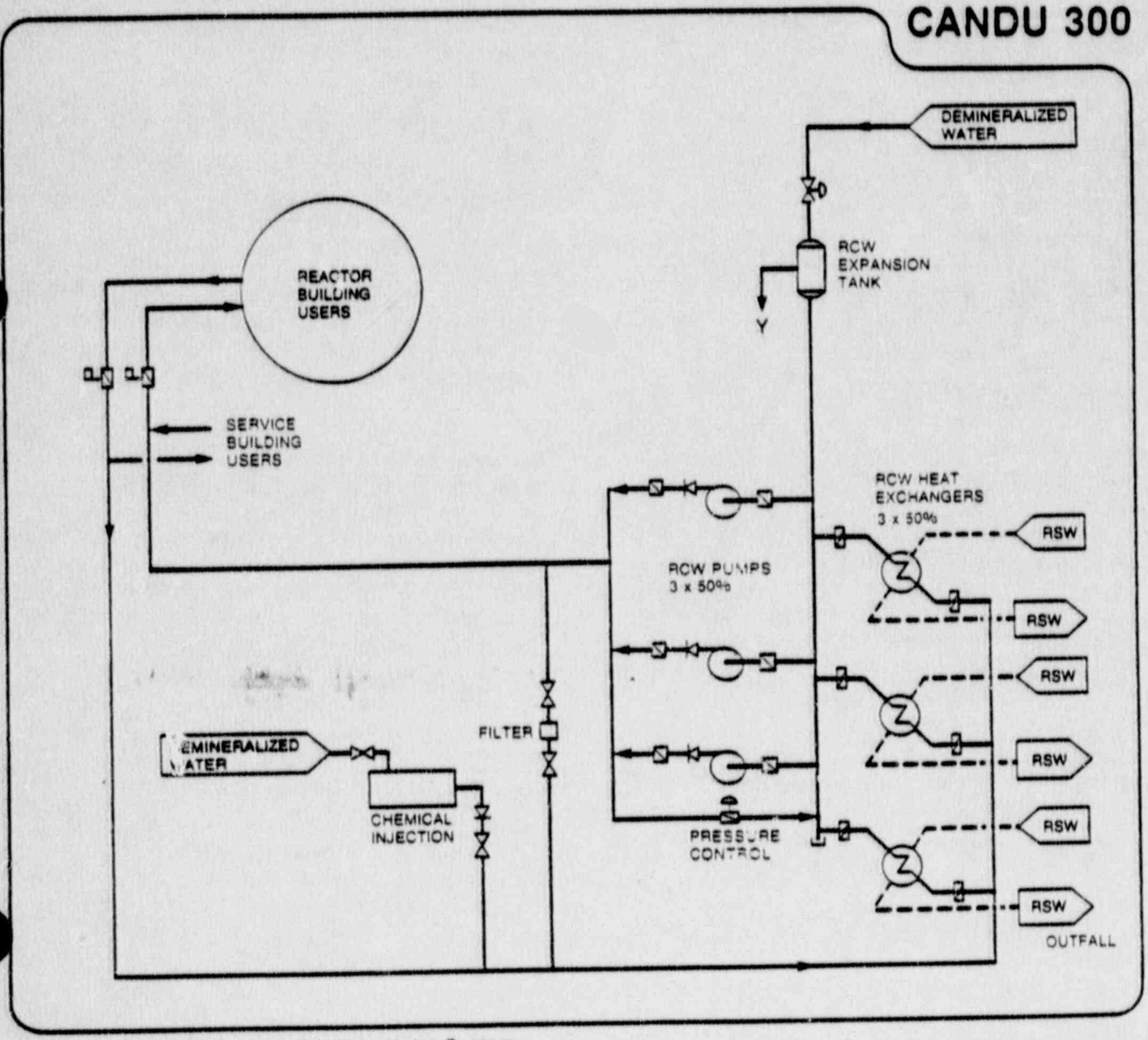


Figure 4.3-2 Recirculated Cooling Water System

and environmental conditions. The station layout facilitates the efficient arrangement of ventilation and air conditioning systems to maximize environmental control and minimize activity transport.

In areas containing potential sources of activity the ventilation and air conditioning systems provide air decontamination facilities that minimize the spread of any activity released during equipment operation or maintenance and prevent the uncontrolled release of this activity through containment leakage.

- **Group 1 and Group 2 Service Buildings**

The Group 1 and Group 2 service buildings are not subject to radioactive contamination during normal station operation.

The Secondary Control Area located in the Group 2 Service Building uses ventilation and air-recirculation systems having 100 per cent redundancy in active components. The air-conditioning systems for the Secondary Control Area are of the conventional air recycling type, and ensure a suitable temperature and humidity in the various rooms, as well as a supply of fresh air.

- **Maintenance Building**

The Maintenance Building has independent heating, ventilation and air-conditioning supplied by the following systems according to the requirements in various areas:

- Central ventilation supply air,
- Non-contaminated exhausts,
- Instrument workshop and health physics air-conditioning,
- Chemical room air-conditioning,
- Contaminated exhaust.

When required, heating of the air-conditioned areas is provided by air-conditioned unit heating coils and local electric heaters. The non-air-conditioned areas are heated by heating coils in the central air supply supplemented by local electric unit heaters.

4.3.4 Compressed Gases

The compressed air needs of the NSP are supplied by the Compressed Air System located in the Turbine Building. Miscellaneous compressed gases are supplied to various nuclear systems from bottle stations located in the Group 1 Service Building. These include:

- Hydrogen for injection into the Heat Transport System,
- Helium for moderator cover gas, injection shutdown and D_2O storage tank and heat transport D_2O collection tank,
- Oxygen for recombination units.

4.3.5 Radioactive Waste Management

- **General**

Radioactive waste management systems permit the on-site collection, processing and handling of all radioactive wastes produced at the site. Facilities are provided for either interim site storage or for disposal by the utility at an on-site or off-site location. The on-site facilities are designed such that the average public exposure from radioactive materials at the exclusion boundary does not exceed 1 per cent of the recommended ICRP limits and that the instantaneous exposures do not exceed 10 per cent of these limits.

The radioactive wastes produced at the site may belong to one of the following categories:

- Irradiated fuel
- Solid wastes
- Liquid wastes
- Gaseous wastes.

The irradiated fuel storage is discussed in Section 3.6.2.

- **Solid Radioactive Waste Management System**

Solid radioactive waste is divided into two general categories: maintenance wastes and purification wastes. The design includes facilities to collect both of these materials in the plant and prepare them for on-site interim storage. Figure 4.3-3 shows the processing and storage flow diagram.

- **Maintenance Wastes**

Maintenance wastes, originating from reactor maintenance operations, consist of cleaning materials, protective clothing, contaminated metal parts, and miscellaneous items. Waste originating from certain radiological areas is often automatically considered radioactive even though they may contain no radioactivity. Approximately 90 per cent of the waste has contact fields less than 0.5 mR/h.

