Crutchfield

ORIGINAL

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BRIEFING ON SYSTEMATIC EVALUATION PROGRAM

(Open to Public Attendance)

Commissioners' Conference Room
D.C. Office
Nuclear Regulatory Commission

Tuesday, May 6, 1980

The meeting convened, pursuant to notice, at 2:35 p.m.

Present:

CHAIRMAN John F. Ahearne
COMMISSIONER Victor Gilinsky
COMMISSIONER Richard F. Kennedy
COMMISSIONER Joseph Hendrie
COMMISSIONER Peter A. Bradford

Also present:

D. Crutchfield
W. Dircks
D. Eisenhut
H. Denton
E. Hanrahan
Mr. Malsch
L. Bickwit
J. Scinto

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P R O C E E D I N G S

CHAIRMAN AHEARNE: I would like, if I could, to call the Commission to the meeting again.

The purpose of this afternoon's second meeting is to discuss the ever-evolving plan of the systematic evaluation program. I remember what it was. It was just as I was going back digging through all these old SECY papers dating back to the Task Force Report of Nov. 1976 and SECY 76-545, decision memo and SECY 77-561 and decision memo and an ACRS letter and an answer to the ACRS letters.

MR. DIRCKS: That is the historical possibility.

CHAIRMAN AHEARNE: No. That has led me to realize that this is something we'd like to hear more about, to at least understand not only what is happening. I guess my first question I would have is, what is it?

Bill?

COMMISSIONER HENDRIE: Since we have been -- every time I turn around we have reorganized the SEP programs. Harold has had major churnings out there, so one of the early things I would like to hear is who has now got this ball so that I know who to glare at.

MR. DENTON: Darrell Eisenhut.

COMMISSIONER HENDRIE: And who works it for you, Denny?

MR. EISENHUT: The Branch Chief is the SEP program branch. It is now vacant. Denny runs the Projects Branch, which is sort of the routine branch.

COMMISSIONER HENDRIE: The project side of it.

MR. EISENHUT: The Assistant Director of it is Gus Lance.

COMMISSIONER HENDRIE: Congratulations, Gus. What happened? Did everybody run away from that branch? It is vacant because nobody will accompany it?

Never mind.

(Laughter)

COMMISSIONER HENDRIE: Sorry.

CHAIRMAN AHEARNE: Would you like to try again?

MR. DIRCKS: We have had briefing on this before and I think it is worthwhile to go back and tell you where we have been and where we are seeing some difficulties and where we see some possible future attempt to cope with some of this.

As pointed out, Darrell has the ball. He has had it for awhile. He is ready to talk about it.

Surprisingly, this fiscal year is pretty much on schedule with what we set out to do. We are about halfway through the year and we have accomplished 55 percent of the objectives, so it is functioning.

MR. DENTON: We're just coming now to the payoff period of the SEP program. For years we only talked about the plans. These are reports on specific grants. The review has been completed and the decisions reached as to the acceptability of those old plants with regard to that area of review. I think there over 200 issues now which are, in essence, resolved for these old plants and an overview of mine is that, in some areas, we are finding even though these old plants were designed before the general design criteria they are able to share conformance with today's requirements. We managed to work these out in detail so, in that sense, we looked better than perhaps we thought they would.

COMMISSIONER HENDRIE: I could have told you that. There was an ancient principle that we were giants in those days.

(Laughter)

MR. DENTON: At the same time, we have also identified those areas where the Commission's requirements have changed drastically and we know that we have got to focus in on to make a commparability finding. We can identify those areas, so we are trying to maintain the manpower in the program from here on out.

It went through a slump a year or so ago before we assigned people, dedicated reviewers, but we are right now in the pay-off period to define those areas where the staff really needs to concentrate and make those changes and one of the fall-outs from this program, for example, was the show cause order with regard to the effect and with that introduction, let me have Darrell give you a more planned review.

MR. EISENHUT: I will just summarize and go through. I wasn't sure how much detail. There is no long track record. I am not going to attempt to go back through the old track record.

If I may have the first slide?

This is just a simple outline of the areas. We would like to go through a very brief introduction and background so we know where it is.

Going through its present status, we will just mention some of the difficulties that Harold touched upon and some of the things we are looking at to keep it going and really build the momentum into the program to reach the hard decisions.

The programming, you recall, has to do with the overall safety confirmation of older plants. Eleven of the older plants are being reviewed.

These plants, in large part, pre-date a lot of the modern plants that we have today -- modern plants where they have a large loss and accident, large ECS systems, with a somewhat rigorous, very elaborate set of safety requirements.

These plants go back, I believe -- the first plant went into operation in 1959 or 1960, about that time. Seven of these old plants still have provisional operating licenses.

Remember, the thought was that a provisional license was a license that was in effect for something like 18 months. After 18 months the thought was you looked at how the plant performed. If the plant performed well you convert it over to a full-term license. If the experience was not too good, you would look at it from the standpoint that you need to keep it going under a POL.

There was an automatic expansion feature of those POLs -- that is, if the licensee requested --

COMMISSIONER HENDRIE: A timely renewal.

MR. EISENHUT: Yes. A timely renewal.

It is similar to other areas we have that, if the request is submitted, it continues. The basic thrust of the program was to compare these old plants against current -- against current safety standards.

COMMISSIONER HENDRIE: It is not an NRC regulation but it has its foundation -- where? Is it not in the Administrative Procedures Act?

MR. EISENHUT: Yes, it is.

MR. MALSCH: The activities are contingent until the application is acted on.

MR. EISENHUT: The basic thrust of the program was to compare these plants against current safety standards. I use "standards" in the overall, larger sense, not as if you were building the plant today, but look at it against today's real thrust of what the safety requirements are trying to do today.

Today's requirements standards, guides, are the yardstick we are looking at. The review was not to be a five by five, line by line, review but rather, as each review progressed to some point, you either decide that you have found a major deficiency that must be fixed now or you find there could be a deficiency that could wait until the end of the program, re-orchestrate them together, hence an integrated assessment.

CHAIRMAN AHEARNE: Can I ask you a question on that, Darrell?

One of the difficulties in trying to read a lot of background quickly, if sometimes there are some ideas that weave in and out, it is a little difficult to conclude which stayed and which didn't. But at various stages, I found in one case the objective was going to be to look at the design basis events and then on the basis of those, see which systems were critical and then analyze those.

There was another flavor at some point that would decide on some other way, which systems are important to safety and look at those.

Is there any simple way of describing the process that you have just said?

MR. EISENHUT: Yes, there is. The basic program was laid out first. The thought was that it was a systematic program.

We started with something like 800 topics not designed as basic events, per se, but 800 topics. There were a collection of topics from the ACRS, many different elements of the organization, from the public, from utilities. Those were put together in a set of topics.

Where we culled those, we could, either for lesser safety significance because they were the development of new requirements, because they didn't affect those family of plants -- we culled it down to 137 topics. Those 137 topics had two parts to them. One part was there were about 50 items which were already undergoing review by some other people -- fire protection, for example.

Those 50 we just said, the review will continue as it needs to and we will, to the extent possible, integrate it with those other 80 or so but the other 80, what we will do is that we will first look at it topic by topic by topic and you would reach an interim feeling of the measure of goodness of the plant. But you would not act on fixing up a crane in the building, necessarily -- that may not be the world's best example, but you would not fix up one particular component until after you had gone through the topics.

You looked at the design basis events. You looked at really the design basis events and said, well, so what? So what is the net effect, if the containment could only take 40 pounds of pressure instead of 60 pounds. You really look at the design basis events and you look at this. This is sort of the integration aspect and by looking at what is important to the design basis of either the terms of the likelihood of the event or the consequences, given the event occurs.

So it is really both of those two things. We have been going through, topic by topic, first and you have some competing things when you lay out to do that. On firsthand, it is always nice to have as many topics resolved as possible.

If there are 130-some topics on each plant times these other plants, you see there are about 1500 plant topics to be done in the first place.

One thing would lead you to do the other things first, because I can knock off the first thousand. Well, that is good. It shows we have progressed.

But on the other hand, you note the more difficult topics -- that is the topics, for example, associated with seismic design are not going to bear any fruit for two years or so.

MR. DENTON: Let me interrupt for a few minutes, Darrell. We have gone at it topic by topic but I still think there are issues of conceptualization as to what this program is intended to do that remain to be decided. We never could decide quite how to approach it, and we kicked it off.

That is how we got into sweeping through this large number of issues we were going to sort of defer what was clearly down the road. How we have actually combined systems, designed basic accidents, or whatever. And the program has suffered a bit from lack of general consensus on how do you approach plants that were designed and built 20 years ago to all different codes and standards of the day.

Today, I think, you will find at the end of the slide, at the end of the presentation, I should think it should go from here more to the risk assessment idea than we were proposing to do when this first started.

MR. EISENHUT: It is really the two things combined.

If I could have the next slide, this is really -- it is a slide that is two or three years old. This was basically the objectives of the program as laid out by the Commission following the first briefings back in 1977.

The program that you will see, as we will mention when we get towards the end, deviates from this slightly, but these are more the general statements.

The last one is the one that we will be addressing somewhat in a moment.

If I could have the next slide, which just catches up with what I've been saying, this just gives you a status of how it is today. This is basically the topic, the plant topic review list, and how far it has proceeded.

The resolution of certain of these topics has already -- it has been a hybrid up to this point, as it goes.

To the extent possible we have been using risk assessment and we have been trying to use it more as we go into this program.

There are two problems. One is, of course, a shortage of that type of expertise to be doing it on a large scale here. But secondly, you must make the comparison against the requirements in the regulation because some of these plants are going to have an opportunity for hearing, and this review formed the legal basis that is necessary for that hearing record.

CHAIRMAN AHEARNE: Careful about "on this legal basis." We have just been deserted by our lawyers.

MR. SCINTO: Not all of them.

COMMISSIONER HENDRIE: I'm glad you stuck, Joe. The General Counsel left as soon as the subject was broached here.

Let me represent some sort of commentary.

CHAIRMAN AHEARNE: Now the Commissioner lawyer is leaving.

(Laughter)

CHAIRMAN AHEARNE: Are you having any trouble getting information from these licensees?

MR. EISENHUT: Yes. This goes back to the last item of the objectives I mentioned in the first place. The basic thrust was we were not going to go out and lay a big requirement on the licensees and say, answer these 137 topics.

The first thought was the staff was one of the key items, even from the Commission's guidance memos, to go back first and look at all the paperwork that was there existing and that is a real problem in these old plants because you have an initial final safety analysis report with maybe 50 amendments, maybe 300 letters with additional technical information and you must put all of this together and really try to decide what the situation is.

Even then you don't have a lot of technical information. So the first cut is, you put the staff to work going through all of the available information, seeing what is there before you go to the licensee and say this is the additional information that I need.

It is a very difficult job, because of the availability of information. That is one of the key ingredients.

CHAIRMAN AHEARNE: And have the licensees been giving you much trouble in getting this?

MR. EISENHUT: The licensees viewed this -- I think it is fair to say they viewed this as an NRC program. They viewed this because, as an NRC program, back from the inception when it was announced, the staff said this is basically an NRC program. The principal burden initially in the program will be on the NRC, not on the licensees.

It is the fifth objective in that initial slide that has caused considerable difficulty in getting up to speed. They are getting up to speed on selecting topics where we made it very clear in the beginning that if left to the staff, these were some areas that we were going to have real problems with.

It is fair to say in the last few months, over the last perhaps year, the licensees have been instructed to work in certain areas.

MR. DENTON: And all of the lessons learned that have swept through since TMI have all applied to these plants. They have been swept up and making all of the order changes and everything else at the same time while trying to reassess old issues.

MR. EISENHUT: This program we are talking about today is over and above everything we have put on all plants and we didn't spare these plants, so they have got a lot of work.

CHAIRMAN AHEARNE: In the information to go out, is it primarily for them to collect information that they have, or do they have to dig out and give it to you and you do the analysis, or have you shifted over to asking the licensee to do the analysis?

MR. EISENHUT: We sent out guidance letters where we broke all the topics and all the plants into three categories, I believe. I believe it was three categories.

One is where we say we think we have enough informatio that after we look at it, the problem is going to go away. We have those where we thought the licensee would have to do some supporting work and those where the licensee would have to do a considerable amount of effort.

So we wrote a letter to him and said, these are the items. I think we had actually four subgroups, but it was essentially on those lines.

So on some we have shifted it to licensees. You will see one of our bottom line recommendations is that we are looking again to see whether we can shift more to licensees in the program at this point.

These are some topics where the SEP is actually doing some of the frontrunning work.

The environmental qualification we talked about, the eleven SEP plants are the first plants being reviewed and they are the lead for all of the other plants. Safe shutdown reviews -- this is going back and seeing what you really need to shut down a plant.

This is what I mentioned earlier.

The seismic program is probably the single biggest program in SEP because from inception we identified this as perhaps the single biggest problem area.

CHAIRMAN AHEARNE: Is this because the NRC's regulations have developed much more in that area?

MR. EISENHUT: That is because four of the plants really didn't have the sites reviewed. Two of them didn't even refer to a seismic design basis. Two were designed to the unified building code and the other plants were designed to the very beginnings of what was later developed into a highly refined NRC program.

The site specific spectra program here is actually a program that is a state of the art program. It is using some 15 to 20 seismic consultants throughout the country and a panel forum to come together with a new approach that is actually a refinement on Appendix A.

It is a program which is giving us some quite definitive guidelines on plants in the eastern part of the United States. This does not address -- it addressed ten of the eleven plants. It does not address plants west of the Rockies. It is the eastern plants, that are generally in a lower seismic region, generally nothing over a .2 or .21.

In fact, this program will likely have an impact back into the process on new plants, if this methodology is fully adopted and it turns out to be one of the ways. We really try to define the sort of deterministic, imperial approach of Appendix A.

MR. DENTON: As Darrell mentioned through technical assistance, you might name the contractors here.

MR. EISENHUT: Yes. The two principal ones are the TERA Corporation and the Lawrence Livermore Lab. And there is a whole slew of consultants including, of course, New Mark Hall.

MR. CRUTCHFIELD: People like --

MR. EISENHUT: This is quite unique because the program that was laid out by the TERA Corporation and Livermore, because it was going to a number of something like 10 to 20 experts around the country and asking the what you really would think about the seismic design of these older plants. They actually sent this approach and got woven into the overall technical group of experts, even people who were intervenors in some of the hearings.

So they tried to get a consensus of not just those people who have been supporting plants, but those people who were actually experts in the fields of -- who had actually opposition in public hearings.

So it is a very well-founded program.

It is clearly one of the biggest.

MR. DENTON: I would just echo, it looks like a very successful program so far. It involves site visits, actual examination of the way the plants are constructed with a very large group of individuals.

COMMISSIONER BRADFORD: Darrell, you said that two of the plants, I think -- correct me if I misparaphrase it --

but were essentially built without a seismic design -- without seismic design considerations.

MR. EISENHUT: That is right. They were built, in fact, prior to the NRC's having a seismic design requirement. Two others were built in accordance with the NFI building code. Therefore, they were not built to any dynamic analysis approach either.

COMMISSIONER BRADFORD: What are you finding when you look at those four plants now?

MR. EISENHUT: When we look at those plants, up to this point in time the basic approach was to develop a, a methodology; second, develop what kind of acceleration you would expect, whether it is peak ground acceleration; and then, third, the spectral shape that needs to be used in analysis.

For each of these plants we have now, the methodology has been pretty well developed and has been generating a draft acceleration and spectral shape for for each of these plants. We are going to a meeting with each of these licensees next week and that will be the first time we will be laying upon them the results of our work, saying this is the new acceleration in the spectrum that our analyses have seen.

MR. DENTON: There are two things to keep in mind about these. Livermore did these for their own reactor. They used Appendix A, and that reactor was designed early on in the manner in Project A. So they used Appendix A and they went to their most elaborate mechanical engineering codes for plastic deformation and concluded that they only needed to make one change.

A serial test reactor, to meet present day standards, but it was a very elaborate analytical job and the same approach is being applied here. We don't really know the outcome until you do all the calculations and see what changes.