Maintenance wastes are further classified as compactible if they can be volume reduced by compaction to have a specific activity of approximately 0.04 mCi/kg. Non compactible wastes have a specific activity of approximately 80 mCi/kg.

Included in the solid wastes may be a very small volume of radioactive liquid waste that cannot be handled in the radioactive liquid waste management systems. These wastes and some organic solvents, such as scintillation solutions, are immobilized using an inert absorbent.

- **Purification Wastes**

Purification wastes, originating from on-line reactor fluid purification systems, consists of filter cartridges and ion exchange resins. Typically this waste has an unshielded radiation field greater than 1R/h on contact. Consequently, additional shielding and greater precautions than for maintenance wastes are required during transportation, handling and storage operations.

- **Spent Filter Cartridges**

Spent filter cartridges from various purification systems are handled by two shielding flasks to protect personnel from radiation during removal and transportation to the radioactive waste storage area.

Each filtration system is designed so that a filter vessel containing a filter assembly requiring replacement is isolated, depressurized, drained and purged of free-standing water before filter removal is started. This method minimizes the problem of handling wet filter assemblies with a potential associated tritium exposure and/or spread of contamination.

The filters are then sent for storage at the on-site interim radioactive waste storage area.

- **Spent Ion Exchange Resins**

Spent radioactive ion exchange resins are produced in the purification systems. The spent resins from each of these systems are transferred, using demineralized light water, into one of the epoxy-lined, concrete spent resin storage vaults, located in the maintenance building. When the water in a tank reaches its normal operating level, the water overflows into the external collection sump.

When high water level is detected in the sump, the sump pump transfers the water to the radioactive liquid waste management system.

The typical solid radioactive waste storage area at a Canadian CANDU plant provides engineered structures in the form of storage buildings or trenches for wastes with low specific activities and cylindrical tubes for wastes with high specific activities. The facilities are designed so that the presence of water within the structure can be detected and removed, and any water leakage from the facility can be intercepted before it reaches the natural

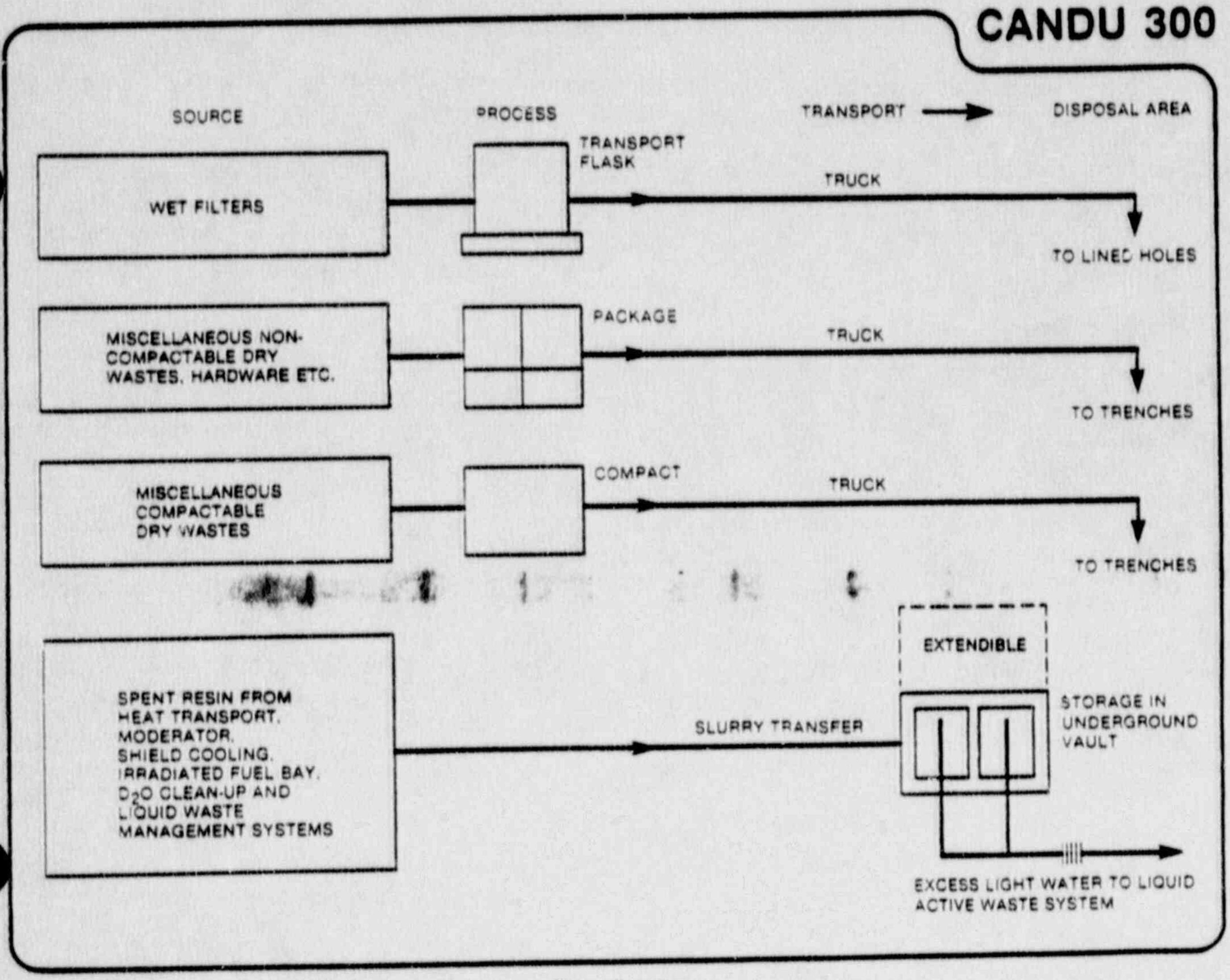


Figure 4.3-3 Typical CANDU active solid waste flow diagram

ground water. Based on operating experience, the site illustrated in Figure 4.3-4 is capable of storing the lifetime production of radioactive solid wastes.

- **Liquid Radioactive Waste Management System**

The Liquid Radioactive Waste Management System, shown on Figure 4.3-5 provides collection, storage, sampling, and necessary decontamination, and dispersal of any liquid waste produced by the station. The system is designed to control the release of radioactivity in the liquid effluent streams to a radiation dose to a member of the public of less than 5 mrem/a based on the Derived Emission Limits for each radionuclide.

This system handles radioactive wastes that are carried in liquid streams from the laundry, active floor drains, decontamination centre and chemical laboratories. It provides storage, sampling, necessary treatment and dispersal into the condenser cooling water duct under controlled conditions. Three sources of active liquid wastes are identified as listed below:

- Low Activity Wastes

These wastes originate in such areas as change rooms, the laundry and the non-active laboratories. The wastes are collected in one of the two low activity tanks.

- **Normal Activity Wastes**

These wastes originate from the decontamination centre, active laboratories, the heavy water upgrading column, excess water from ion exchange resin slurrying, Reactor Building drains, and the rubber goods laundry. The wastes are collected in one of two tanks.

- **Special Source Wastes**

Wastes from the special sources such as decontamination wastes are normally treated as active wastes, but they may be treated directly before entering the active waste tanks.

The Radioactive Liquid Management System consists of four concrete storage tanks, which are located in the Maintenance Building. Two of the tanks are used for low activity wastes while two are used for the normal activity liquid wastes.

CANDU 300

Figure 4.3-4 Typical waste storage site

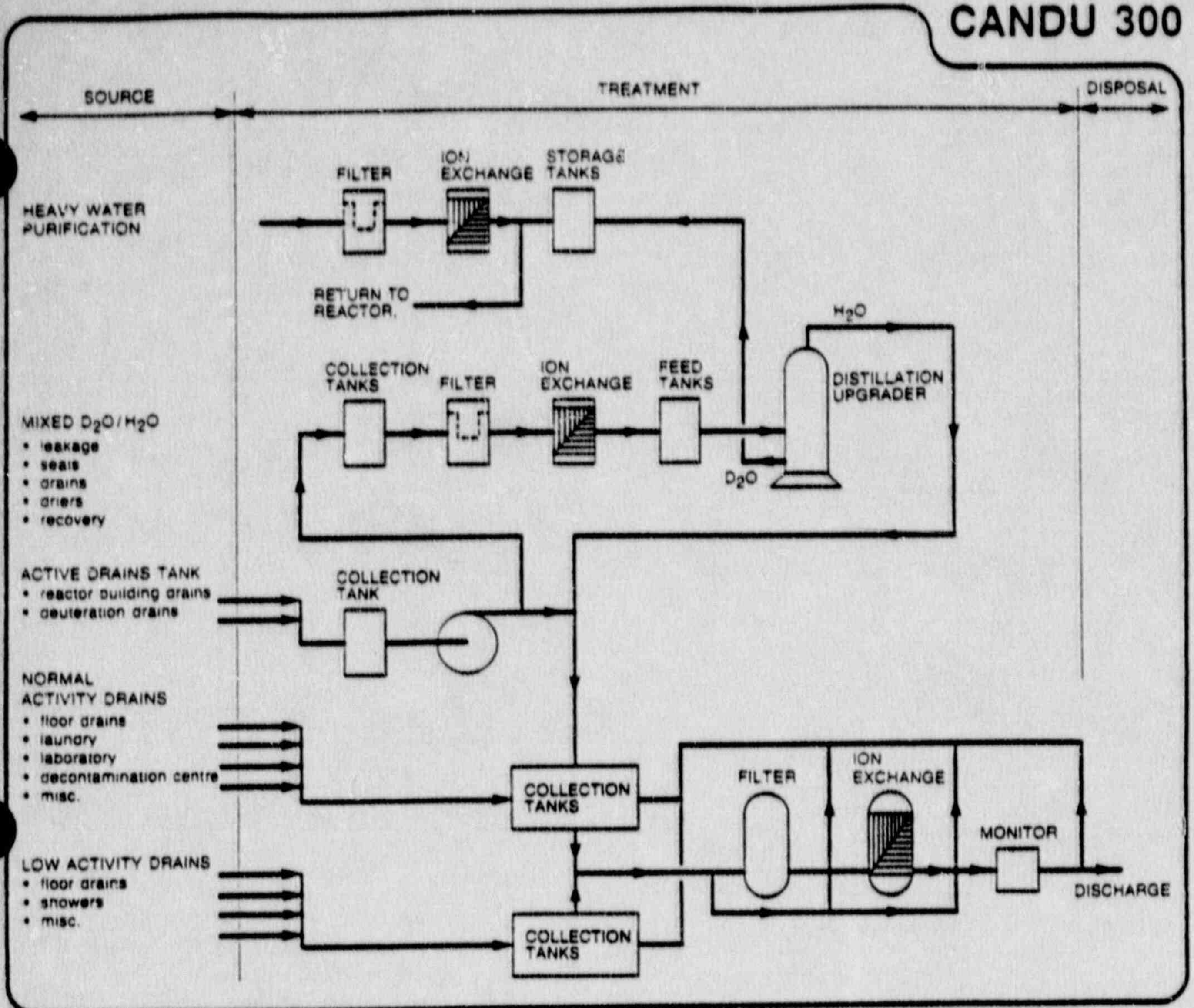


Figure 4.3-5 Liquid waste management

Should decontamination be required to reduce radioactivity from some of the normal activity wastes so that effluent concentration does not exceed regulatory release limits, the effluent is treated using a filter and an ion exchanger.

As part of the design process, the effluents are divided into those whose specific activity, excluding tritium, is never expected to exceed 0.4 Bq/mL* and those whose specific activity, excluding tritium, may on occasions exceed 0.4 Bq/mL. The latter may require processing prior to discharge to the cooling water discharge channel. A filter/demineralizer is provided for such processing. Following dilution by the cooling water, the resultant specific activity does not exceed 0.4 mBq/mL.

All liquid effluents are sampled three times during their passage through the system. Each collection tank is sampled when it is filled and the results from the analysis used to determine what treatment, if any, is required prior to discharge. During each discharge a sample of the undiluted effluent is passed through an integrating liquid effluent monitor, and should this indicate that the total activity being discharged exceeds the permissible limits, the discharge is automatically terminated. The third sample is taken from the condenser cooling water discharge duct and is used as a final check on the quantity of activity released.

Operating experience of Canadian Nuclear Power Plants shows that CANDU reactors operate well within the 1 per cent of the ICRP Limit Target set by Canadian utilities.

- **Gaseous Radioactive Waste Management System**

An extensive ventilation system shown in Figure 4.3-6 collects potentially active exhaust air from such areas as the Reactor Building, the irradiated

*Bq = Becquerel = 1 disintegration per second.