The other thing to keep in mind is they are low power plants. They tend to be located in remote areas so in terms of their consequent side of the risk equation they are at the bottom parts of the comparison.

But we don't know until these results are further along.

COMMISSIONER BRADFORD: When do you expect to have the analysis you have done matched up with what you know is in those four plants?

MR. EISENHUT: So you can make a determination? Probably later on this year.

Let me clarify what I said before. Two of these plants did not have a seismic design input at all. Two others were designed to a static, unified building code, which is a very small acceleration.

The fifth plant, San Onofre, had a static design also. These five plants, I should point out, have been doing considerable work.

San Onofre has spent tens of millions of dollars already redesigning work, trying to upgrade their plant. So a group of these plants -- they recognized this from the very beginning and they are doing considerable work.

We think some plants will have structural modifications. Some will have mechanical equipment modifications and almost every one will have electrical equipment modifications in order to assure that they can resist an earthquake.

COMMISSIONER HENDRIE: In the early days, Peter, where there was not any sort of organized seismic design basis, even if your spec for the design jobs that mention it, the structural people pretty generally would throw in some static horizontal forces, a la the Unified Building Code, which would cover seismic and some wind bloating, amplified wind bloatings, and things like that.

And because of the generally conservative design practice in structures, that turns out often not to be too bad.

I can remember when the first Brookhaven Graphite Reactor was designed in late -- I guess it was 1945 or the beginning of '46. They sought advice and received a letter from a very eminent and ancient Jesuit seismologist up at

Fordham who, among other things in his letter said, "I can assure you on the highest authority that there is very little seismic activity to be concerned about on Long Island." I said, by George, you can't do any better than that.

(Laughter)

COMMISSIONER HENDRIE: Nevertheless, there was a twentieth of a G unified building code put into the structures. But when you get things like equipment qualification, at the present time you have a seismic design basis and you have got a piece of mechanical or electrical equipment that is important to safety.

You have to go put it on a shaker table and shake it with a prescribed spectrum and see if it holds up.

..e was nothing like that contemplated in those days, so the mechanical gear, the electric gear, it was whatever good quality industrial instrumentation and mechanical equipment was designed to in those days. It didn't have that kind -- it certainly didn't have seismic --

COMMISSIONER BRADFORD: In the case of the two that didn't have the seismic factored in, is that like saying they were built, in effect, on the assumption there would be no earthquake?

MR. EISENHUT: No. At the time they were built, which means, if you look at some of these plants, they were designed back in the late 50s. That wasn't one of the considerations.

MR. DENTON: I think it means you can't find that the AEC gave any attention to this and they were probably built at that time. Good practice for hazardous structures, suc as dams, intended to follow that kind of industrial practice.

But the AEC didn't get into the review at all.

The same way with floods.

COMMISSIONER BRADFORD: What is puzzling me, I would have thought that somewhere you would be able to find that they used some kind of acceleration factor regardless of where they got it.

MR. EISENHUT: Two of the plants even predated that. The others were like the .02 G that Dr. Hendrie just mentioned.

COMMISSIONER HENDRIE: I expect that in order to find that, you see in those days there wsa not the requirements for the sort of documentation of what you put into your design that there is now. Now we have requirements. Now you have to keep documents and show that you met all the requirements, and so on.

In those days, the chances are that the project owners, the people who are buying the plant, simply didn't even ask their engineers, what was your basis? They went to an engineer and said, I want a building. And the engineer sat down as a competent professional in the field in the context of the practice at the time. I dare say that the structural people cranked some things in there to provide themselves elbow room. But I doubt very much it is documented.

The only way you would ever know is to go back and find the chief designer of this or that and the other thing and ask him if he could remember what he put in there.

MR. DENTON: The intent of this program was --

COMMISSIONER HENDRIE: That doesn't mean that the steel and the concrete may not be pretty good, but you don't have a paper trail that you can follow along.

COMMISSIONER BRADFORD: The business of verifying that against what the program is coming out with obviously will be quite a challenge.

MR. DENTON: I think this program is going at it the other way. It is really what is there and how they have their supports and piping arranged in estimating and comparing that with what Appendix A required.

MR. EISENHUT: It is really doing both of those. You want to look at the existing plant, the existing concrete and you want to estimate what it will take. But if it can take X amount, you have to look at the regulations and say what would the NRC's present current approach require? Not to state that it would require something much greater.

So it is a very difficult job, particularly in this area and there is quite a large gap.

MR. DENTON: Even the mechanical codes have changed. It used to be SEY, SP3. I think this is why we have such an elaborate array of consulting assistants in this area. If there is anything to do with seismic reviews, even any modern plant is complicated and doing over a plant that wasn't built with that in mind is even more difficult.

MR. EISENHUT: Just one other thing on this program. There are other things coming out. In passing on the seismic program, for example, when the teams have been going to sites and requiring a look at the design of the plant, you get some spin-off effects.

For example, after looking at some plants they found with the DC power supply that the batteries in the plant were sitting on battery racks. In these old plants, they didn't think of bolting down the batteries. They didn't think of bolting down some vital equipment, of putting restraints to hold down. Some pretty simple things that you know are going to have to be done, regardless of how this program comes out.

Those kinds of things. We have issued information notice to all operating plants, not just the SEP plants, but we have said, you ought to look at these things and you ought to consider -- we went to everyone.

These are the kinds of things that have spin-off effects.

CHAIRMAN AHEARNE: You will be sending out some sort of a bulletin which will say, batteries ought to bolted down?

MR. EISENHUT: Things like batteries are, in fact, bolted down so there is a good chance they won't fall off the racks with an earthquake. We felt at one plant there was a good chance that they would.

CHAIRMAN AHEARNE: Let's take that one plant. Is that telling it that it should, or is it saying ---

I'm not clear what you are really telling us.

MR. EISENHUT: The item that went out is saying that you should look. We have asked them to follow up to be sure that they are looking, letting them follow up and do the job.

On plants that we found that there really is a problem we are pretty much telling them, but it is an informal telling them at this juncture, though.

We haven't issued an order, or anything like that. It says, put on bolts and bolt down your equipment. We are trying to wait until we see the overall program, but we sent them a formal letter which said this item we don't think you should wait until the end of the program.

MR. AHEARNE: You have formally told them, for example, that they have to bolt on batteries?

MR. CRUTCHFIELD: We have asked them to survey their facility. They have come back to us in many areas and said, we seem to think we are satisfactory. We have a great number of bolts there. I have notes on the bolts, and things like that. In other areas, they think they are weak.

Right now, what we are doing is assessing what we have received back on all eleven plants. Then we will be going out with instructions as to what action they are to take.

MR. EISENHUT: On the batteries, for example, the first site they were found --

MR. CRUTCHFIELD: I think the batteries have been taken care of.

MR. EISENHUT: So it is an informal exchange. We try to get licensees where every time they find something, say we ought to can it.

CHAIRMAN AHEARNE: There will be --

MR. EISENHUT: There will be a formal way at the end of the program.

CHAIRMAN AHEARNE: You will be sending letters back out to the SEP facilities saying here are the things you have found in that review that ought to be fixed.

MR. EISENHUT: Yes, that is our intent and they will all be on a nice, neat document at the end of the program.

I won't go through the rest of these. I will just mention the last one. That was control room habitability. That was an item that was identified under the SEP program that, coincidentally, came out as one of the action items under the post issues also. This was an issue on the SEP program that was identified even prior to that.

This is just the tightness of the control room.

Can I have the next slide?

P·ogram difficulties. I would just mention a couple of these.

It is a difficult program, as you can imagine. When you find a deviation is actually when your work really begins and you have to really assess those deviations. The designs are different than current plants. That is, some of these eleven plants are really unique. Reviewers quite often are not familiar with these kinds of plants.

So the person who has been doing a lot of review work on this modern vintage plant has a really difficult time going back. It is a learning process for that plant.

I have already mentioned that licensees are not aggresively pursuing the program and Harold mentioned that they had considerable amount of competing activities over the years.

CHAIRMAN AHEARNE: Darrell, you say, as you mentioned before, it is viewed as an NRC program. Are we doing anything to dissuade them of that view?

MR. EISENHUT: On the next slide, I might address that.

CHAIRMAN AHEARNE: Wait. Peter had a question.

COMMISSIONER BRADFORD: You mentioned this problem of reviewers having to accustom themselves to these older plants. In the course of your reorganization and just general turnover, you must have had a fairly high turnover of people involved in the program as well.

MR. EISENHUT: Yes. We had a considerable number of people who previously were assigned to the program who will not be -- but you will see were addressing this third bullet up there.

MR. DENTON: To one extent, Commissioner, that was deliberate. I was concerned that we were building up a group of people who were wiling to say that the plants that met less than today's standards were okay for some technical reason and another group in the same technical discipline who were insisting that today's standards be applied.

We tended to put all of those technical people together in the very specialty branches and I want to have a corporate memory in those branches that we have a plant of varying designs. And I thought by building up two technical groups, one of whom could approve the system one way and another group who could view the system another way, we would eventually lead to major conflicts between those.

So I hope, by putting them together, they will be able to rationalize more fully.

CHAIRMAN AHEARNE: Are you saying, Harold, that you are no longer going to have a separate group of people looking at SEP plants?

MR. DENTON: We have about half the people who are looking at SEP plants are still together. The other half we put back into a technical home where it may not be the same reviewer.

CHAIRMAN AHEARNE: So you are saying, instead of having all the people working on SEP on one group, you are having some of the people still working in SEP in that group and, in addition, you will be pulling people out of these other centers to work on SEP?

MR. DENTON: Yes.

For an example, to pick an example, in the structural seismic area, there is still one person working for Denny in the seismic design area, but he is getting assistance in geology and seismology from that branch, for example. So rather than have a geologist just assigned to SEP plants, we debated back and forth which way to handle that, whether to dedicate people or to go the other way. And I guess we have gone about halfway towards putting everybody back in the technical home.

COMMISSIONER BRADFORD: What is the relationship between the SEP Branch and what I guess is the SPE Branch, the Safety Program Evalation? The latter is developing criteria across the board?

MR. EISENHUT: The Safety Program Evaluation Branch is a branch that is more looking at the overall business. It looks at any new requirement.

For example, it is the built-in process to insure that new requirements, whether they be on an old plant or a new plant, et cetera.

MR. DENTON: I didn't pick up and jump to the other division. We do have a program now in the Division of Safety Technology that I hope will perform the functions at the branch level that the ratchet committee used to perform, that whenever any division initiates a new requirement or thinks up a new way to improve his particular discipline, we go over to that branch under Roger Mattson, and that will be looked at for impacts in other areas, and total risk improvements and Mattson would endorse it.

It is a permanent ratchet committee that interacts with standards and ACRs and then comes back and does it. That is different than the small branch of dedicated professionals who are still working with SEP plants.

COMMISSIONER BRADFORD: Right.

What I am trying to get at, though, is at some point the question has to arise once they have made a decision that something ought to be back here, whether that decision would apply to the SEP plants as well.

Are the SEP plants treated any different with regard to decisions of that type than other, older units.

MR. CRUTCHFIELD: I think, in general, we have tried to historically take the R^3C, category 2 and 3 positions, which were backfitted on a case by case basis and forced back, if you will. We factored them into these eleven facilities.

I would foresee that we continue on that proposal with respect to continuation of the SEP. New postures and positions that come out of this group will them be fed back into the SEP group with applicability to these older facilities.

COMMISSIONER BRADFORD: What then becomes of my reasoning that one doesn't want to impose too man ad hoc changes on the SEP plants as you go along because they are going to have to be sort of major, far-reaching changes.

MR. DENTON: I think that consideration is still there in certain areas and some of these backfitting issues will probably be addressed through bunkered systems where they will be solved in one complete redesign and many of the isolated problems.

If you take one like the show technical advisor, that is an easy one. They can put that one in. So there are a number of plants, I think, that are considering bunkered systems that will have to design a whole new tray of safety systems to encompass all the new requirements.

MR. EISENHUT: So far, we have laid on all these new operating plants, including these eleven, all TMI issues. We haven't given SEP relief on any of them.

One of the things they would like to request is rather than put in some of those post-TMI fixes, they would like to consider looking at the SEP, looking at an integrated, brisk assessment, and then deciding on what needs to be fixed in their plant and maybe going to something like a dedicated shutdown system where rather than fix up systems A, B, C, D, F they would give us a brand new one and add on one brand new system with its own source of water, its own power supplies, capable to do the job which could help out all of the systems.

MR. DENTON: I think the answer is, we have not bent the system. We have backfitted some things that they would have preferred could have been dealt with in a larger context and some things we have agreed in the larger context. It is almost case by case specific.

If you look at each plant, there is a different ensemble of issues to be solved. There is one plant that is proposing -- and maybe you should turn to that next slide -- to do an integrated risk assessment.

CHAIRMAN AHEARNE: What happened to NRR manpower?

MR. DENTON: We have problems with that one.

CHAIRMAN AHEARNE: Back one slide, please.

MR. EISENHUT: Budget assumptions are 32 man years of effort devoted to the SEP program. That has been effect since 1978, FY 1978. In FY '78 there was considerably less than 32. It started picking up a little bit in FY '79, as you can see on the slide.

FY 80, there is -- it looks like it dropped off in FY 80. But there is a mistake in the computation on FY 80 because 13 did not include an overhead factor and really what it is, it is in fact the information we have for FY 80 is that it is right on the money. We are expending it at almost precisely the rate at which it should be.

In fact --

CHAIRMAN AHEARNE: Are you saying 16 for the first half?

MR. EISENHUT: It would be equivalent to 16 within a fraction. That is reflected because, as Harold mentioned, about last July was when we made the decision to, in fact, take the individuals and dedicate them to the program.

MR. DENTON: That is when we dedicated the resolved safety issues. Since that time it has been getting about the right manpower.

CHAIRMAN AHEARNE: The logical next question is, you are saying when you concentrate all in one place on getting the right manpower, but your decision is not to put it all in one place.

Continue.

MR. DENTON: The manpower now is assigned to the SEP but it is not all reporting to the same branch chief.

CHAIRMAN AHEARNE: I think what Darrell has just pointed out is that when they are all in the same branch it is clear that is what they will work on. When they go to other branches, which is now part of their job, apparently -- what are you planning in FY 81?

MR. DENTON: In FY 81 I think it is the same level of effort. The original effort was to complete this.

MR. EISENHUT: It is essentially the same. I think the real answer --

MR. DENTON: It was to continue the same level of effort until we complete all of these same eleven.

CHAIRMAN AHEARNE: How many are in this branch?

MR. CRUTCHFIELD: The SEP branch has ten professionals, two section leaders and a branch chief.

MR. EISENHUT: So it is essentially 13 out of 32. There is a standard conversion factor of 1.4.

MR. DENTON: If we really wanted to do it the other way, then we would take these people who are assigned here and put them all under Denny and have a 32-person branch.

CHAIRMAN AHEARNE: I was just trying to make sure I understood it.

MR. EISENHUT: The real difference here is we tried it in FY 78 to get it the way we were proposing, but there is a difference. In FY 78, we said that we would have people assigned and we gave the branch chief some flexibility on who that assigning could be.

The thing we didn't do was, we didn't move up in the division's organization and hold the division management accountable.

For example, we are going to have a pretty firm tracking system to see that the manpower is coming out of the system if it doesn't get out of these other divisions, the division's management and the accountant.

MR. DENTON: What I really hope will happen, if you take degrees like mechanical engineering, I would hope that branch chief would realize he is responsible for operating amendments, day to day fire drills, SEP-resolved safety issues.

We have given him resources to do all of these tasks that we have said we are going to do. And he has to juggle -- maybe decide who is the right person to do which task. But his net line-up each month will be to put that much effort into each one. So he should be a little more efficient than if we had dedicated it out and had no flexibility.

But obviously we have got to watch each branch to be sure that it doesn't all get gobbled up and tomorrow is a fire drill exercise.

And we have put in place a reporting system that should do that.

CHAIRMAN AHEARNE: That reporting system is one --

MR. DENTON: We are starting but like we did the SEP program. The caseload is to look at each two week period to see if we are actually getting that much work on operating reactors out of each branch as we budgeted and we are going to do the same thing for our unresolved safety issues, SEP and so forth.