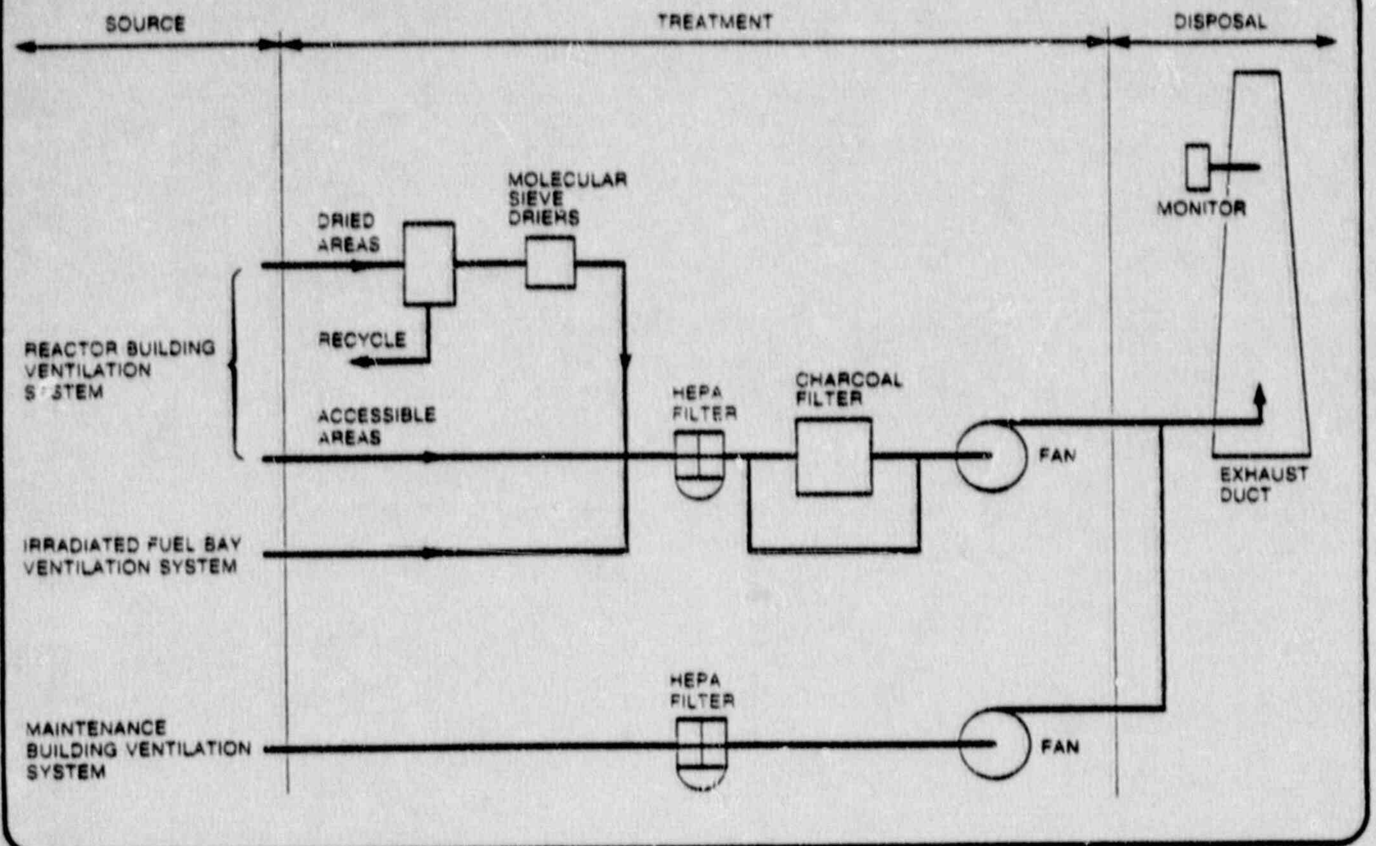


Figure 4.3-6 Radioactive gaseous waste storage

fuel handling and storage area, the decontamination centre and the heavy water management area. The active and potentially active exhaust air flows and gases are all routed to a gaseous effluent exhaust duct. This exhaust flow is monitored for active noble gases, tritium, iodine and active particulates before being released. Facilities for filtration are provided. Signals from the iodine, widerange gamma and particulate monitors are recorded in the control centre. Tritium monitoring is carried out by laboratory analysis of bubbler samples.

5.0 BALANCE OF PLANT

5.1 Introduction

The BOP comprises the Turbine Building, Pumphouse and the Switchyard, including the systems, equipment and facilities housed within them. A summary description of the BOP is presented in the following sections.

5.2 Buildings and Structures

5.2.1 Turbine Building

The Turbine Building is a steel frame structure and consists of a turbine hall with a water treatment annex. The Turbine Building is connected to the Reactor Auxiliary Building via an enclosed above ground structure housing the umbilicals and serving as a personnel access route. The Turbine Building houses the turbine generator and its auxiliary process and electrical equipment, the water treatment plant and the station instrument, service and breathing air supplies.

5.2.2 Pumphouse

The Pumphouse structure has a reinforced concrete substructure. The superstructure is structural steel, masonry clad and insulated. This structure houses the following major equipment: condenser cooling water pumps, raw service water pumps, screen wash pumps, trash racks, screens, and chlorination equipment.

5.3 Balance of Plant (BOP) Systems

5.3.1 Turbine Generator and Auxiliaries

The turbine-generator, feedwater and condensate plant is of conventional design following standard commercial practice. These systems are usually influenced by the requirements, preferences and practices of the utility and the standards and practices of the turbine-generator supplier. There are, however, requirements specified for the Turbine Island to assure performance and integrity of the Nuclear Steam Plant. These include materials requirements (Titanium condenser, absence of copper alloys in the feed train, etc.), feed train reliability requirements, feedwater inventory requirements and condenser steam discharge capability.

Site differences affect condenser cooling water design temperature which, in turn, affects turbine exhaust conditions and the amount of energy it is possible to extract from the steam. CANDU stations have been constructed and operated with North American, European and Japanese turbine-generators operating on fresh water, contaminated water, and sea water condenser coolant supplies with temperatures ranging up to 32°C.

The turbine assembly for a cold water site typically consists of one 2 flow high pressure turbine and two 2 flow low pressure turbines, operating at 1800 rpm. For a warm water site a full speed turbine generator may be utilized.

The condenser consists of separate shells, one per LP turbine casing. The condenser is of a double tube sheet design with the tube sheet interspace pressurized by condensate, thus eliminating leakage of condenser cooling water into the condenser.

The turbine extraction steam system typically supplies five stages of feedwater heating. The low pressure regenerative feedwater heating system consists of a single bank of three low pressure closed feedwater heaters, and the high pressure consists of a single high pressure heater. In addition, the deaerator is heated with the drains from the high pressure heater, or if unavailable, with pegging steam from the steam mains.

The feedwater system includes three 50 percent capacity electrically driven main feedwater pumps that take suction from the deaerator storage tank and two 3 percent capacity auxiliary feedwater pumps, one of which is steam turbine driven.

A turbine by-pass system can discharge steam directly to the condenser when the turbine is unavailable, in order to prevent a reactor poison-out.

6.0 BALANCE OF PLANT SERVICES

6.1 Introduction

Balance of Plant (BOP) services include supply water systems, heating, ventilation, air conditioning, chlorination, fire protection, compressed gas and electric power systems.

6.2 Supply Water Systems

The water systems provide cooling water, demineralized water and domestic water to station users. They comprise a pumphouse, a water treatment plant, a chlorination system, various cooling water systems and a domestic water system. Group 1 pumphouse systems, Condenser Cooling Water System and water systems within the Turbine Building are part of the BOP.

6.2.1 Group 1 Pumphouse Systems

The Pumphouse mechanical equipment supplies raw water via screens to:

- Condenser cooling water pumps
- Raw service water pumps
- Screen wash pumps

The raw water enters the pumphouse through intake passages that supply raw water to the condenser cooling water (CCW) pumps and to the other raw water system pumps.

6.2.2 Condenser Cooling Water System

The Condenser Cooling Water System supplies once-through cooling water to the main condensers. The system pumps cooling water through the main condenser in sufficient quantities to condense turbine exhaust steam and to maintain rated back pressure conditions at the turbine exhaust. The system components and materials are specified to minimize deterioration of the condenser heat transfer capability under normal operating conditions and to ensure a high degree of availability.

6.2.3 Raw Service Water System

The raw service water is supplied to the RSW/RCW heat exchangers located in the Group 1 Service Building for all the station Group 1 loads.

The Raw Service Water System in the Turbine Building is a once-through system and consists of:

- RSW supply and return from condenser air exhauster.
- RSW supply and return from turbine lube oil and other coolers.

The RSW return from the Turbine Building discharges into the condenser cooling water return duct.

6.2.4 Recirculated Cooling Water System

The RCW system within the Turbine Building supplies cooling water to the following equipment:

- Turbine hydrogen, stator water coolers
- Mechanical seal coolers of all pumps
- Lube oil cooler of feedwater pumps
- Other miscellaneous coolers within the turbine building

6.2.5 Water Treatment Plant

The water treatment plant, integrated with the Turbine Building, supplies water used by the Domestic Water System and demineralized water make-up water to the Condensate System, the Chilled Water System, the RCW and other systems. Demineralized water is stored in sufficient quantity in both the Group 1 and Group 2 systems to meet the station requirements during both normal and emergency conditions.

The Domestic Water System supplies treated water from the clearwell of the water treatment plant for personnel, laboratory and miscellaneous plant system uses.

6.2.6 Chlorination

Chlorination systems are used for treatment of domestic water, fresh water supply to the pretreatment plant, raw service water and the condenser cooling water if necessary. Two separate chlorination systems are provided. One system is located in the water treatment plant and the other system is located in the pumphouse.

6.3 Heating, Ventilation and Air Conditioning

The Building Heating Plant satisfies the steam and hot water demands of the station. Steam extracted from the turbine is the normal "building heating" steam source.

The auxiliary boiler is an optional package unit which is delivered as a module.

Ventilation and air conditioning is supplied to the BOP structures to ensure a suitable environment for personnel and equipment during winter and summer.

6.4 Compressed Gases

The Compressed Air Systems, shown in Figure 6.4-1, supply instrument air, service air, and breathing air as required to both Group 1 and Group 2 areas of the station. These systems are supplied by 2 x 100 per cent oil-free water cooled compressors located in the Turbine Building via moisture separators and air receivers.

The Instrument Air System, which includes prefilters, dual-shell air dryers, after filters and a pressure sta-

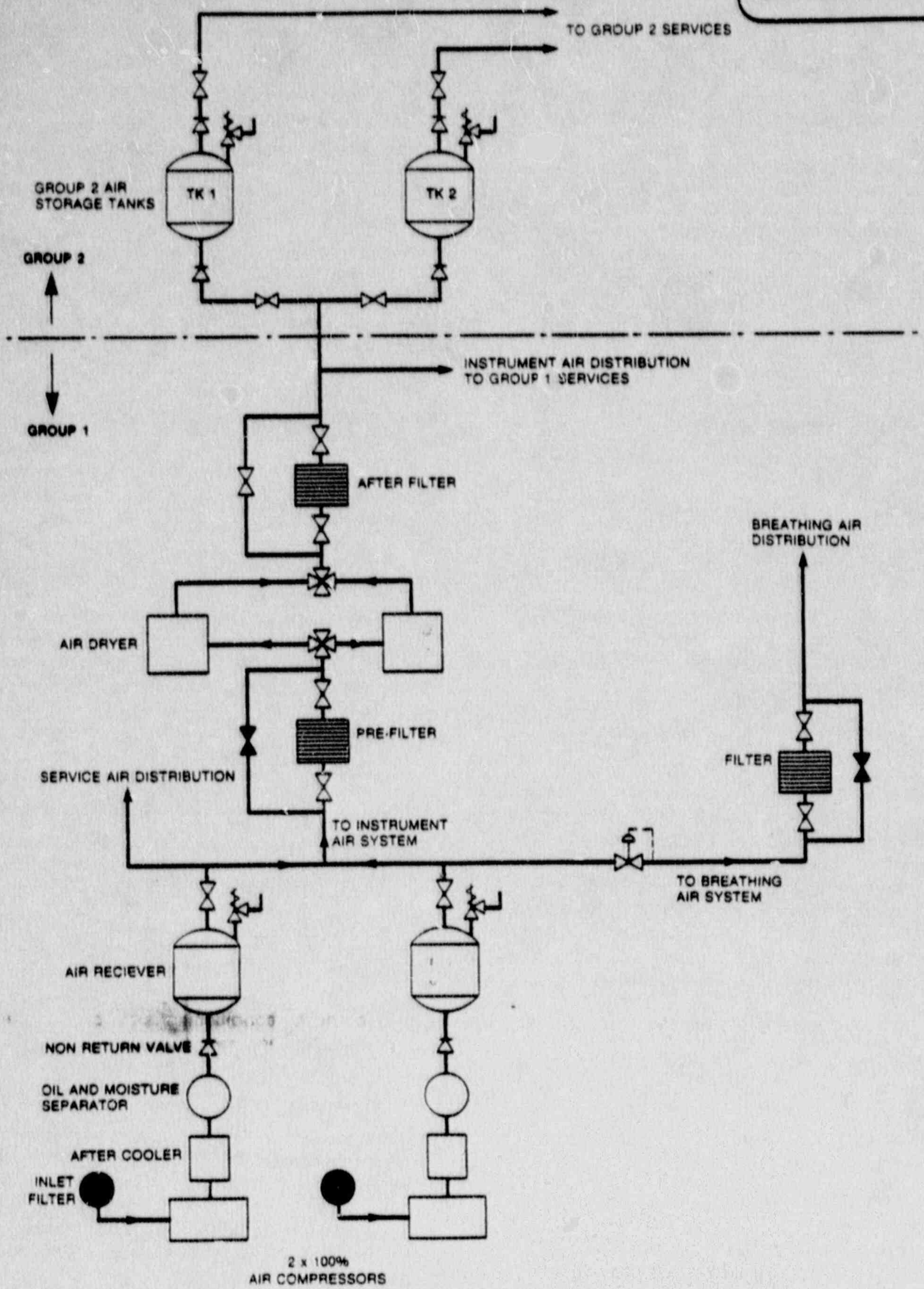


Figure 6.4-1 Compressed air systems

bilizer, supplies low dew-point, low-particulate air to both Group 1 and Group 2 areas. The Group 2 air supply is assured by two air storage tanks located in the Group 2 Service Building.