We get all the data in these manpower reporting systems. It is just a matter of breaking it out now in the right order.

MR. DIRCKS: You might pass that around.

MR. DENTON: That is aggregated data. You need to check it branch by branch.

CHAIRMAN AHEARNE: All right.

Now, since you had also mentioned that you have a fairly sizable contract effort in this, how is your money breaking out?

MR. EISENHUT: Basically it is about $1 million. it is going to continue to be administered out of the systematic evaluation program branch itself where there are ten professionals.

CHAIRMAN AHEARNE: Is the '81 money also being resolved?

MR. CRUTCHFIELD: It is in three similar ones. It is a bit rough.

CHAIRMAN AHEARNE: So that is decreasing the level of effort?

All right.

MR. EISENHUT: If I could go back to the last slide, just to wrap it up, we state here that we consider this to be a high priority program. We are shooting for completion in April of '82. We are, as we mentioned, considering having committed full-time reviewers, these other bodies that are setting the branches wherever they are setting. We need to know who they are. They would be committed with their management and their counsel.

We will be looking at plants as we go down the line. We will have 80 or so draft safety assessments. You will have to integrate that together.

There will be two things that are integrated as project power manager, although we don't have him on board today. That is one thing we will be recruiting for, filling some positions.

MR. DENTON: I think we will find it necessary that once we get a good number for each plant to have a person who is full-time then trying to integrate the places where that doesn't perform without having an individual discipline do it, because if they do it it would violate one of our original charters to try to do it all at one time.

The project manager assigned in the old plant, for example, in any of the eleven, has a full-time job anyway dealing with the ongoing amendments and the ongoing action plan items and so forth.

So I see the need, as we make headway on the particular SEP plant, to assign one project manager with a full-time job to take these inputs as we get them to show areas and continue with that plant until he has documented th entire plant.

So that would be like eventually eleven more people that we have budgeted for during that time phase when it has a high pay-off.

MR. EISENHUT: We are also considering different alternatives to the program. That is putting more burden back on licensees very specifically, in specific areas, not just a broad brush program -- especially where we are getting it down to the point where it is becoming more finetuned in the major problem areas.

CHAIRMAN AHEARNE: Is that in any way responsive to the ACRS or is it more gee, I believe it is now time to do that?

MR. EISENHUT: Even before we had the ACRS letter we were thinking of doing that, over the last year. You are very familiar with other problems. In 1979, the licensees were extremely busy with seismic matters and then there was the wave of post-TMI matters.

MR. DENTON: I think it is really the change in the perception of burden of proof. Before TMI, it was kind of on the staff to prove that there were some defects in the design.

MR. EISENNUT: It has also proceeded far enough along to where you are able to do that. Now we don't just send out and say, review these 1500 topics.

I think we would be able to point them in the direction we want to point them to. These are the things where the biggest safety pay-off is, and I think that is the difference.

COMMISSIONER GILINSKY: When this program is completed, will these plants then be roughly on a par, at least in terms of documentation, with the other plants in our system? Where will that put them?

In other words, after that point, will we be able to deal with all the plants uniformly? Or will we still have to --

MR. EISENHUT: There will be still some in the middle. Remember, when we started this, we thought we need to get these eleven up to the par where they are either on the par or there is a documented record. Either they meet a requirement or they don't meet it, and if they don't meet it, here is why, so you don't continually go through question after question after question concerning the safety adequacy of all plants.

When these eleven plants are done, plant number 15, of course, still has some questions about it and plant number 20 because it wasn't a stepwise process. It was of an evolving nature.

COMMISSIONER GILINSKY: Where do you see us going after this?

MR. EISENHUT: I think what we will have to do, we will have to look at -- this is called Phase II. We will have to look at it and assess where it is.

Personally I can't see going through, even though there are a lot of merits to a systematic evaluation program, perhaps the POL to FTO record that was needed helped drive it. I really can't see going through 137 issues on all of the rest of the 70 operating plants because I think the safety play-off, the real physical improvement in plants, just isn't worth it.

We may have a lot of difficulty with people asking questions, but I think we will just have to figure a way around that, that if you go through these eleven plants on some of the topics and you find that what is there is adequate, it is likely that the rest of the operating plants are also adequate in that area.

MR. DENTON: We will postpone a decision on where to go from here until the results become clear and it might be in some areas we would decide that if they were adequately addressed, unknowing to the AEC in the earliest plants, they probably were addressed that way from there on.

But if we find areas where they were not, we will have to keep plugging away on the later plants until we find where the trend changed.

But I think in general there is a lot of sentiment today for a national reliability, a national risk assessment approach where eventually we would have to be able to specify the type of risk assessments that would be valid and useful results, and really focus the results plant by plant in order of the highest pay-off areas for improvements.

CHAIRMAN AHEARNE: Following that, will you mesh with the other program?

MR. DENTON: The IREP program was intended to disclose how best to approach the entire population.

CHAIRMAN AHEARNE: But it still eats its way through plants.

MR. DENTON: The original IREP program was going to be six plants. Hopefully that will teach us what to ask for for all plants.

CHAIRMAN AHEARNE: But for example, would you see in some of the states, returning this approach back to the SEP plants?

MR. DENTON: Yes. I hope some day we would be able to do something in the risk assessment line on all plants.

MR. EISENHUT: They sort of cross cut two different directions.

COMMISSIONER GILINSKY: Let me go back to a remark you made earlier, that if these plants are basically okay or don't depart too far in safety terms, we can expect that probably the other plants are okay, too.

MR. DENTON: I wouldn't want to stretch that too far. It depends on, I guess, the design and the vendor and the ASME.

COMMISSIONER GILINSKY: I understand that, but just as a very general proposition you were making the point earlier that, in terms of possible consequences, these are the low end of the scale. They are small plants, if nothing else.

Now, as you go up in the CP number, the plants both are more increasingly conformed to current standards but they also get bigger.

The question is, is their conformance to current standards, say, getting bigger?

MR. DENTON: If we have done our job properly --

MR. EISENHUT: They should at least offset. We want to keep a uniform approach.

COMMISSIONER GILINSKY: Is it immediately obvious?

MR. EISENHUT: No. I don't think you could go so far as to say that it ought to be immediately obvious, because it is a very complicated process.

MR. DENTON: The other thing you have to look at with these old plants, too, are the operating history. That is a factor that really wasn't revved in strongly in the original part of the SEP program.

You have to ask them that. Be sure that they look at it. Not just the design per se, but we have 20 years of history on some of these plants.

MR. EISENHUT: One of the principles early in the SEP -- this is in fact the way probability got there in the first place, even though you migth not be able to show that something is very reliable. You have 20 years of data in that particular plant.

In fact, that has been factored in to a number of the items.

CHAIRMAN AHEARNE: Since a lot of those plants go back many of those years before the AEC or the NRC were asking for live data to be supplied, don't you have to get a lot of that information out of licensee records?

MR. EISENHUT: You have to get it from the licensees, yes.

CHAIRMAN AHEARNE: Have we done that?

MR. EISENHUT: Yes.

Where we have the question we ask the licensee -- we made it very clear in our opening letters. That is one vehicle operating experience of the facility, so utilities can go back and they have those vehicles. It is optional to them.

CHAIRMAN AHEARNE: But we haven't actually asked?

MR. EISENHUT: We have, in some areas.

MR. DENTON: Pre-TMI there was the idea not to burden them with this. There was this exploratory on our part.

I would like to ask them if they are going to do that for us.

CHAIRMAN AHEARNE: Our point was we are obviously agreeing that you have an old plant. It is going to have a lot of data available. I doubted that it was -- it wasn't sent to us. You'd have to ask them for it, because they may not even have kept it.

MR. EISENHUT: Most design information is not sent to us either. That is why when someone starts working in one of these plants, it takes the first six months to basically get acclimated with the plant and get aware of the information.

MR. DENTON: That is why I like the idea of a prescient manager on this plant, to do this integration so you are not just looking at technical isolation, bits and pieces of the entire plant, but someone who can say, considering all of this together, and what I know about the design, the operating industry, the site, where does this whole plant stand and what needs to be changed should be changed first.

I don't think we will get it if we just have one project manager per plant, because he is really burdened down with ongoing activities.

MR. EISENHUT: That concludes our presentation.

CHAIRMAN AHEARNE: Joe, do you have anything?

Peter?

COMMISSIONER BRADFORD: One of the things that concerned me is about the program, as we have been wrestling with the fire protection and environmental question separately, was the way in which it seemed that the SEP plants for other reasons than one might have thought would be the areas of greatest concern, have turned out to be the plants which were hardest to bring into compliance.

The point was made that they had been told that these would not be applied to them until the end of the line.

Is that a problem in other areas as well?

MR. EISENHUT: Let me make sure that I clarified that.

We didn't tell the licensees they did not have to fix fire protection into the program.

COMMISSIONER BRADFORD: Go ahead.

MR. EISENHUT: We had 70 operating plants that needed to have a fire protection review. Rather than do the eleven SEP plants first of the 70, we made the last 70 but still part of the program only because we had laying next to it an SEP schedule where we wanted to get the maximum benefit of the fire protection reviews and the SEP seismic reviews, safe shutdown reviews, and call for them to come together at the same point in time.

So if you had brought first the fire protection reviews first, we would want to go ahead and fix those plants. We wanted to have the two converge together.

We did not give the SEP plants relief on fire protection matters in any other way.

COMMISSIONER BRADFORD: I understand. I am not even sayng that was an irrational way to go about it.

The business of trying to get all the problems fixed up in a halfway coordinated manner, but the concern that one comes across there does leave it in the oldest plants we have we have some of the longest running deadlines as far as coming into compliance.

And I just wondered whether they were -- I suppose seismic is another area.

MR. EISENHUT: The only item that I can remember which we actually put last in the program was in fact fire protection because we thought -- and in fact, there is a benefit there that you end up with a better fire benefit program than in the past because, just based on fire protection, none of these plans, we would expect, would be required to have a dedicated shutdown system. Just fire protection would not drive them there.

We faced ourselves the question, does the situation look bad enough with respect to fire protection required, and the answer is no. But when you take that in connection with a lot of other considerations, the answer may well be that you are looking for something better in the long term.

So we really didn't forego -- if you find the major problem, if you remember back from the objectives, one of them was that you had to have, built into the program, a system that if you find a major design deficiency or a major problem, you go ahead and fix it. Environmental qualifications is a good example.

The utilities, all eleven, all argued very strenuously that they thought this was contrary to the SEP philosophy. Our answer was we think it is important enough to be contrary to the SEP philosophy.

The LaCross liquefaction was another, so there were a number of them where we decided to put the fix in place before the completion of the SEP program.

COMMISSIONER BRADFORD: What is the role of the resident inspectors in the SEP scheme of things?

MR. CRUTCHFIELD: We have been utilizing them to help us in utilizing capability of the licensee -- how good he is performing and using him to help us locate where information is. We may be overlooking that, so we can keep in touch with him through the project management side as to what is going on at the facility that could impact the SEP efforts.

CHAIRMAN AHEARNE: Any more questions?

Joe, Bill, do you have anything else?

MR. DIRCKS: We could have another crack at this program --

COMMISSIONER BRADFORD: I have another question.

In terms of a sense of priorities, where does this fit in in the current NRR. If you had $5 to allocate $1 apiece in five areas, would the SEP program be --

MR. DENTON: It is both casework and OL and CPs so it is up there with operating amendments and unresolved safety issues.

MR. DIRCKS: It wasn't touched during the scouring of the resources for the action plan financing which is something, because resources is almost everything.

So, to that extent, it was held apart and given that priority that we wouldn't even touch it.

CHAIRMAN AHEARNE: I guess in running back over some of these whole things I have found the notes I have made from August 3rd of 1978 which clearly predates it, that -- and at that stage, there are -- and what you have said was that SEP was identified as second in priority for NRR.

MR. EISENHUT: That is right.

CHAIRMAN AHEARNE: The first was other problems.

So I guess my only concern would be that as one of the problems with each of the shifts of the organizational type structure I am sure carries along with it a good rationale of why that is the right thing to do.

There was a notice in here in July of last year which had a different organizational structure with a good reason why it was the right thing to do. Now there is a good reason why this is the right thing to do, and I am not taking any disagreement with that.

But, of course, one of the problems with the constant shifting organization is that people are trying to run the program.

They have difficulty keeping track of what it is they are trying to run.

As you have pointed out, these are the eleven oldest plants and are the most difficult to review, but they are obviously ones that the Commission in the past, and you in the past, have indicated they are ones that must be done with very high priority.

I hope that in another six months, or in a year, you will actually reach that conclusion of the effort rather than another set of changes.

I recognize that it is very difficult.

Thank you for the information.

(Whereupon, at 3:45 the meeting was adjourned.)

This is to certify that the attached proceedings before the

NUCLEAR REGULATORY COMMISSION

in the matter of: Discussion and Vote on Briefing System and Evaluation Program

Date of Proceeding: May 6, 1980

Docket Number:

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Marilyn Shockey

Official Reporter (Typed)

Marilyn Shockey

Official Reporter (Signature)

KEY TO TOPIC STATUS LISTING

G - GENERIC

D - DELETED

N - NOT STARTED

S - STARTED BUT NO WRITTEN OUTPUT

Q - QUESTIONS DEVELOPED

I - WRITING OF SAFETY ASSESSMENT STARTED

IC - DRAFT SAFETY ASSESSMENT COMPLETE

R - REVISED SAFETY ASSESSMENT UNDERWAY

C - TOPIC COMPLETE-SAFETY ASSESSMENT COMPLETED

TOPIC NUMBER	SEP PLANT											COMMENTS
	PALISADES	GINNA	Q. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	
II-1.A	C	S	N	N	Q	C	N	S	N	Q	IC	FK
II-1.B	C	S	N	N	Q	C	N	S	N	Q	IC	[illegible]
II-1.C	C	S	N	N	S	I	N	S	N	S	IC	[illegible]
II-2.A	IC	IC*	IC	IC	IC*	IC	IC*	IC*	IC*	IC*	IC*	TQ * MORE DETAILED ANALYSIS REQUIRED
II-2.B	N	R	N	R	N	R	C	R	N	N	N	TQ
II-2.C	IC	R	N	N	N	R	N	C	N	N	N	TQ
II-2.D	N	N	N	N	N	C	C	N	N	N	N	[illegible]
II-3.A	I	IC	N	N	I	IC	IC	S	N	I	I	SS
II-3.B	I	IC	N	N	I	IC	IC*	S	N	I	I	SS * ONLY BIG FLOOD
II-3.C	I	IC	N	N	I	IC	I	S	N	I	I	[illegible]
II-4.A	S	S	S	S	S	S	S	S	S	S	S	HL
II-4.B	S	S	S	S	S	S	S	S	S	S	S	HL
II-4.C	S	S	S	S	S	S	S	S	S	S	S	HL
II-4.D	S	N	N	N	N	N	S	S	S	N	S	HL
II-4.E	D	D	D	D	N	D	S	D	N	N	D	HL

TOPIC NUMBER	SEP PLANT											COMMENTS
	PALISADES	GINNA	Q. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	
II-4.F	S	N	N	N	N	N	S	S	S	N	S	H.L.
III-1	S	S	S	S	I	S	S	S	S	S	S	JK
III-2	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	T.C/H. LEE
III-3.A	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	T.C/H. LEE
III-3.B	*	*	*	*	D	D	*	D	*	D	*	H. LEE T.C & done as a result of II-3.B
III-3.C	S	N	N	N	N	N	S	N	S	N	S	H.L
III-4.A	IC	S	N	N	S	S	N	S	N	S	S	KC
III-4.B	IC	IC	S	N	I	IC	IC	I	N	I	S	KC
III-4.C	N	S	S	S	N	N	S	N	S	N	N	T.C/PD-B to be done with III-5.A & III-5.B
III-4.D	I	S	N	N	S	S	N	S	N	S	IC	done with II-1.C FK
III-5.A	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	KS/DB Review suspended for the outcome pipe break study
III-5.B	S	IC	IC	S	IC	S	S	S	S	S	S	AF
III-6	S	I	S	S	I	S	S	S	S	S	S	T.C /H. LEE

TOPIC NUMBER	SEP PLANT PALISADES	GINNA	Q. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	COMMENTS
III-7.A	G	G	D	D	D	D	D	D	D	D	D	H. LEE 10/25/79 CRUTCHFIELD TO VOLLMER
III-7.B	S	S	S	S	S	S	S	S	S	S	S	T.C./H. LEE
III-7.C	S	S	D	D	D	D	D	D	D	D	D	H. LEE
III-7.D	S	I	S	S	Q	Q	Q	I	S	Q	S	T.C./H. LEE
III-8.A	G	G	G	G	G	G	G	G	G	G	G	KJ DELETED PER MEMO FROM CRUTCHFIELD TO [illegible]
III-8.B	S	S	S	S	S	S	S	S	S	S	S	KJ
III-8.C	IC	IC	IC	IC	IC	IC	IC	IC	IC	IC	IC	IL
III-8.D	G	G	G	G	G	G	G	G	G	G	G	KJ
III-9	G	G	G	G	G	G	G	G	G	G	G	KS 5/4/79 MEMO DAVIS TO VOLLMER
III-10.A	S	I	S	S	S	I	S	S	S	S	S	JK
III-10.B	IC	IC	D	D	D	IC	D	IC	D	D	D	IL
III-10.C	D	D	D	C	C	D	D	D	D	D	D	PDB
III-11	S	I	I	S	IC	S	S	S	S	S	S	KJ/PEC
III-12	S	S	S	S	S	S	S	S	S	S	S	?