Service air is supplied to the Service Air distribution system to all shops, air tools, and hydro-pneumatic tanks.

Low-particulate, oil-free air is supplied by the Breathing Air System to serve face masks and plastic suits for breathing and body cooling. The air from the supply system travels through a carbon particulate filter before entering the distribution system.

Miscellaneous compressed gases are supplied to various nuclear and conventional systems including:

- Carbon dioxide for the reactor fuel channel annulus gas system and for purging the turbine generator.
- Hydrogen for cooling the turbine generator.
- Nitrogen for blanketing the Heat Transport System and the feedwater heaters.

7.0 STATION SERVICES

7.1 Communications

7.1.1 General

The Communication System includes the Telephone System, the Public Address System, the Maintenance Communication System and the Plastic Suit Communication System. These systems serve the entire CANDU 300 station.

7.1.2 Telephone System

The internal Telephone System provides communication between working areas within the station, and has the capability of simultaneous conversations.

Power supply is from Class IV supply with automatic switch-over to a standby battery system with an eight hour capacity. The equipment racks and power supply including batteries are housed in the telecommunications room.

7.1.3 Public Address System

A Public Address System is provided for paging personnel and for issuing routine, operational and emergency instructions to the station operators.

The Plastic Suit Communication System is used in lieu of the Public Address System for personnel wearing suits. The public address inputs to this system are controlled by a push-button on the station operator's call director.

An alarm tone generator is included and can be switched into the Public Address System by the control room operator. Two separate and distinct tones are provided; one for fire and the other for radiation hazard.

7.1.4 Maintenance Communication System

This system provides ready means of communication between all points in the plant and common services area without tying up the main telephone systems. It is a simple telephone system in which two or more points can be connected by means of a patch panel in the control equipment room.

7.1.5 Plastic Suit Communication System

Plastic suits are worn by maintenance workers in designated areas within the reactor building and maintenance building. The Communication System allows for normal conversations through the use of head-sets.

The plastic suits are designed to be connected to wall-mounted breathing air stations located at strategic locations. Each wall outlet provides both communication and air supply using combined connectors.

7.2 Clock System

The Clock System consists of a master clock, synchronized to the supply line frequency, which maintains repeater clocks in synchronism throughout the plant. The system is self-correcting in the event of power loss or other disruptions.

7.3 Meteorological Monitoring

Instrumentation is provided to measure wind speed and direction, barometric pressure and atmospheric temperature. The instrumentation is connected to recording and read-out equipment in the control equipment room and Main Control Room.

7.4 Fire Protection

7.4.1 General

The concept of fire protection is to provide a defense-in-depth approach towards fire prevention, fire detection and suppression and mitigation by optimizing layout and structures.

7.4.2 Fire Prevention Measures

The requirement to reduce combustible contents is incorporated into the design of CANDU 300.

As part of the fire hazards assessment, an inventory of combustible content in the plant is prepared. This inventory forms the basis for the detailed design of fire protection systems. It also serves as input to the preparation of procedures for fire prevention in the design and operation stages.

7.4.3 Fire Detection System

A Fire Detection System provides early warning of fires in the plant and to actuate certain automatic fire extinguishing systems. It consists of automatic detector units, manual pull stations, the control unit and alarm display unit.

The control unit is microprocessor-based and provides constant supervision of the state of the system and all circuits. Display of system trouble and fire alarm messages is on CRT and printer. In addition to providing alarms, the system is programmed to control halon systems, supervise sprinkler valves and to control fire dampers and ventilation fans.

The fire alarm display terminal is located in the Main Control Room and an auxiliary printer is located in the Secondary Control Area.

7.4.4 Mitigation of the Effects of Fire

The Fire Protection System design considers the possibility of failures of automatic or manual fire systems to extinguish a fire. The distinct Group 1 and Group 2 separation guarantees that if a fire can be confined to its area of origin, the safety functions of at least one group is maintained.

Buildings are laid out such that they are connected only at access points. This greatly reduces the potential of fire spread between buildings. Connections between buildings are provided with three-hour rated fire walls equipped with fire doors.

Within each building, a number of fire zones are established. Each fire zone is separated from another by fire walls. These fire areas establish the practical limits of any single fire.

All penetrations for cables, pipings and air ducts through fire cells and fire areas are sealed with a fireproof material.

7.4.5 Firewater System

The water supply is taken from a reliable source of fresh water such as a lake, a large river or a large pond. Two fire pumps are provided each with 100% design capacity. One pump is driven by diesel engine while the other is driven by electric motor. A small jockey pump is provided to maintain pump header pressure.

The fire pumps discharge into the firewater main ring which surrounds the plant and supplies fire hydrants and indoor fire systems. The system is sized to meet the requirements of the largest design basis fire for a minimum of two hours.