TOPIC NUMBER	PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	COMMENTS
	SEP PLANT											
IV-1.A	C	C	C	C	C	C	C	R	C	C	C	PDB need to redo Hel.
IV-2	N	N	N	N	N	N	N	N	N	N	N	PDB
IV-3	D	D	D	G	G	D	D	D	D	D	D	PDB
V-1	G	G	G	G	G	G	G	G	G	G	G	RS. 3/7/79 MEMO DAVIS TO ZIEMANN
V-2	D	D	D	D	D	D	D	D	D	D	D	TO 3/7/79 MEMO DAVIS TO ZIEMANN
V-3	G	G	G/D	G/D	G	G	G	G	G/D	G/D	G/D	JK/PDB [illegible]
V-4	G	G	G	G	G	G	G	G	G	G	G	TQ 4/4/79 MEMO DAVIS TO ZIEMANN
V-5	IC	IC	IC	I.C	I.C	IC	I.C	I.C	IC	IC	IC	PDB
V-6	C	C	C	C	C	C	C	C	C	G	G	RS
V-7	G	G	D	D	D	G	G	G	D	D	D	FL
V-8	G	G	D	D	D	G	G	G	D	D	D	RS
V-9	D	D	D	D	D	D	D	D	D	D	D	TQ
V-10.A	C	C	C	C	C	R	C	C	IC	C	C	RS
V-10.B (. sect.)	N	N	N	I	N	N	N	N	N	N	N	JK

	SEP PLANT											
TOPIC NUMBER	PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	3.R.P.	COMMENTS
V-10.B (syst.)	R	R	R	R	R	R	R	R	R	R	R	MF
V-11.A	I/S	I/S	I/S	I/S	I/S	I/S	I/S	I/S	I/S	I/S	I/S	JK/PDP 1st elect 2nd syst
V-11.B (elect.)	I	N	N	N	I	N	N	N	N	N	N	JK
V-11.B (syst.)	R	R	R	R	R	R	R	R	R	R	R	MF
V-12.A	D	D	IC	IC	IC	D	D	D	IC	IC	C	I.I.
V-13	G	G	G	G	G	G	G	G	G	G	G	MF
VI-1	Q*	Q*	Q	Q	Q*	Q*	Q*	Q	Q*	Q*	Q*	T.Q * LICENSEE RESPONSE REC'D
VI-2.A	D	D	G*	G*	G*	D	D	D	D	D	D	S.B/TC * MARK I PROGRAM
VI-2.B	(A-2) G	(A-2) G	G	G	G	(A-2) G	(A-2) G	(A-2) G	G	G	G	SB
VI-2.C	D	D	D	D	D	D	D	D	D	D	D	SB
VI-2.D	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	SB
VI-3	I	I	I	Q	Q	I	Q	Q	Q	Q	Q	SB/CT
VI-4	S	S	N	N	N	N	N	N	N	N	N	SD
(Purge and vent VI-4 issue)	N	N	N	N	N	N	I	I	IC	N	Q	SB/JL

TOPIC NUMBER	SEP PLANT											COMMENTS
	PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	
VI-5	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	SB WILL BE HANDLED BY RULE-MAKING
VI-6	G	G	G	G	G	G	G	G	G	G	G	SB
VI-7.A.1	D	C	D	D	D	C	C	C	D	D	D	HF.
VI-7.A.2	D	G	D	D	D	D	D	D	D	D	D	PL. 3
VI-7.A.3	I	IC	S	S	I	N	N	N	N	N	N	JK
VI-7.A.4	D	D	G	G	G	D	D	D	D	G	G	PD, B
VI-7.B	S/I*	N/I*	D	D	D/I*	N	N	N	D	D	N	EMK/MF * 1st SYS 2nd LLL
VI-7.C	N	N	N	N	N	N	N	N	N	N	N	EMK/MF CT
(EI&C rereview) VI-7.C.1	S	S	N	N	N	N	N	N	N	N	N	JK
(indep. of onsite power) VI-7.C.1	I	I	I	I	I	I	I	I	I	S	I	JK
VI-7.C.2	N	N	N	N	N	N	N	N	N	N	N	EMK/MF/CT.
VI-7.C.3	D	D	D	D	D	D	I	N	D	D	D	JK
VI-7.D	G	G	G	G	G	G	G	G	G	G	G	TQ
VI-7.E	G	G	D	D	D	G	G	G	D	G	G	PAL. 3/LOOLL BRP closed [illegible]
VI-7.F	I	I	D	D	D	D	I	D	D	D	D	JK

TOPIC NUMBER	SEP PLANT											COMMENTS
	PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	
VI-8	IC	I	N	N	S	S	N	I	N	S	S	KC
VI-9.A	D	D	D	D	D	D	D	D	D	D	D	evaluate TQ seal need in XV-10
VI-10.A	I	IC	S	S	I	N	N	N	N	N	N	JK
VI-10.B	D	D	D	D	I	D	D	D	D	D	D	JK
VII-1.A	I	I	S	S	I	N	N	N	N	N	N	JK
VII-1.B	G	G	G	G	G	G	G	G	G	G	G	JK
VII-2	I	I	S	S	I	S	N	N	N	N	N	JK
VII-3 (syst.)												
VII-3 (elect.)	S	N	N	N	I	N	N	N	N	N	N	JK
VII-4	G	G	G	G	G	G	G	G	G	G	G	MF
VII-5	G	G	G	G	G	G	G	G	G	G	G	JK
VII-6	G	G	G	G	G	G	G	G	G	G	G	JK
VII-7	D	D	D	D	D	D	D	D	D	D	D	JK

	SEP PLANT											
TOPIC NUMBER	PALISADES	GINNA	Q. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	COMMENTS
VIII-1.A	G	G	G	G	G	G	G	G	G	G	G	JK
VIII-2	I/N	I/N	I/N	I/N	I/N	I/N	I/N	I/N	I/N	I/N	I/N	H.F. I - ELECT. N - SYSTEMS
VIII-3.A	IC	IC	IC	IC	IC	IC	IC	IC	IC	IC	IC	JK
VIII-3.B	I	I	I	I	I	I	I	I	I	I	I	JK
VIII-4	IC	IC	IC	IC	IC	IC	IC	IC	IC	IC	IC	JK
IX-1	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	Q	H.F.
IX-2	G	G	G	G	G	G	G	G	G	G	G	H.F.
IX-3	I	I	IC	I	I	I	N	N	N	N	N	H.F.
IX-4	S	S	D	D	D	S	S	S	D	D	D	ADD this topic should be completed [illegible] generically
IX-5	N	N	N	N	N	N	N	N	N	N	N	H.F.
IX-6	G	G	G	G	G	G	G	G	G	G	G	H.F.
X			D	D	D				D	D	D	

TOPIC NUMBER	SEP PLANT											COMMENTS
	PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	
XI-1	G	G	G	G	G	G	G	G	G	G	G	[illegible]
XI-2	G	G	G	G	G	G	G	G	G	G	G	[illegible]
XIII-1	G	G	G	G	G	G	G	G	G	G	G	T.W
XIII-2	G	G	G	G	G	G	G	G	G	G	G	[illegible]
XV-1	Q	I	I	I	I	N	S	N	N	N	N	E M.K/CT/MF
XV-2	Q	Q	D	D	D	N	S	N	D	D	D	E M.K/CT/MF
XV-3	I	Q	I	I	I	N	S	N	N	N	N	E M.K/CT/MF
XV-4	I	Q	I	I	I	N	S	N	N	N	N	E M.K/CT/MF
XV-5	I	Q	I	I	I	N	S	N	N	N	N	E M.K/CT/MF
XV-6	Q	S	D	D	D	N	S	N	D	D	D	E M.K/CT/MF
XV-7	Q	Q	I	I	I	N	S	N	N	N	N	E M.K/CT/MF
XV-8	S	Q	S	N	N	N	N	N	N	N	N	[illegible]
XV-9	I	Q	I	I	I	N	N	N	N	N	N	E M.K/CT/MF

TOPIC NUMBER	SEP PLANT											COMMENTS
	PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	
XV-10	I	Q	D	D	D	N	N	N	D	D	D	E McK/CT/MF
XV-11	N	N	N	N	N	N	N	N	N	N	N	SEPM'S
XV-12 (doses)	C	I	D	D	D	C	N	S	D	D	D	TQ
XV-12 (syst.)	I	Q	D	D	D	N	N	N	D	D	D	E McK/CT/MF
XV-13 (doses)	D	D	N	N	S	D	D	D	N	N	N	TQ
XV-13 (syst.)	D	D	S	S	S	D	D	D	N	N	N	E McK/MF
XV-14	I	Q	I	I	I	N	N	N	N	N	N	E McK/CT/MF
XV-15	S	Q	S	S	S	N	N	N	N	N	N	E McK/CT/MF
XV-16	I	Q	N	N	Q	Q	N	Q	N	N	N	TQ
XV-17 (doses)	C	S	D	D	D	C	N	S	D	D	D	TQ
XV-17 (syst.)	S	N	D	D	D	N	N	N	D	D	D	E McK/CT/MF
XV-18	C	S	N	N	S	C	N	S	N	N	N	TQ done with VI-9.A
XV-19 (doses)	C	S	N	N	S	C	N	S	N	N	N	TQ
XV-19 (syst.)	I	S	I	I	I	N	S	N	N	N	N	E McK/CT/MF
XV-20	IC	S	N	N	S	C	IC	I	N	IC	N	TQ

TOPIC NUMBER	SEP PLANT: PALISADES	GINNA	O. C.	MILLSTONE	D-2	S. O.	Y. ROWE	H. N.	LACBWR	D-1	B.R.P.	COMMENTS
XV-21	G	G	G	G	G	G	G	G	G	G	G	JR
XV-22	G	G	G	G	G	G	G	G	G	G	G	JR/MF
XV-23	G	G	D	D	D	G	G	G	D	D	D	JR/MF
XV-24	G	G	G	G	G	G	G	G	G	G	G	JK.
XVI	N	C	C	C	N	C	N	N	N	N	N	
XVII	G	G	G	G	G	G	G	G	G	G	G	AIB

DEFINITION

TOPIC: II-1.A Exclusion Area Authority and Control

1. Definition:

   The establishment of the exclusion area and the licensee's control over it are reviewed at the CP/OL stage. Thereafter the licensees are required to report any changes with safety implications. The concern exists, however, that (1) the original review may not have been as thorough as currently done, or (2) changes may have occurred but have not been reported and reviewed. In particular, new activities within the exclusion area (e.g., new recreational facilities or offshore oil drilling) and topographical changes (e.g., changes in water levels) may need to be reviewed.

2. Safety Objective:

   To assure that appropriate exclusion area authority and control is maintained by the licensee.

3. Status:

   Selective reviews have been performed (SONGS 1) or are underway (Fort Calhoun) where changes in exclusion area boundary have become necessary.

4. References:

   1. 10 CFR Part 100
   2. SRP 2.1.2

DEFINITION

TOPIC: II-1.B Population Distribution

1. Definition:

Population distribution in the vicinity of operating plants may have changed since the initial review was performed at the CP stage. Special attention should be given to new housing and commercial, military, or institutional installations established since the initial population distribution review.

2. Safety Objective:

New population distributions may require revision of LPZ and population center to [illegible] appropriate protection for the public by complying with the guide[illegible] CFR Part 100. Adjustments may have to be made in emergency p[illegible] accident analyses may have to be performed to determine consequence conformance with 10 CFR Part 100 at new LPZ distances. Potential need for additional ESF (e.g. chemical sprays or better filters).

3. Status:

Has been done on a selective basis only--i.e., Pilgrim 1 new population center.

4. References:

1. 10 CFR Part 100
2. SRP 2.1.3

DEFINITION

TOPIC: II-1.C Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities

1. Definition:

For operating plants there are three concerns:

(1) New hazards created since the facility was licensed,
(2) Hazards considered for licensing but that have expanded beyond projections or which were not reviewed against current criteria, and
(3) Hazards that were not analyzed at the licensing stage because of lack of regulatory criteria at the time.

Nearby transportation, institutional, industrial and military facilities may be threats to safe plant operation due to:

(1) Control room infiltration of toxic gases,
(2) On-site fires triggered by transport of combustible chemicals from offsite releases,
(3) Shock waves due to detonation of stored or transported explosives and military ordinance firing, and
(4) On-site aircraft impact.

2. Safety Objective:

To assure that the control room is habitable at all times and that the postulated hazards will not result in releases in excess of the Part 100 guidelines by disabling systems required for safe plant shutdown.

3. Status:

Action has been taken on a selective basis only, e.g., curbing of military air activity in the vicinity of the Big Rock Point Plant. LNG hazards at Calvert Cliffs under review. The review of older plants did not consider off-site hazards in detail (e.g., aircraft traffic in the vicinity).

4. References:

1. SRP 2.2.1, 2.2.2

DEFINITION

TOPIC: II-2.A Severe Weather Phenomena

1. Definition:

   Safety-related structures, systems, and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include tornadoes, snow and ice loads, extreme maximum and minimum temperatures, lightning, combinations of meteorology and air quality conditions contributing to high corrosion rates, and effects of sand and dust storms.

2. Safety Objective:

   To assure that the designs of safety-related structures, systems, and components reflect consideration of appropriate extreme meteorological conditions and severe weather phenomena. This effort would identify deficiencies in designs and/or operation that may contribute to accidental releases of radioactivity to the atmosphere resulting in doses to the public in excess of 10 CFR Part 100 or Part 20 guidelines (as appropriate to the design of the component or system).

3. Status:

   Generic studies have been initiated to develop guidelines for extreme temperatures and lightning, and to review the current Branch Positions on snow loads. Estimated completion dates are 6/1/78 or later.

4. References:

   1. 10 CFR Part 100
   2. R. G. 1.76
   3. SRP Section 2.3]
   4. I&E Circular "Freeze Protection for Safety-Related Instrumentation and Components"
   5. Branch Technical Position-Winter Precipitation Loads 3/24/75
   6. Inquiry by Chairman Rowden concerning Lightning Protection 7/9/76
   7. ANSI A58.2
   8. Licensee Event Reports
   9. 10 CFR Part 50

DEFINITION

TOPIC: II-2.B Onsite Meteorological Measurements Program

1. Definition:

To review the onsite meteorological measurements program to determine the extent that the licensee complies with 10 CFR Part 50, Appendix E, and Appendix I.

2. Safety Objective:

To assure that adequate meteorological instrumentation to quantify the off-site exposures from routine releases is available and maintained.

3. Status:

Onsite meteorological measurements programs are being reviewed as a part of the Appendix I evaluations.

4. References:

1. 10 CFR 50, Appendix E, Appendix I
2. R. G. 1.97, Rev. 1
3. R. G. 1.23
4. SRP Section 2.3.3

DEFINITION

TOPIC: II-2.C Atmospheric Transport and Diffusion Characteristics for Accident Analysis

1. Definition:

To review the atmospheric transport and diffusion characteristics assumed to demonstrate compliance with the 10 CFR 100 guidelines with respect to plant design, control room habitability, and doses to the public during and following a postulated design basis accident. This effort would examine the assumptions for:

(1) effects of explosive concentrations from onsite or offsite releases of hazardous material for consideration in structural design,

(2) calculation of relative concentration (X/Q) values for releases of radioactivity and toxic chemicals for consideration in control room habitability, and

(3) calculations of doses to the public resulting from releases of radioactivity to the atmosphere during and following a postulated design basis accident.

This effort is considered necessary because most original reviews were performed using the assumptions provided in Regulatory Guides 1.3 and 1.4 which have been found to be generally non-conservative based on evaluation of over 50 sites with actual meteorological observations.

2. Safety Objective:

To assure that the atmospheric transport and diffusion characteristics originally assumed to demonstrate compliance with the 10 CFR 100 guidelines are appropriate, considering additional onsite meteorological data and results of recent atmospheric diffusion experiments.

3. Status:

A review of long-term (annual average) atmospheric transport and diffusion characteristics is ongoing for Appendix I evaluations independent of the SEP effort. A study has also recently been performed by HMB for DOR for review of the meteorological assumptions for estimating control room dose consequences resulting from post-LOCA purges through tall stacks.

4. References:

1. 10 CFR 20
2. 10 CFR 50, Appendix A, Appendix I
3. 10 CFR 100
4. R. G. 1.3, 1.4
5. SRP Sections 2.3.4, 6.4, 2.2.1, 2.2.2, 2.2.3
6. TAC #4367

DEFINITION

TOPIC: II.2.D Availability of Meteorological Data in the Control Room

1. Definition:

Data from the onsite meteorological program should be available in the control room.

2. Safety Objective:

To assure that the licensee has appropriate meteorological logical data displayed in the control room to assess conditions during and following an accident to allow for: (1) early indication of the need to initiate action necessary to protect portions of the off-site public; and (2) an estimate of the magnitude of the hazard from potential or actual accidental releases.

3. Status:

No work currently being done on this subject for operating plants.

4. References:

1. 10 CFR 50. Appendix E, Appendix I
2. R. G. 1.97, Rev. 1
3. R. G. 1.23
4. SRP Section 2.3.3

DEFINITION

TOPIC: I-3.A Hydrologic Description

1. Definition:

   Hydrologic considerations are the interface of the plant with the hydrosphere, the identification of hydrologic causal mechanisms that may require special plant design or operating limitations with regard to floods and water supply requirements, and the identification of surface and ground water uses that may be affected by plant operation.

   These hydrologic considerations may have changed since they were reviewed at the licensing stage. A review of such changes, if any, should be performed including an assessment of their impact on the plants.

2. Safety Objective:

   To assure that the designs of safety-related structures, systems and components reflect consideration of appropriate hydrologic conditions, and to identify deficiencies in designs and/or operations that could contribute to accidental radioactive releases.

3. Status:

   No work currently being done on this subject for operating plants.

4. References:

   1. 10 CFR Parts 20, 50 and 100
   2. ANSI N170-1976
   3. R. G. 1.59
   4. SRP 2.4.1

DEFINITION

TOPIC: II-3.B Flooding Potential and Protection Requirements

1. Definition:

If the potential for floods exists and protection is required, the type of protection (sand bags, flood doors, bulkheads, etc.) will be reviewed to assure that equipment is available and that provisions have been made to implement the required protection.

2. Safety Objective:

To assure that safety-related structures, systems and components are adequately protected against floods.

3. Status:

Flooding protection requirements were reviewed on selected operating plants during the winter of 1976 due to the potential for flooding caused by ice accumulation and predictions for abnormally high spring runoff for some areas.

4. References:

1. 10 CFR Parts 50 and 100
2. R. G. 1.59
3. ANSI N170-1976
4. SRP 2.4.10

DEFINITION

TOPIC: II-3.B.1 Capability of Operating Plants to Cope with Design Basis Flooding Conditions

1. Definition:

Protection against postulated floods is accomplished, if necessary, by "hardening" the plant and by implementing appropriate technical specifications amd emergency procedures.

These technical specifications and flood emergency procedures need to be reviewed for plants licensed prior to 1972 to establish the degree of comformance with current criteria. Flooding criteria used for the design of older plants in not known.

2. Safety Objective:

Same as II-3.B

3. Status:

Same as II-3.B

4. References:

1. 10 CFR Part 100
2. ANSI N170-1976
3. R. G. 1.59
4. SRP Sections 2.4.3, 2.4.4, 2.4.5 and 2.4.7

DEFINITION

TOPIC: II-3.C Safety-Related Water Supply (Ultimate Heat Sink (UHS))

1. Definition:

To determine the adequacy of onsite water sources with respect to providing safety-related water during emergency shutdown and maintenance of safe shutdown. The location and inventory of safety-related water sources and the meteorological conditions to be used in evaluating both temperature and inventory of the sources should be established. Considerations of ice, low water, leak potential and underwater dams should be included. In most cases, plants operating prior to 1973 will have to be reviewed to establish the degree of conformance with current criteria. Prior to the issuance of Regulatory Guide 1.27 in 1973, the Standard Format and Content (now Regulatory Guide 1.70) provided the only guidelines to prospective applicants on UHS requirements. Since compliance was not required and hydrologic and meteorologic criteria had not been established, usually only minimal data was provided.

2. Safety Objective:

To assure an appropriate supply of cooling water during normal and emergency shutdown procedures.

3. Status:

No work currently being done on this subject for operating plants.

4. References:

1. 10 CFR Part 100
2. R. G. 1.27
3. SRPs 2.4.11 and 9.2.5

DEFINITION

TOPIC: II-4 Geology & Seismology

1. Definition:

Prior to the adoption of Appendix A to 10 CFR Part 100 in 1973, the Standard Format provided the only guidelines to prospective applicants regarding the type of geologic and seismic information needed by the AEC staff. The applicant, because compliance with Regulatory Guide 1.70 was not required, usually provided only minimal data. Therefore, a re-review of plants licensed prior to 1973 is needed in order to determine the adequacy of the plants design with respect to geologic and seismologic phenomena such as earthquakes, landslides, ground collapse and liquefaction. The review will also include ground motion and surface faulting and will establish the ground motion values and foundation conditions to be input into the structural reevaluation for seismic loads. (It is possible that some of the older plants would require assessing only the effects of new geologic and seismic discoveries on the site safety and the resulting design acceleration and/or the response spectra.)

2. Safety Objective:

To assure that accidents (e.g. LOCA) do not occur and that plants can safely shutdown in the event of geologic and seismologic phenomena which may occur at the site.

3. Status:

Selected plants are undergoing reevaluation of geology and seismology (SONGS 1 and Humboldt Bay). A plan for reevaluating operating plants was developed in 1975/76 but has not been implemented pending formation of the SEP.

4. References:

1. Standard Review Plan Section 2.5.1, 2.5.2, 2.5.3, 2.5.4 and 2.5.5
2. Appendix A to 10 CFR Part 100
3. Memo listing of values for operating plants (early 1976)

DEFINITION

TOPIC: II-4.A Tectonic Province

1. Definition:

   This sub-topic covers a specific area within the major topic Geology & Seismology. Its purpose is to reassess the tectonic province for operating plants based on more current knowledge. (A tectonic province is a region characterized by a relative consistency of the geologic structural features contained within. Tectonic provinces are used operationally as regions within which risk from earthquakes not associated with tectonic structures or faults is considered uniform. Usually the largest historical earthquake not associated with a specific structure can be assumed to occur anywhere within the same province.)

2. Safety Objective:

   To assure that plants can be safely shutdown in the event of geologic and seismologic phenoma which may occur at the site.

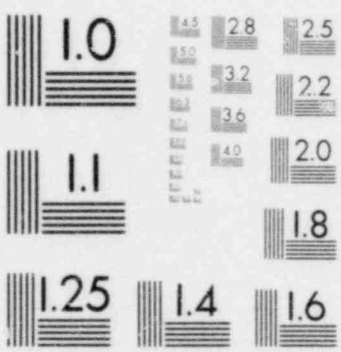
3. Status:

   The Geosciences Branch is currently attempting to delineate the boundaries of specific tectonic provinces (estimated completion date Fall-1977). The Site Safety Standards Branch is attempting to revise Appendix A to 10 CFR Part 100 so that the definition of tectonic province will more closely conform to its operational use (estimated completion date, 1978). We presently accept such provinces as generally proposed by King, Rogers or Eardley. Limited subdivision of these provinces has been allowed based on thorough geological and seismic analyses.

4. References:

   1. Appendix A to 10 CFR Part 100
   2. King, P. B., 1969, Tectonic Map of North America: U.S. Geological Survey
   3. Rogers, John, 1970, The Tectonics of the Appalachians: Wiey-Interscience N.Y., 271p
   4. Eardley. A.H., "Tectonic Divisions of North America" Bulletin of the American Association of Petroleum Geologists, Vol 35, pages 2229-2237, 1951

IMAGE EVALUATION TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

DEFINITION

TOPIC: II-4.B Proximity of Capable Tectonic Structures in Plant Vicinity

1. Definition:

This sub-topic covers a specific area within the major topic Geology & Seismology. Its purpose is to determine the expected shaking characteristics at a plant site from known capable faults. The ground motion associated with an earthquake generated by a capable fault or a tectonic structure may be larger than that associated with earthquakes in the same tectonic province not related to the structure.

2. Safety Objective:

To assure that plants can be safely shutdown in the event of geologic and seismologic phenomena which may occur at the site.

3. Status:

No work currently being done on this subject for operating plants.

4. References:

1. Appendix A to 10 CFR Part 100
2. Standard Review Plan, Section 2.5.2
3. Schnabel & Seed, 1973
4. R. G. 1.60

DEFINITION

TOPIC: II-4.C Historical Seismicity Within 200 Miles of Plant

1. Definition:

Determination of the Safe Shutdown Earthquake, SSE, is made with consideration of past seismicity in the vicinity of the plant. However, there is somet' es disagreement or inconsistency in reporting older earthquakes i .ne literature. Current high seismicity may also indicate possible hidden tectonic features.

The historical seismicity within 200 miles of the plants will be reviewed including all earthquakes of Richter magnitude greater than 3.0 or of Modified Mercalli intensity greater than III. Association with tectonic features and provinces should be included.

2. Safety Objective:

To assure that the SSE is compatible with past seismicity in the area.

3. Status:

No work currently being done on this subject for operating reactors.

4. References:

1. Richter, C. F. 1958, Elementary Seismology
2. 10 CFR 100 Part A

DEFINITION

TOPIC: II-4.D Stability of Slopes

1. Definition:

   Overstressing a slope may cause sudden failure with rapid displacement or shear strain which may damage safety related structures. The possibility of movement is evaluated by comparing forces resisting failure to those causing failure. An assessment of this ratio should be made to determine the safety factor.

2. Safety Objective:

   To assure that safety related structures, systems and components are adequately protected against failure of natural or man-made slopes.

3. Status:

   No work currently being done on this subject for operating plants.

4. References:

   1. SRP 2.5.5
   2. Appendix A to 10 CFR Part 100
   3. NAVFAC DM-7

DEFINITION

TOPIC: II-4.E Dam Integrity

1. Definition:

Dam integrity is the ability of a dam to safely perform its intended functions. These functions would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressures or erosion of soil materials and providing sufficient freeboard and outlet capacity to prevent overtopping.

2. Safety Objective:

To assure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained liquid are prevented. For many projects an important consideration is the necessity of assuring that an adequate quantity of water is available in times of emergency.

3. Status:

Additional guidance on assuring the integrity of dams is currently being developed by OSD in Regulatory Guide on "Inspection of Water Control Structures Associated with Nuclear Plant Facilities" and through the geotechnical engineering service contract with the U. S. Army Corps of Engineers on design of structures such as ultimate heat sinks.

4. References:

1. SRP 2.5.6
2. Appendix A to 10 CFR Part 100
3. EM 1110-2-1902, U. S. Army Corps of Engineers
4. EM 1110-2-2300, U. S. Army Corps of Engineers
5. R. G. 3.11

DEFINITION

TOPIC: II-4.F Settlement of Foundations and Buried Equipment

1. Definition:

Structural loads develop pressures in compressible strata which are not equivalent to the original geostatic pressures. Settlement and differential settlement should be evaluated.

2. Safety Objective:

To assure that safety related structures, systems and components are adequately protected against excessive settlement.

3. Status:

No work currently being done on this subject for operating plants.

4. References:

1. SRP 2.5.4
2. Appendix A to 10 CFR Part 100
3. NAVFAC DM-7

DEFINITION

TOPIC: III-1 Classification of Structures, Components and Systems (Seismic and Quality)

1. Definition:

Plant structures, systems, and components that are required to withstand the effects of a safe shutdown earthquake and remain functional should be classified as Seismic Category I. Systems and components important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Review the classification of structures, systems and components important to safety to assure they are of the quality level commensurate with their safety function.

2. Safety Objective:

To assure that structures, systems and components will fullfill their intended safety functions in accordance with design requirements. To assure that structures, systems and components necessary for safety will withstand the effects of the designated safe shutdown earthquake and will remain functional.

3. Status:

There is presently no DOR activity to confirm the classification of structures, components and systems important to safety of operating reactors.

4. References:

1. SRP 3.2.1
2. SRP 3.2.2
3. R. G. 1.26
4. R. G. 1.29

DEFINITION

TOPIC: III-2 Wind and Tornado Loadings

1. Definition:

Review the capability of the plant structures, systems and components to withstand design wind loadings in accordance with 10 CFR 50, Appendix A. The review includes the following: A: Design Wind Protection; B: Tornado Wind and Pressure Drop Protection; C: Effect of Failure of Structures not Designed for Tornado on Safety of Category I Structures, Systems and Components; D: Tornado Effects on Emergency Cooling Ponds.

2. Safety Objective:

To assure that Category I structures, systems and components are adequately designed for tornado winds and pressure drop, that any damage to structures not designed for tornado generated forces will not endanger Category I structures, systems and components, and that tornado winds will not prevent the water in the cooling ponds from acting as a heat sink.

3. Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

4. References:

1. 10 CFR 50, Appendix A, GDC 2
2. Standard Review Plans 3.3, 3.8, 9.2.5
3. Regulatory Guides 1.76, 1.117

DEFINITION

TOPIC: III-3.A Effects of High Water Level on Structures

1. Definition:

   If the high water level for the plant is reevaluated and found to be above the original design basis, then review the ability of the plant structures to withstand this water level.

2. Safety Objective:

   To provide assurance that floods or high water level will not jeopardize the structural integrity of the plant seismic Category I structures and, that seismic Category I systems and components located within these structures will be adequately protected.

3. Status:

   This review applies to all plants. There are no ongoing reviews concerning this matter.

4. References:

   1. 10 CFR 50, Appendix A, GDC 2
   2. Standard Review Plans 2.4, 3.4, 3.8
   3. Regulatory Guide 1.59, 1.102

DEFINITION

TOPIC: III-3.B Structural and Other Consequences (e.g. Flooding of Safety-Related Equipment in Basements) of Failure of Underdrain Systems

1. Definition:

Some plants rely on underdrain systems to limit the water table elevation at the plant to a safe level. Review underdrain systems of those facilities in which they are used.

2. Safety Objective:

To assure that the integrity of underdrain systems is maintained because a failure could lead to a rise in water table elevation which, in turn, could jeopardize the integrity of structures or the safety equipment within such structures.

3. Status:

The structural consequences of the failure of underdrain systems were thoroughly reviewed during the CP review of Douglas Point Units 1 and 2 and Perry Units 1 and 2. There are no ongoing reviews of this topic for operating facilities.

4. References:

1. 10 CFR 50, Appendix A, GDC 2
2. Standard Review Plans 2.4.13, 3.4 and 3.8

DEFINITION

TOPIC: III-3.C Inservice Inspection of Water Control Structures

1. Definition:

   Review the adequacy of the inservice inspection program of water control structures for operating plants to assure conformance with the intent of R. G. 1.127.