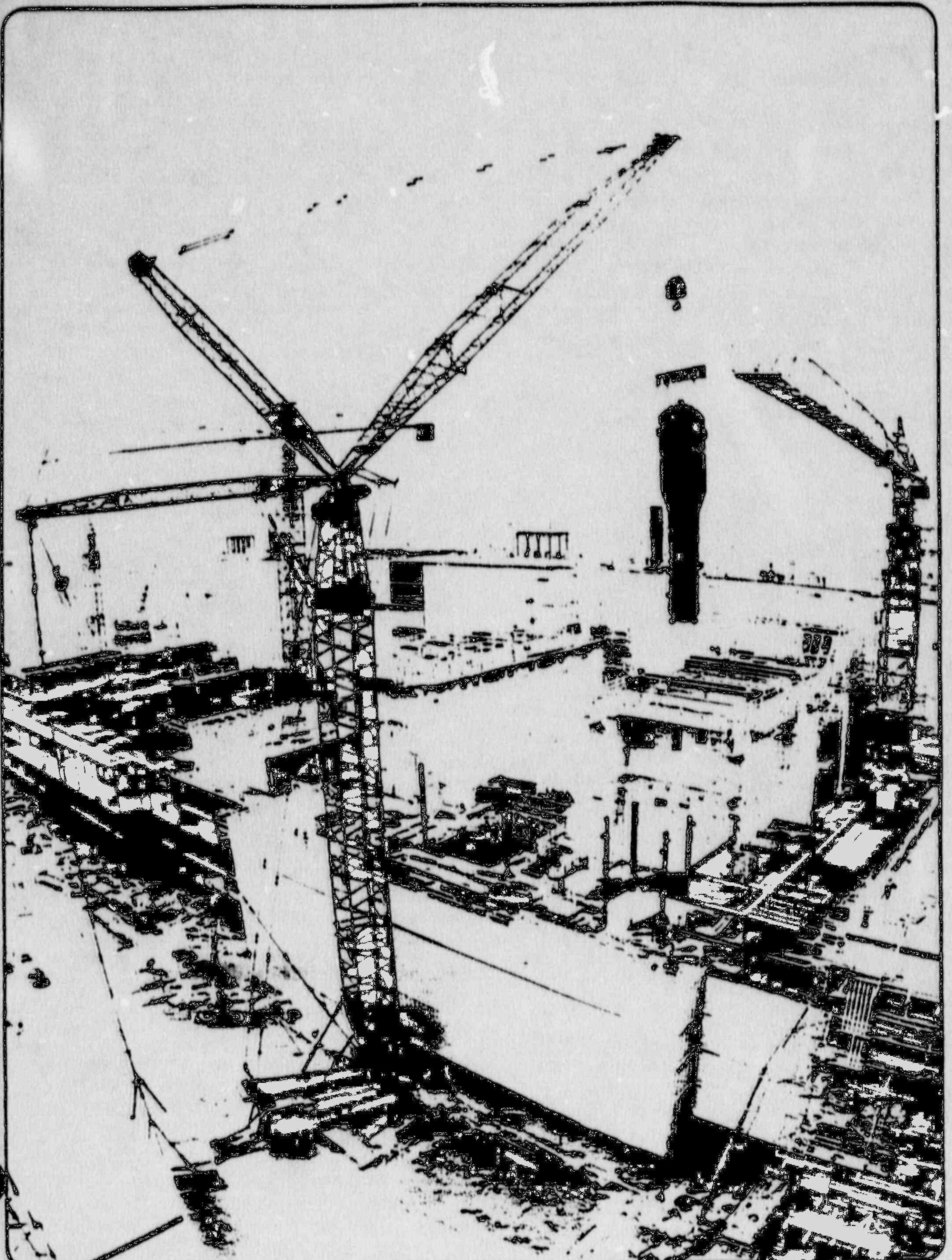


Figure 8.1-1 VHL crane placing 350 tonne steam generator at Darlington CANDU station

8.0 CONSTRUCTION

8.1 General

A reduced construction schedule saves interest during construction (IDC), lowers capital costs and gives greater flexibility to the utility in adjusting its generating capacity increase program to meet actual load growths.

In the last few years, very heavy lift (VHL) construction cranes as shown in Figure 8.1-1 have become generally available on the market. VHL cranes open up significant opportunities for the reduction of schedule, namely:

- Use of "Open Top" construction with the Reactor Building designed for vertical instead of horizontal access during construction.
- Modularization

These two modern construction methods utilized in the CANDU 300 are discussed in the following sections.

8.2 Open Top Construction

The dome of the Reactor Building is left off until near the end of the construction program. This eliminates the need for temporary construction openings, which results in a significant cost and schedule saving, since temporary openings are costly and time consuming to incorporate in the Reactor Building structure.

The "open top" allows good access to the interior for construction of concrete internals and for the installation of heavy equipment and modules. Equipment and modules are fabricated in large pieces and installed quickly and efficiently into the building, using the VHL crane. For example, a steam generator can be installed in less than one day, where use of the old horizontal access method required over two weeks. Since the Reactor Building is designed to accommodate equipment installation from above (vertical access), there is no need to delay installation of process equipment and piping in any one area in order to allow passage of large equipment into another area.

8.3 Modularization

Modularization, facilitated by the station layout, contributes substantially to cost and schedule reduction on the CANDU 300. In addition, modularization opens up a new dimension of flexibility in both construction methods and component supply.

The contents of each building of the CANDU 300 are subdivided into modules on a system or subsystem basis. Interfaces between modules are simple to facilitate site assembly, and to minimize site construction time. Figures 8.3-1 illustrates Computer Aided Design and Drafting (CADD) produced drawings of the shutdown cooling module. Fuel channels are an example of component modularization; they are factory assembled, eliminating most field installation work and thereby shortening the schedule. Similarly, the steel calandria shield tank and the shield tank extension combined with the reactivity mechanisms deck are factory assembled as modules, eliminating field work.

The major advantages of modularization are as follows:

- **Parallel Module Construction**

The station modules are constructed in parallel with each other, and with the Reactor Building concrete and civil work. This yields a very significant schedule reduction compared to in-situ installation after completion of concrete.

The modules themselves are constructed in parallel and may be simultaneously constructed in many different countries or geographical locations, thereby allowing supply to be optimized based on financial, technical or contractural criteria. Significant schedule and cost reductions result.

- **Enhanced Productivity**

Modules are constructed in a shop environment utilizing developed fabrication facilities and trained personnel. Productivity is therefore much greater than for comparable in-place construction. The 360° access to the module, and the ability to move or rotate the module during construction, further allows many parallel fabrication activities, with various trades working together.

- **Quality Assurance**

Quality Assurance is more easily provided, at reduced cost, in the shop environment.

- **Reduced Site Facilities**

In the event that modules are fabricated off-site, site facilities for fabrication, construction equipment, and personnel accommodation are dramatically reduced. If the modules are fabricated in on-site fabrication shops, the centralized and organized facility reduces both schedule and cost relative to in-place construction.

- **Flexibility**

Modularization provides a high degree of flexibility in sourcing, since modules may be constructed in parallel at many different geographical locations.

The CANDU 300 modules are constructed complete, including structural steel framing, platforms and ladders, shielding walls, equipment, piping, supports, cable trays, conduit, lighting, junction boxes, instrument tubing and instrument racks. Hydrostatic or other testing, painting and thermal insulation are completed where possible. This minimizes the work after module installation.