2. Safety Objective:

   To assure that water control structures of a nuclear power facility (e.g., dams, reservoirs, conveyance facilities) are adequately inspected and maintained so as to preclude their deterioration or failure which could result in flooding or in jeopardizing the integrity of the ultimate heat sink for the facility.

3. Status:

   This review applies to all plants. There are no ongoing reviews concerning this matter.

4. References:

   1. Regulatory Guide 1.127

DEFINITION

TOPIC: III-4 A Tornado Missiles

1. Definition:

Plants designed after 1972 have been consistently reviewed for adequate protection against tornadoes. The concern exists, however, that plants reviewed prior to 1972 may not be adequately protected, in particular those reviewed before 1968 when AEC criteria on tornado protection were developed.

An assessment of the adequacy of a plant to withstand the impact of tornado missiles would include:

(1) Determination of the capability of the exposed systems, components and structures to withstand key missiles (including small missiles with penetrating characteristics and larger missiles which result in an overall structural impact),

(2) Determination of whether any areas of the plant require additional protection.

The systems, structures, and components required to be protected because of their importance to safety are identified in Regulatory Guide 1.117.

2. Safety Objective:

To assure that those structures, systems and components necessary to ensure:

1. The integrity of the reactor coolant pressure boundary,

2. The capability to shut down the reactor and maintain it in a safe shutdown condition, and

3. The capability to prevent accidents which could result in unacceptable offsite exposures,

can withstand the impact of an appropriate postulated spectrum of tornado generated missiles.

TOPIC: III-4 A Tornado Missiles (Continued)

3. Status:

R^3C has approved case-by-case rereview of plants against criteria in Regulatory Guide 1.117 which establishes the systems, structures and components required to be protected against tornado missiles. This rereview was deferred pending the formation of the SEP.

The R^3C is in the process of rereviewing the SRP 3.5.1.4 which establishes appropriate missiles and impact velocities for new applications.

Electric Power Research Institute (EPRI) has missile research in progress.

4. References:

1. Standard Review Plan 3.5.1.4
2. Regulatory Guide 1.117

DEFINITION

TOPIC: III-4.B Turbine Missiles

1. Definition:

A number of non-nuclear plants and one nuclear plant (Shippingport) have experienced turbine disk failures. Rancho Seco has had chemistry problems leading to sodium deposits which caused stress-corrosion cracking of disks. Failure of turbine disks and rotors can result in high energy missiles which have the potential for resulting in plant releases in excess of 10 CFR 100 exposure guidelines.

Two areas of concern should be considered:

a. Design overspeed failures - material quality of disk and rotor, inservice inspection for flaws, chemistry conditions leading to stress-corrosion cracking, and

b. Destructive overspeed failures - reliability of electrical overspeed protection system, reliability and testing program for stop and control valves, inservice inspection of valves.

The focus of the review would be on turbine disk integrity and overspeed protection, including stop, intercept, and control valve reliability.

2. Safety Objective:

To assure that all the structures, systems, and components important to safety (identified in Regulatory Guide 1.117) have adequate protection against potential turbine missiles either by structural barriers or a high degree of assurance that failures at design (120%) or destructive (180%) overspeed will not occur.

3. Status:

No work currently being done on this subject for operating plants. Electric Power Research Institute (EPRI) has missile research in progress.

4. References:

1. Regulatory Guides 1.115 and 1.117
2. Standard Review Plan 3.5.1.3

DEFINITION

TOPIC: III-4.C Inte ally Generated Missiles

1. Definition:

Review the probability of missile generation and the extent to which safety-related structures, systems and components are protected against the effects of potential internally generated missiles (including missiles generated inside or outside the containment).

2. Safety Objective:

To provide assurance that the integrity of the safety-related stru res, systems and components will not be impaired and that they may be re ed on to perform their safety functions following any postulated internally generated missile.

3. Status:

No work currently being done on this subject for operating plants. Electric Power Research Institute (EPRI) has missile research in progress.

4. References:

1. Standard Review Plan 3.5.1.1 and 3.5.1.2

DEFINITION

TOPIC: III-4.D Site Proximity Missiles (Including Aircraft)

1. Definition:

   Review the extent to which safety-related structures, systems and components are protected against the effects of missiles postulated in Topic ~~III.4.C,~~ including postulated aircraft crashes and resulting fires. II-1.c

2. Safety Objective:

   To provide assurance that the integrity of the safety-related structures, systems and components will not be impaired and that they will perform their safety functions in the event of site proximity missile.

3. Status:

   No work currently being done on this subject for operating plants. Electric Power Research Institute (EPRI) has missile research in progress.

4. References:

   1. Standard Review Plan 3.5.1.5, 3.5.1.6, 3.5.2, 3.5.3

DEFINITION

TOPIC: III-5.A Effects of Pipe Break on Structures, Systems and Components Inside Containment

1. Definition:

Review the licensee's break and crack location criteria and methods of analysis for evaluating postulated breaks and cracks in high and moderate energy fluid system piping inside containment. The review includes consideration of compartment pressurization, pipe whip, jet impingement, environmental effects and flooding. Regulatory Guide 1.46 does not require that cracks be postulated inside containment. However, the recent proposed revision to SRP, Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" recommends that cracks be postulated inside containment. Old and current plants are not postulating cracks.

2. Safety Objective:

To assure that the integrity of structures, systems and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break is maintained.

3. Status:

This program has not been started for facilities licensed prior to about early 1974. Subsequent to that date, this topic was included in the OL review and has been completed for later facilities.

4. References:

1. 10 CFR 50, Appendix A, GDC 4
2. ASME Section III
3. Standard Review Plans 3.6.2, 3.8
4. Regulatory Guides 1.46 and 1.29

DEFINITION

TOPIC: III-5.B Pipe Break Outside Containment

1. Definition:

   Review the licensee's break and crack location criteria and methods of analysis for evaluating postulated breaks and cracks in high and moderate energy fluid system piping located outside containment. The review includes consideration of compartment pressurization, pipe whip, jet impingement, environmental effects and flooding.

2. Safety Objective:

   To assure that pipe breaks would not cause the loss of needed functions of safety-related systems, structures and components and to assure that the plant can be safely shutdown in the event of such breaks.

3. Status:

   This task is complete for all operating plants with the exception of 3 plants for which the review is in progress.

4. References:

   1. 10 CFR 50, Appendix A, GDC 4
   2. ASME Section III
   3. Standard Review Plan 3.6.1
   4. Regulatory Guides 1.46 and 1.29
   5. MEB 3-1
   6. Giambusso and O'Leary letters
   7. Pink Book 3-25
   8. Standard Review Plan 3.6.2

DEFINITION

TOPIC: III-6 Seismic Design Considerations

1. Definition:

Review and evaluate the original plant design criteria in the following areas: Seismic Input, Analysis and Design Criteria, Qualification of Electrical and Mechanical Equipment, Seismic Instrumentation, Seismic Categorization and the effect of failure of Non-Category I structures on the safety of Category I structures, systems and components.

2. Safety Objective:

To ensure the capability of the plant to withstand the effect of earthquakes.

3. Status:

Humboldt Bay and San Onofre plants are currently undergoing seismic review. Technical Assistance Contracts:

1. Seismic Conservatism (LLL)
2. Elasto-Plastic Seismic Analysis (LLL)
3. Seismic Review of Operating Plants (Newmark)

4. References:

1. Standard Review Plan, Sections 2.5, 3.7, 3.8, 3.9, and 3.10
2. Regulatory Guides 1.12, 1.60, 1.61, 1.92, 1.122

DEFINITION

TOPIC: III-7.A Inservice Inspection Including Prestressed Concrete Containments With Either Grouted or Ungrouted Tendons.

1. Definition:

Review licensee's inspection program for all Category I structures including steel, reinforced concrete and prestressed concrete containments. The program should include investigations for possible corrosion and cracking of steel containments, excessive cracking of concrete structures, lift-off tests of tendons, periodic testing of prestressing tendons for containments with grouted tendons, possible deterioration of prestressed containments.

2. Safety Objective:

To assure that the licensee's inspection program will detect any damaging deterioration of the structures and that they will be capable of performing as required by 10 CFR 50 Appendix A.

3. Status:

This review applies to all plants. There are no ongoing reviews concerning ths matter.

4. References:

1. 10 CFR 50, Appendix A
2. Standard Review Plan 3.8
3. Regulatory Guides 1.35 and 1.90

DEFINITION

TOPIC: III-7.B Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria

1. Definition:

Review the design codes, design criteria and load combinations for all Category I structures (i.e., containment, structures inside containment, and structures outside containment).

2. Safety Objective:

To provide assurance that the plant Category I structures will withstand the NRC specific design conditions without impairment or structural integrity or the performance of required safety functions.

3. Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

4. References:

1. 10 CFR 50, Appendix A, GDC 2 and 4
2. Standard Review Plan 3.8

DEFINITION

TOPIC: III-7.C Delamination of Prestressed Concrete Containment Structures

1. Definition:

Review the design of prestressed concrete containment structures to assess the likelihood of delamination occurring in the shell-walls or dome and to evaluate the consequences, if any.

2. Safety Objective:

To assure that the licensee's design and construction methods have provided a structure which will maintain its integrity and will perform its intended function. Delaminations (internal cracking of concrete in planes roughly parallel to the surface) could possibly reduce the capability of the concrete to withstand compression.

3. Status:

This review applies to all plants with prestressed concrete containments. A delamination occurred in the domes of the Turkey Point and Crystal River prestressed concrete containments. No evidence of such occurrences have been reported at other plants; however, no specific inspection have been made for any delaminations. It is not clear if the Structural Integrity Test or the existing ISI Programs would discover the existance of any delaminations.

4. References:

1. Safety Evaluation Reports for Turkey Point (50-250/251) and Crystal River (50-302)

DEFINITION

TOPIC: III-7.D Containment Structural Integrity Tests

1. Definition:

   Review the licensee's structural integrity testing procedure to assure compliance with the requirements of 10 CFR 50 Appendix A.

2. Safety Objective:

   To assure that the licensee's design and constructive methods provide a structure which will safely perform its intended functions.

3. Status:

   This review applies to all plants. To our knowledge all containments have had a structural integrity test. This opinion should be verified.

4. References:

   1. 10 CFR 50, Appendix A
   2. Standard Review Plans 3.8.1 and 3.8.2

DEFINITION

TOPIC: III-8.A Loose Parts Monitoring and Core Barrel Vibration Monitoring

1. Definition:

Inservice surveillance programs to detect loose parts and excessive motion of the main core support structure.

2. Safety Objective:

To detect loose parts or excessive vibration before they can cause flow blockage or mechanical damage to the fuel or other safety related components.

3. Status:

The NRC staff presently requires applicants to describe and licensees to implement a loose part detection program. Guidance for such a program is provided in a newly proposed R. G. 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors". The regulatory guide outlines the minimum system characteristics which the NRC staff feels are necessary for a workable system and combines this with a technical specification and reporting procedures for a complete and enforceable loose-part detection program.

The concept of detecting core barrel motion through the use of ex-core neutron detectors is well established. A proposed regulatory guide that describes an acceptable core barrel vibration monitoring program has been temporarily placed on "hold" to permit the NRC staff and its consultants (ORNL I&E Group) time to evaluate apparently anomalous data from core barrel motion monitoring programs that are presently in service as part of the technical specification requirements for certain licensees.

4. References:

1. "Operating Experience on Loose-Parts Monitoring Systems", RSB:EB/DOR (Draft)
2. CE Report CEN-5(P), "Palisades Reactor Internals Wear Report".
3. Regulatory Guide 1.133, "Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors".

DEFINITION

TOPIC: III-8.B Control Rod Drive Mechanism Integrity

1. Definition:

   Review and evaluate the reliability, operability and any reported mechanical failures in control ro drives.

2. Safety Objective:

   To assure that the integrity and operability of control rod drives is adequately maintained so that they will be capable of normal reactor control and prompt reactor shutdown, if required.

3. Status:

   The DOR Engineering Branch is currently evaluating the failure modes and internal component redesigns of BWR control rod drives to preclude stress corrosion and thermal fatigue cracking. There have been no reported generic failures of PWR drives.

4. References:

   NEDE-21021-P

DEFINITION

TOPIC: III-8.C Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

1. Definition:

Review the safety aspects that affect reactor vessel internals integrity for compliance with 10 CFR Part 50, including radiation damage, use of sensitized stainless steel and fatigue resistance.

2. Safety Objective:

To assure continued reactor vessel internals integrity and compliance with 10 CFR Part 50 and applicable industry Codes and Standards.

3. Status:

The Engineering Branch, DOR, currently has no review programs relating to reactor vessel internals integrity.

4. References:

1. 10 CFR 50, Appendix A
2. ASME Section III
3. ASTM A-262-70
4. Regulatory Guides 1.37, 1.44, 1.61

DEFINITION

TOPIC: III-8.D Core Supports and Fuel Integrity

1. Definition:

   Abnormal loading conditions on the core supports andd fuel assemblies due to seismic events or LOCAs could cause fuel damage due to impact between fuel assemblies and upper and lower grid plates or lateral impact between fuel assemblies and the core baffle wall. The resulting damage could result in loss of coolable heat transfer geometry, make it impossible to insert control rods, or cause releases of radioactive materials due to fuel pin failure.

2. Safety Objective:

   To assure that all credible loading conditions on core supports and fuel assemblies will not result in unacceptable fuel damage or distortion.

3. Status:

   DOR is currently reviewing the dynamic loads imposed on the fuel assemblies during a LOCA. Independent analyses are being conducted by staff consultants.

4. References:

   1. ASME Section III

DEFINITION

TOPIC: III-9 Support Integrity

1. Definition:

Review the design, design loads, and materials integrity including corrosion and fracture toughness and the inservice inspection programs of supports and restraints including bolting for the reactor vessel, steam generator, reactor coolant pump, torus and other class 1, 2 and 3 safety related components and piping systems.

2. Safety Objective:

To assure adequate support and/or restraint of safety related systems and components under normal and accident loads so that they will not be prevented from performing their intended functions because of support failures.

3. Status:

DOR has ongoing programs to review component supports. Current emphasis is on primary system supports and on piping system supports and restraints (snubbers).

4. References:

1. ASME Section III
2. Pink Book Generic Topics 3-5 and 3-43

DEFINITION

TOPIC: III-10.A Thermal-Overload Protection for Motors of Motor-Operated Valves

1. Definition:

The primary objective of thermal overload relays is to protect motor windings of motor-operated valves (MOV) against excessive heating. This feature of thermal overload relays could, however, interfere with the successful functioning of a safety related system. In nuclear plant safety system application, the ultimate criterion should be to drive the valve to its proper position to mitigate the consequences of an accident, rather than to be concerned with degradation or failure of the motor due to excess heating.

2. Safety Objective:

To assure that: (1) thermal overload protection, if provided for MOV's, should have the trip setpoint at a value high enough to prevent spurious trips due to design inaccuracies, trip setpoint drift, or variation in the ambient temperature at the installed location; (2) the circuits which bypass the thermal overload protection under accident conditions should be designed to IEEE Std. 279-1971 criteria, as appropriate for the rest of the safety related system; and (3) in MOV designs that use a torque switch instead of a limit switch to limit the opening or closing of the valve, the automatic opening or closing signal should be used in conjunction with a corresponding limit switch and thermal overload should remain as backup protection.

3. Status:

The staff position (Reference 1) is implemented on designs of new applications (CP and OL).

4. References:

1. Branch Technical Position EICSB 27, "Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves"
2. IEEE Std. 279-1971
3. RG 1.106

DEFINITION

TOPIC: III-10.B Pump Flywheel Integrity

1. Definition:

Review the PWR reactor coolant pump flywheel inservice inspection programs of operating plants to assure that they comply with the intent of Regulatory Guide 1.14 and review reports of flywheel flaws if found by inservice inspections. (BWR reactor coolant pumps do not have flywheels).

2. Safety Objective:

To assure that pump flywheel integrity is maintained to prevent failure at normal operating speeds and at speeds that might be reached under accident conditions and thus preclude the generation of missiles.

3. Status:

The inservice inspection programs for flywheels of older PWRs have not been reviewed for compliance with the intent of Regulatory Guide 1.14.

4. References:

1. Regulatory Guide 1.14

DEFINITION

TOPIC: III-10.C Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves

Definition:

At facilities which have completed the Low Pressure Coolant Injection System (LPCIS) modification, the recirculation pump discharge valves and bypass valves are now required to close upon initiation of LPCIS. The closure of these discharge valves is necessary to isolate a pipe break in a suction line to prevent loss of cooling water by reverse flow through the recirculation pump or its bypass line and out the break.

Safety Objective:

To assure effective core cooling in the event of a BWR recirculation line break on the pump suction line by closing the pump discharge valve and bypass line valve.

Status:

All licensees of facilities with completed LPCIS modification have been sent letters requesting that they apply for a license amendment to incorporate technical specification surveillance requirements on recirculation pump discharge valves and bypass valves. New BWRs have the LPCIS modification and technical specification surveillance requirements.

References:

1. Pink Book Issue 3-46, June 17, 1977

DEFINITION

TOPIC: III-11 Component Integrity

1. Definition:

   Review licensee's criteria, testing procedures, and dynamic analyses employed to assure the structural integrity and functional operability of safety related mechanical equipment under faulted conditions and accident loads. Included are mechanical equipment such as pumps, valves, fans, pump drives, heat exchanger tube bundles, valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays.

2. Safety Objective:

   To confirm the ability of safety related mechanical equipment having experienced problems to function as needed during and after a faulted or accident condition. The capability of safety related mechanical equipment to perform necessary protective actions is essential for plant safety.

3. Status:

   This review is not currently underway in DOR.

4. References:

   1. 10 CFR 50.55a
   2. 10 CFR 50, Appendix A, GDC 2, 4, 14, 15
   3. Standard Review Plan 3.9.2
   4. ASME Section III
   5. Regulatory Guides 1.20 and 1.68
   6. IEEE 344-1975
   7. Standard Review Plan 3.9.3

DEFINITION

TOPIC: III-12 Environmental Qualification of Safety-Related Equipment

1. Definition:

Safety-related electrical and mechanical equipment that is required to survive and function under environmental conditions calculated to result from a loss-of-coolant accident (LOCA) or a postulated main steam line break (MSLB) accident inside containment must be environmentally qualified. In addition, determine whether environment induced failures of non-safety-related equipment could interfere with the operation of safety equipment. Special attention should be given to the effect of beta radiation on exposed organic surfaces, such as gaskets.

2. Safety Objective:

To assure that the mechanical and Class IE electrical equipment of safety systems have been qualified for the most severe environment (temperature, pressure, humidity, chemistry and radiation) of design basis accidents.

3. Status:

Westinghouse is conducting a verification program which is expected to be completed by the end of 1977 for those plants qualified to IEEE - 323 (1971). The Office of Nuclear Regulatory Research (RES) is sponsoring programs relating to Class IE equipment qualification, the results of which can be utilized to determine the adequacy of the equipment previously qualified.

4. References:

1. NUREG 0153, Item 25, "Qualification of Safety-Related Equipment" December 1976
2. DOR Technical Activities, Category B, Item 34, "Environmental Qualifications of Safety-Related Equipment (Post LOCA)", May 1977
3. DSS Technical Activities, Category A, Item 33, "Qualification of Class IE Safety-Related Equipment", April 1977
4. R. G. 1.89

DEFINITION

TOPIC: IV-1.A Operation With Less Than All Loops In Service

1. Definition:

A number of BWR and PWR licensees have requested authorization to operate with one of the recirculation loops (BWR) or steam generator loops (PWR) out of service. These proposals are being reviewed generically with regard to analytical methods. Plant specific reviews will be done to determine appropriate Technical Specifications limits. Plant specific reviews will address results of LOCA analyses using generically approved methods. Analysis of accidents (other than LOCA) and operating transients resulting from operation in the (N-1) loop mode have been reviewed on a "lead plant basis". Most of this effort has been completed. Tests have been conducted by GE which show that significant core flow assymetries do not exist with single loop operation for two loop plants, however, there is backflow through inactive jet pumps. Therefore, for single loop operation, modifications are necessary in trip settings which take inputs from jet pump drive flow. These will be determined on a plant specific basis.

2. Safety Objective:

To provide assurance that operation with less than all coolant loops in operation will not result in decreased safety margins.

3. Status:

A combination of generic and plant specific reviews are being performed on both BWRs and PWRs.

4. References:

DEFINITION

TOPIC: IV-2 Reactivity Control Systems Including Functional Design and Protection Against Single Failures

1. Definition:

General Design Criterion 25 requires that the reactor protection system be designed to assure that fuel damage limits are never exceeded in the event of any single failure of the reactivity control systems. Reactivity control systems need not be designed single failure proof, but the protection system (which is designed against single failures) be capable of limiting fuel damage in the event of a reactivity control system single failure.

2. Safety Objective:

To assure that for all credible reactivity control system failures, the protection system will limit fuel damage to acceptable limits.

3. Status:

NRC has concluded that revisions to existing licenses is not warranted. Staff effort on this issue will continue at a low level.

4. References:

1. NUREG 0138, Issue No. 6
2. SRP 15.4.3

DEFINITION

TOPIC: IV-3 BWR Jet Pump Operating Indications

1. Definition:

If a jet pump BWR operates with a failed jet pump, it may be impossible to reflood the core in the event of a LOCA. Some BWRs have experienced jet pump instrument sensing line failures. With a sensing line failed, it may not be possible to accurately measure core flow or to detect failure of a jet pump.

2. Safety Objective:

To assure that the core flow can be determined. Also to assure the ability to detect a jet pump failure for a range of crack/break sizes at various locations on the pump.

3. Status:

This issue is currently being reviewed for Dresden 2/3 and Quad Cities 1/2. The topic has generic implications for all jet pump BWR plants.

4. References:

1. Letters from Commonwealth Edison Company to NRC dtd. September 19, 1975, March 3, 1976 and June 7, 1976.
2. Letter from NRC to Commonwealth Edison Company dtd. January 19, 1976.
3. Memo from J. H. Sniezek to D. L. Ziemann dtd November 19, 1975.

DEFINITION

TOPIC: V-1 Compliance with Codes and Standard (10 CFR 50.55a)

1. Definition:

   Review the licensee's inservice inspection and testing programs for Class 1, 2 and 3, pressure vessels, piping, pumps and valves and other safety related components to assure compliance with ASME Code, Section III and XI as required by 10 CFR 50.55a. This review will also include review of the inservice inspection and testing program applicable to isolation condensers of the early operating BWR's.

2. Safety Objective:

   To assure that the initial integrity of components is maintained throughout service life.

3. Status:

   NUREG N0081 was completed for reactor vessels not designed to Section III. The Engineering Branch conducts a generic review of all plants for compliance with inspection requirements of 50.55a(g) and fracture toughness require ments of 50.55a(i). This program will continue for the life of operating reactors.

4. References:

   1. 10 CFR 50.55a
   2. ASME Code, Sections III and XI
   3. NUREG 0081
   4. Memorandum from V. Stello to B. H. Grier, October 12, 1976.

DEFINITION

TOPIC: V-2 Applicability of Code Cases

1. Definition:

Review Code Cases currently accepted by the NRC, as indicated in Regulatory Guides 1.84 and 1.85.

2. Safety Objective:

To assure that only those Code Cases which are acceptable to the NRC are utilized by the licensee in the design, fabrication or repair of the plant. The use of Code Cases other than those contained in Regulatory Guides 1.84 and 1.85 are addressed on a case-by-case basis to assess their acceptability.

3. Status:

DOR Engineering Branch routinely reviews design modifications and component repairs (e.g., reactor vessel nozzles) to assure compliance with NRC acceptable Code Cases. The program is ongoing on an as-needed basis.

4. References:

1. Regulatory Guides 1.84 and 1.85

DEFINITION

TOPIC: V-3 Overpressurization Protection

1. Definition:

Inadvertent overpressurization of the primary system ~~at temperatures~~ below the nil ductility transition temperature may result in reactor vessel failure during heatup and pressurization. Such overpressure transients are caused by pressure surges when the primary system is water solid. The most severe transients have occurred when a charging pump starts up or inadvertent closing of a letdown valve with a charging pump running. Pressure temperature limits as a function of neutron fluence of the material at the reactor vessel beltline are specified in 10 CFR 50, Appendix G. All PWR licensees have been directed to institute interim administrative procedures to prevent damaging pressure transients and on a longer time scale to provide permanent protection which will probably include hardware changes such as high capacity safety/relief valves.

2. Safety Objective:

To protect the primary system from potentially damaging overpressurization transients during plant pressurization and heatup.

3. Status:

Generic review of all PWR licensee submittals is underway. Criteria for evaluation have been developed and refined by NRR/RES. An effort is being made to complete the review sufficiently early to ensure installation of mitigating systems by the end of 1977.

4. References:

1. NUREG 0138

DEFINITION

TOPIC: V-4 Piping and Safe End Integrity

1. Definition:

   Review the safety aspects that affect BWR and PWR piping and safe end integrity for compliance with 10 CFR Part 50, including fracture toughness, flaw evaluation, stress corrosion cracking in BWR and PWR piping, and control of materials and welding.

2. Safety Objective:

   To assure continued piping integrity and compliance with 10 CFR Part 50 and applicable industry codes and standards.

3. Status:

   The Engineering Branch, DOR, is conducting an ongoing program that includes the as-needed review of those aspects necessary to ensure the continuing integrity of piping systems important to safety including stress corrosion cracking of BWR colant pressure boundary piping. This program will continue for the life of operating reactors.

4. References:

   1. Technical Position, Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping
   2. ASME Section XI

DEFINITION

TOPIC: V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

1. Definition:

   Reactor Primary Coolant Leakage detection systems are a significant means of preventing primary system boundary failure by identifying leaks before failures occur.

2. Safety Objective:

   To provide reliable and sensitive leakage detection systems to identifying primary system leaks at an early stage before failures occur.

3. Status:

   This issue has been resolved for all plants which have recently received an OL by requiring conformance to Regulatory Guide 1.45. Individual older plants have not been systematically reviewed and leakage detection systems may need upgrading on a plant by plant basis.

4. References:

   1. R. G. 1.45
   2. TSAR, Dec. 1975
   3. SRP 5.2.5

DEFINITION

TOPIC: V-6 Reactor Vessel Integrity

1. Definition:

Review the safety aspects that affect BWR and PWR reactor vessel and nozzle integrity for compliance with 10 CFR Part 50, including fracture toughness, neutron irradiation, evaluation of surveillance programs, operating limitations, inservice inspection and flaw evaluation, and transient analyses.

2. Safety Objective:

To assure continued reactor vessel integrity and compliance with 10 CFR Part 50 and applicable industry Codes and Standards.

3. Status:

The Engineering Branch, DOR, is conducting ongoing programs that include the periodic review of aspects necessary to ensure the continued integrity of reactor vessels. These programs include BWR feedwater and CRD nozzle cracking, low upper shelf toughness, radiation effects, reactor vessel materials surveillance and updating of operating plants ISI programs and will continue for the life of operating reactors.

4. References:

1. NRC Status Report, BWR Feedwater Nozzle Cracking NUREG 0312
2. 10 CFR 50, Appendix G
3. Regulatory Guide 1.99
4. ASME Section III, Appendix G
5. ASTM E185
6. ASME Section XI
7. Pink Book 3-9, 3-21, 3-41

DEFINITION

TOPIC: V-7 Reactor Coolant Pump Overspeed

1. Definition:

   Review the potential for reactor coolant pumps to fail because of overspeed in the unlikely event of a major loss-of-coolant accident (LOCA).

2. Safety Objective:

   To assure that, in the event of a major LOCA, a reactor coolant pump assembly is not driven to a speed which would cause structural failure of the unit and result in missiles which could increase the consequences of the LOCA. Of greatest concern are the PWR pump flywheels because of their mass and rotational energy.

3. Status:

   An in-depth review of this topic was performed by the AEC staff and reported to the ACRS in 1973 (Reference 1). The staff concluded that, because of the small likelihood for the occurrence of a pump overspeed event that could seriously increase the consequences resulting from a LOCA (less than 10^{-8} per plant year), the action taken by the staff to assess this problem in a generic fashion outside the context of individual application reviews is an acceptable course to follow. A generic experimental program to be completed in 1978 by EPRI is expected to provide data to verify pump model overspeed predictions.

4. References:

   1. Letter, R. C. DeYoung to Harold G. Mangelsdorf (ACRS), August 6, 1973 transmitting "Report on Reactor Coolant Pump Overspeed During a LOCA", August 3, 1973.

   2. Regulatory Guide 1.14

DEFINITION

TOPIC: V-8 Steam Generator (SG) Integrity

1. Definition:

Review the safety aspects affecting operation of steam generators including secondary water chemistry, tube plugging criteria, inservice inspection, possibly including a dimensional inspection for proper evaluation of denting, steam generator tube leakage, tube denting, flow induced vibration of steam generator tubes, tube repair, and tube bundle or steam generator replacement.

2. Safety Objective:

To ensure that acceptable levels of integrity of that portion of the reactor coolant pressure boundary made up by the steam generator are maintained in accordance with current codes, standards, and/or regulatory criteria during normal and postulated accident conditions. The integrity of the steam generator is needed to ensure that leakage following a postulated design basis accident will not result in doses to the public in excess of 10 CFR Part 100 guidelines and that the emergency core cooling systems will be able to perform their safety functions.

3. Status:

Review of this topic is being performed by the Division of Operating Reactors. This effort will continue for the life of operating reactors.

4. References:

1. Regulatory Guide 1.83 (Revision 1)
2. Regulatory Guide 1.121
3. 10 CFR 50, Appendix A, GDC 30 and 32
4. Pink Book 3-27

DEFINITION

TOPIC: V 9 Reactor Core Isolation Cooling System (BWR)

1. Definition:

   RCIC has not been classified as a safety system. On GESSAR, for certain small breaks, GE assumed credit for RCIC as a backup for HPCI. The staff required GE to reclassify the RCIC system on the GESSAR 238 standard NSSS as a safety system.

2. Safety Objective:

   To ensure that the RCIC system is qualified as a safety system where credit is assumed in the safety analysis.

3. Status:

   GE has agreed to reclassify RCIC as a safety system on the GESSAR docket.

4. References:

DEFINITION

TOPIC: V-10.A Residual Heat Removal System Heat Exchanger Tube Failures

1. Definition:

RHR heat exchangers are designed to remove residual and decay heat so that the reactor can be placed in a safe cold shutdown condition and to maintain core cooling following a postulated loss-of-coolant accident. Some LWRs have a pressure control system on the cooling water piping system which maintains the pressure of the cooling water higher than the primary coolant pressure in the primary coolant side of the heat exchanger during plant cooldown operations, a leak in the tubes could result in back leakage of coolant water into the primary loop. Pressure in the cooling water side is maintained higher than that in the primary coolant side so that in the event of a tube failure there would be no leakage of radioactive fluids into the environment. Cooling water passing from the cooling water side of the heat exchanger into the primary coolant water could introduce impurities such as chlorides into the primary coolant system.

2. Safety Objective:

To assure that impurities from the cooling water system are not introduced into the primary coolant in the event of an RHR heat exchanger tube failure.

3. Status:

Recently there have been several RHR heat exchanger tube failures at operating BWRs. This issue has been defined as a DOR Category B Technical Activity.

4. References:

DEFINITION

TOPIC: V-10.B Residual Heat Removal System Reliability

1. Definition:

In all current plant designs the RHR system has a lower design pressure than the reactor coolant system (RCS). In most current designs the system is located outside of containment and is part of the emergency core cooling system (ECCS). However, it is possible for the RHR system to have different design characteristics. For example, the RHR system might have the same design pressure as the RCS, or be located inside of containment. The functional, isolation, pressure relief, pump protection, and test requirements for the RHR system are of concern in the safety review of reactor plants. Three types of RHR system designs are defined in Branch Position RSB 5-1.