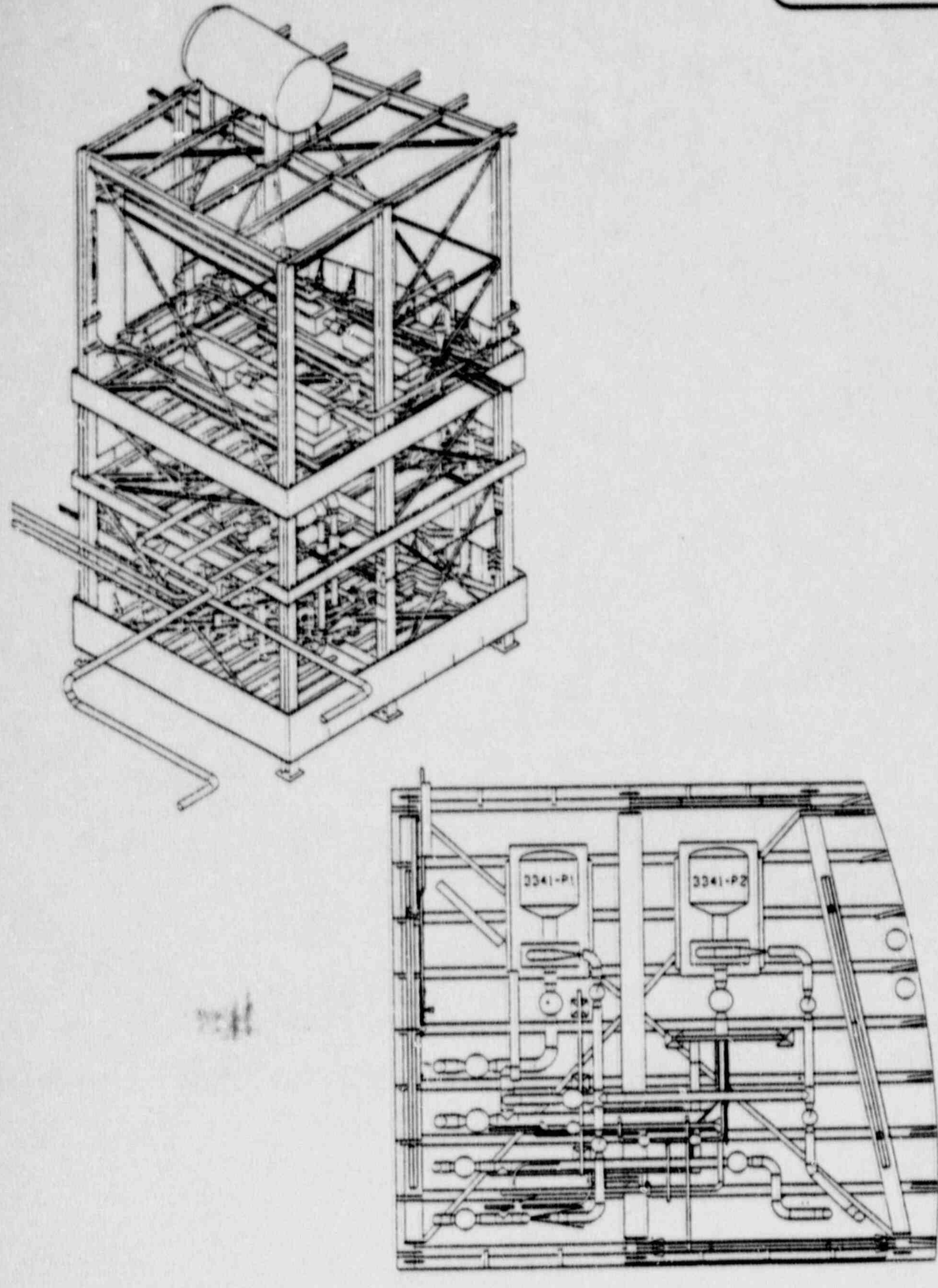


Figure 8.3-1 Shutdown cooling module

9.0 STATION AND COMPONENT LIFE

9.1 Station Life

With the exception of the Reactor Building structures and the calandria/shield tank assembly, which have design lives in excess of 100 years, all components of the CANDU 300 can be easily replaced. This assures that station life is not limited by component life.

9.2 Component Life

9.2.1 General

The following factors impact upon component life and are fully considered in determining the life of CANDU components.

- Creep and growth due to neutron irradiation
- Embrittlement due to irradiation
- Erosion and corrosion
- Fatigue
- Wear
- Thermal creep
- Obsolesence
- Other factors of degradation

Operating experience, experimental data and advanced analytical methods form the basis for determining component life. Life expectancy is verified during plant operation by a comprehensive in-service inspection program which includes the periodic removal of a fuel channel for laboratory inspection, and the LIFE (an acronym for Local In-Plant Fatigue Evaluation) transient logging and analysis program that facilitates an ongoing fatigue life evaluation for all key components. The LIFE system also facilitates identification of unpredicted or premature fatigue usage factor consumption and the subsequent modification of operating procedures.

All components of the CANDU 300 have a design life of 40 years or more, except the reactor assembly and the fuel channels which have design lives of 100 years and 24 years respectively. Many major components may therefore be replaced once or twice in the operating life of the station.

9.2.2 Reactor Assembly

The reactor assembly, replacement of which would require a 24 month outage, is designed to assure a life in excess of 100 years. A paramount consideration in establishing reactor assembly life is the effect of neutron irradiation on reactor materials.

The neutron fluxes and the integrated neutron fluxes for critical CANDU 300 calandria components over a 100 year operation are presented in Table 9. These values are significantly below the threshold fluences for detectable mechanical property degradation.

The effect of transmutations on CANDU 300 reactor assembly components has also been considered, and shown to be negligible.

There is a slight change in the absorption cross section of Type 304L stainless subjected to radiation. For example, after 100 years at a thermal flux of 2×10^{13} n/cm^2.s, the absorption cross section changes from 0.23 cm^{-1} to 0.19 cm^{-1}. This does not significantly impact on CANDU 300 core physics.

9.3 Component Replacement

9.3.1 General

Ease of component replacement offers two advantages to the station owner. One is plant life extension. As components reach the end of their operational life, they can be easily replaced, permitting the station life to be extended indefinitely. This not only avoids the cost of constructing replacement stations but also defers the cost of decommissioning.

The second major advantage of easy component replacement is the insurance value. In the event that any component reaches end of life prematurely due to any reason (design, manufacturing, or operating procedure deficiency for example) it can be easily replaced, therefore avoiding an extended outage.

A target of the CANDU 300 design is to facilitate the replacement, modernization, or rehabilitation of any aspect of the station within a 90 day outage. This could include steam generator replacement, fuel channel replacement, turbine internals replacement, or modernization of instrumentation.

TABLE 9
NEUTRON FLUXES IN CANDU 300 REACTOR COMPONENTS

	Max. neutron flux n/cm^2·s 1 MeV	Thermal	Integrated fast neutron flux over 100 years (n/cm^2)
Calandria mainshell	2.3×10^{8}	1.3×10^{13}	7.2×10^{17}
Calandria tubesheet	6.7×10^{11}	1.1×10^{13}	2.1×10^{21}
Calandria tubesheet/lattice tube weld	8.3×10^{11}	1.1×10^{12}	2.6×10^{21}

The CANDU 300 is designed utilizing "CANDID Engineering", an acronym for CANDU Integrated Design Engineering. A key element of CANDID Engineering is the full size electronic model of the station created during the design process. This model is used to simulate all major maintenance and component replacement activities. The simulation verifies lifting capability, personnel and component access and removal routes, clearances during component movement, radiation fields, and other data pertinent to maintenance or replacement activity.

9.3.2 Reactor Components

All in-core components can be replaced. The reactivity control units can be easily replaced (within two or three shifts) during routine outages, if necessary. Replacement of the fuel channels can be accommodated within a 90 day outage.