On June 24, 1976, RRRC approved a revision of SRP 5.4.7 requiring a capability to go from hot to cold shutdown without offsite power and that all components necessary for cooldown from hot shutdown must be designed to safety grade seismic I standards, and be operable from the control room. System must be designed to meet the single failure criterion.

2. Safety Objective:

To ensure reliable plant shutdown capability using safety grade equipment.

3. Status:

Because of vender concern over the impact of the revision a review was conducted of three PWR plants, and as a result of this review the staff is proposing that Branch Position RSB 5-1 be modified but that the functional requirements be retained.

4. References:

1. BTP RSB 5-1
2. SRP 5.4.7
3. Memorandum E. G. Case to L. V. Gossick, July 15, 1976.
4. Summary of meeting September 22, 1976, "Capability to Achieve Cold Shutdown Using Safety Grade Systems and Equipment", C. O. Thomas, Docket No. STN-50-545, dated October 5, 1976.

DEFINITION

TOPIC: V-11.A Requirements for Isolation of High and Low Pressure Systems

1. Definition:

Several systems that have a relatively low design pressure are connected to the reactor coolant pressure boundary. The valves that form the interface between the high and low pressure systems must have sufficient redundancy and interlocks to assure that the low pressure systems are not subjected to coolant pressures that exceed design limits. The problem is complicated since under certain operating modes (e.g., shutdown cooling and ECCS injection) these valves must open to assure adequate reactor safety.

2. Safety Objective:

To assure that adequate measures are taken to protect low pressure systems connected to the primary system from being subjected to excessive pressure which could cause failures and in some cases potentially cause a LOCA outside of containment.

3. Status:

A preliminary review of a representative operating plant of each NSSS vendor was undertaken. Each low pressure system connected to the reactor coolant pressure boundary and penetrating the containment was examined. The investigation of a few potential areas of concern is continuing.

4. References:

DEFINITION

TOPIC: V-11.B RHR Interlock Requirements

1. Definition:

The RHR System is normally located outside of primary containment. It is an intermediate pressure system (usually 600 psia) and has motor operated valve (MOV) isolation valves connecting it to the RCS. If the RHR system were inadvertently connected to the RCS while the RCS is at pressure, a LOCA could result with a loss of all capability of core reflooding since the coolant inventory could be lost outside of containment. To prevent inadvertent opening of the MOV's while the RCS is at pressure, an "OPEN PERMISSIVE" interlock is provided.

If the operator shuts only 1 of the isolation valves prior to pressurizing the RCS there is a single valve RCS pressure boundary.

To ensure that both MOV's are shut during a startup and heatup an "AUTO-CLOSURE" interlock is provided that close the MOV's.

2. Safety Objective:

To ensure that operating reactor plants are adequately protected from overpressurizing the RHR system and potentially causing a LOCA outside of containment.

3. Status:

Several PWR plants do not have the auto closure feature on the RHR and at least 1 does not have the open permissive feature. Plants should be reviewed on a case-by-case basis factoring in (1) ASME code safety valve setting and capacity, (2) interlocks, (3) closure time of MOV's and (4) location of RHR.

4. References:

1. Proposed BTP RSB-5-1
2. RRRC Meeting #50, 6/24/76
3. GDC 34
4. Memorandum to R. C. DeYoung, V. Stello, et. al., from John Angelo entitled "RP-TR Staff Meeting of February 13, 1974 Regarding the Requirements on Shutdown Cooling Systems," February 28, 1974.
5. Letter to Mr. Clement Eicheldinger, Westinghouse Electric Corporation from Roger Boyd, November 12, 1975.
6. Letter to Mr. Ivan Stuart, General Electric Company, from Roger Boyd, November 12, 1975.
7. Letter to Mr. J. D. Geier, Illinois Power Company, from Robert Minogue, July 8, 1975.

DEFINITION

TOPIC: V-12.A Water Purity of Boiling Water Reactor Primary Coolant

1. Definition:

   Review the primary water monitoring and reactor water cleanup system capabilities, including the water purity, to determine if the maintenance of the necessary purity levels comply with Regulatory Guide 1.56. Review limits on quality control and defined provisions in the event of demineralizer break through.

2. Safety Objective:

   To assure that the water purity level is acceptably low to minimize the potential for intergranular stress corrosion cracking of austenitic stainless steel piping in the RCPB of BWRs, including assuring the implementation of the Regulatory Guide 1.56.

3. Status:

   Recommendations for specifying the use of additional conductivity measurements, and monitoring at various locations plus the use of pH and chloride measurements have been submitted to the Division of Standards Development to initiate a revision of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors", dated June 1973. To date, a generic review of operating BWRs has not been initiated and the current Regulatory Guide has been implemented in the Technical Specifications of only a few operating plants.

4. References:

   1. Memo to R. B. Minogue from R. E. Heineman, "Request for Revision of Regulatory Guide 1.56."

DEFINITION

TOPIC: V-13 Water Hammer

1. Definition:

Water hammer events have occured in light water reactor systems. Water hammer events increase the probability of pipe breaks and could increase the consequences of certain events such as the loss of coolant accident. The types of water hammer, the vulnerable systems (for example, containment spray, service water, feedwater and steam) and the safety significance of water hammer have been identified and defined in a staff report of May 1977.

2. Safety Objective:

To reduce the probability of water hammer events that have the potential to lead to pipe ruptures in LWR systems which are needed to mitigate the consequences of accidents or that might increase the consequences of accidents previously analyzed.

3. Status:

Generic review is underway. On March 10, 1977, an interdivisional DOR/DSS technical review group was formed to investigate the water hammer issue and to develop a program for its appropriate consideration in licensing reviews and for operating reactors. Consultant work has been performed by CREARE and Livermore Labs.

4. References:

1. "Water Hammer in Nuclear Power Plants", NRC Staff Report, June 1, 1977
2. "An Evaluation of PWR Steam Generator Water Hammer" by G. B. Wallis, P. H. Rotne, et. al. of CREARE Inc., draft, February 1977.
3. Lawrence Liver ore Laboratory "An Investigation of Pressure Transient Propagation in ressurized Water Reactor Feedwater Lines" (Preliminary) S. B. Sutton, April 15, 1977.
4. NRR Technical Activities, Category A, Item 1, Water Hammer, May 1977.

DEFINITION

TOPIC: VI-1 Organic Materials and Post Accident Chemistry

1. Definition:

   a. Organic materials

      The design basis for selection of paints and other organic materials is not documented for most operating reactors. Therefore, there is a need to review the suitability of paints and other organic materials used inside containment including the possible interactions of the decomposition products of organic materials with ESF's (such as filters).

   b. Post-accident chemistry

      Low pH solutions that may be recirculated within containment after a design basis accident may accelerate chloride stress corrosion cracking which may lead to equipment failure or loss of containment integrity. Low pH may also increase the volatility of dissolved iodines with a resulting increase in radiological consequences.

2. Safety Objective:

   a. Organic materials

      To assure that organic paints and coatings used inside containment do not behave adversely during accidents when they may be exposed to high radiation fields. In particular the possibility of coatings clogging sump screens should be minimized.

   b. Post-accident chemistry

      To assure that appropriate methods are available to raise or maintain the pH of solutions expected to be recirculated within containment after a DBA.

3. Status:

   No work currently being done on this subject for operating plants.

4. References:

   1. Standard Review Plan 6.1.2, 6.1.3
   2. Regulatory Guide 1.54

DEFINITION

TOPIC: VI-2.A Pressure-Suppression Type BWR Containments

1. Definition:

BWR pressure-suppression type containments (e.g., Mark I containment) are subjected to hydrodynamic loads during the blowdown phase of a LOCA. Those loads have the potential for damaging the components and structures (wetwell, internal structures, restraints, supports and connected systems) of the containment. During a relief valve blowdown into the suppression pool the wetwell (torus) shell and safety/relief valve restraints may be overstressed. The hydrodynamic loads were not explicitly identified and included in the design of the Mark I pressure-suppression containment.

2. Safety Objective:

To assure that the structural integrity of pressure suppression pool containments is maintained under hydrodynamic loading conditions. It has been determined that the upward forces during the blowdown phase following a LOCA potentially cause the Mark I torus to be lifted, causing failure of connecting systems and supports and leading to loss of the containment integrity. Structural modifications and/or changes in the mode of operation might be necessary to assure adequate safety margins.

3. Status:

Mark I containments are currently evaluated in a two step generic review program: The Short-Term Program (STP), completed May 1977, has focused on the determination of the magnitude and significance of hydrodynamic loads. In the Long-Term Program (LTP), to be completed by late 1978, the design basis loads will be finalized and the capability of the containment to withstand the loads within the original design structural margins will be verified. This verification will be based in part on research results from NRC and industry sponsored programs. As a result of the STP, the staff required that Mark I plants be operated with a drywell to wetwell differential pressure of at least one psi to reduce the vertical loads. In addition some licensees have modified the torus support system for additional safety margin.

4. References:

1. Pink Book - Generic Issues (April 1977)
   a. Mark I Containment - STP Technical Specifications
   b. Mark I Containment Evaluation - STP
   c. Mark I Containment Evaluation - LTP
   d. Mark I Safety/Relief Valve Line Restraints in Torus

2. DOR Technical Activities, Category A, April 1977
   a. Item 2, "Mark I Containment STP"
   b. Item 3, "Mark I Containment LTP"
   c. Item 23, "Mark II Containment"

3. DOR Technical Activities, Category B, May 1977, Item 12, "Assessment of Column Buckling Criteria"

4. DSS Technical Activities, Category A, April 1977, Item 31, "Determination of LOCA and SRV Pool Dynamic Loads for Water Suppression Containments"

DEFINITION

TOPIC: VI-2.B Subcompartment Analysis

1. Definition:

The rupture of a high energy line inside a containment subcompartment can cause a pressure differential across the walls of the subcompartment. In the case of a rupture of a PWR main coolant pipe adjacent to the reactor vessel, the subcooled blowdown produces pressure differentials in the annulus between the reactor vessel and the shield wall and also within the reactor vessel across the core barrel. This asymmetric pressure distribution generates loads on the reactor vessel support and on reactor vessel internals on other equipment supports and on subcompartment structures which have not been analyzed previously for most operating reactors.

2. Safety Objective:

To assure that the reactor vessel supports, reactor vessel internals, other equipment supports and subcompartment structures are designed with an adequate margin against failure due to these loads. The failure could result in a loss of ECCS capability.

3. Status:

The staff is reviewing the NSSS vendor and architect engineer design codes used to calculate the loads produced by the asymmetric pressure distribution. Analyses have been completed for a limited number of operating plants. The W TMD code is approved. Bechtel, Gilbert and United Engineering have submitted codes for review.

4. References:

1. Pink Book - Generic Issue, Item 3-5, "Asymmetric LOCA Loads - PWR", April 1977
2. DOR Technical Activities, Category A, Item 32, "Asymmetric LOCA Loads (Reactor Vessel Support Problem)", April 1977
3. DSS Technical Activities, Category A, Item 14, "Asymmetric Blowdown Loads on Reactor Vessel", April 1977
4. DPM Technical Activities, Category A, Item 2, "Reactor Vessel Supports (Asymmetric LOCA Loads from Sudden Subcooled Blowdown), April 1977

DEFINITION

TOPIC: VI-2.C Ice Condenser Containment

1. Definition:

   Operating experience from the D. C. Cook plant has indicated that sublimation and melting of ice causes a loss of ice inventory and related functional performance problems for the ice condenser system.

2. Safety Objective:

   To assure that a sufficient ice inventory is maintained and to assure the functional performance of the ice condenser system.

3. Status:

   The results of the surveillance program for ice inventory and of the functional performance testing (e.g., operation of vent doors) are periodically reviewed by the staff to determine whether the surveillance frequencies should be increased or other action should be taken. Recent surveillance testing indicates that the ice inventory is acceptable and that the D. C. Cook plant can be operated safely for the current fuel cycle. CONTEMPT-4 long term ice condenser code is expected to be completed by EG&G in October 1977.

4. References:

   1. DOR Technical Activities, Category B, Item 53, "Ice Condenser Containments", May 1977

DEFINITION

TOPIC: VI-2.D Mass and Energy Release for Postulated Pipe Breaks Inside Containment

1. Definition:

Review the methods and assumptions of the mass and energy release model, including containment temperature and pressure response, that was used in previously performed analyses of high energy line breaks inside containment, including the main steam line break.

2. Safety Objective:

To assure that design basis conditions (e.g., design pressure and temperature) for the containment structure and safety-related equipment are adequate. Determine if the models used in the earlier analyses provide adequate margins of safety when compared with the assumptions and models for current analytical techniques.

3. Status:

Mass and energy release models, including containment response models, are being reassessed to determine the degree of conservatism in the prediction of the containment pressure and temperature transient resulting from a PWR main steam line break. Application of those models to operating plants is contingent on the results of this reassessment. Mass and energy release models for operating BWR plants are considered in the Mark I Long Term Program and other BWR review efforts.

4. References:

1. DOR Technical Activities, Category B, May 1977,
    a. Item 1, "Pipe Break Inside Containment "
    b. Item 2, "Mass and Energy Release to Containment"
2. DSS Technical Activities, Category A, April 1977,
    a. Item 7, "Pipe Rupture Design Criteria",
    b. Item 29, "Main Steam Line Break Inside Containment"
3. DSS Technical Activities Report, December 1975, Item I-C.B.1, "Mass and Energy Release to Containment"

DEFINITION

TOPIC: VI-3 Containment Pressure and Heat Removal Capability

1. Definition:

The temperature and pressure conditions inside containment due to a postulated LOCA, main steam line or feedwater line break depends on the effectiveness of passive heat sinks and active heat removal systems (e.g., containment spray system).

2. Safety Objective:

To assure that the maximum temperature and pressure following a LOCA, main steam, or feedwater line break have been calculated with conservative assumptions and to assure that the passive heat sinks and active heat removal systems provide the full heat removal capability required to maintain the pressure and temperature below the design pressure and temperature of the containment, of safety-related equipment, and instrumentation inside containment.

3. Status:

The modified CONTEMPT computer code properly accounts for the condensation of superheated steam on containment passive heat sinks. The effects on the design temperatures within the containment is being studied for plant under licensing review.

4. References:

1. SRP, 6.2.1.1.A
2. DSS Technical Safety Activities Report, December 1975.
3. DOR Technical Activities, Category B, Item 62 "Effective Operation of Containment Sprays in LOCA", May 1977

DEFINITION

TOPIC. VI-4 Containment Isolation System

1. Definition:

Isolation provisions of fluid system of nuclear power plants limit the release of fission products from the containment for postulated pipe breaks inside containment and thus prevent the uncontrolled release of primary system coolant as a result of postulated pipe breaks outside containment. This must be accomplished without endangering the performance of post-accident safety systems. Review the primary containment isolation provisions, in particular, the containment sump lines and fluid systems penetrating containment. Review the design bases for containment ventilation system isolation valves to determine potential releases from the containment. Review the containment purge mode during normal operation with respect to various accident scenarios and consequences including operation of containment purge valves, closure times and leak tightness.

2. Safety Objective:

To assure that the primary containment isolation provisions meet the requirements of the GDC of 10 CFR 50 Appendix A, Criteria 54 through 57. Some of the operating plants may have too few or too many isolation provisions. Containment purging during normal operation in PWRs has raised a concern regarding the ability of the ventilation system isolation valves to close upon receipt of an accident signal. The use of resilient sealing materials in conjunction with the cycling of these valves has resulted in an increased degradation in the leakage integrity of the valve seats. To assure the adequacy of the maintenance and repair schedule to maintain the leakage integrity of the valves for the service life of the plant. To assure that containment purge operations will not adversely affect the consequences of postulated accidents.

3. Status:

The functional performance of the sump lines and ECCS systems is being reviewed in conjunction with the Appendix K submittals. Implementation criteria are being developed to apply the requirements of BTP - CSB 6-4 to containment purging practices and to improve the leakage integrity of ventilation system isolation valves.

4. References:

1. 10 CFR 50, Appendix A, Criteria 54 through 57
2. SRP 6.4.2
3. BTP CSB 6-4

DEFINITION

TOPIC: VI-5 Combustible Gas Control

1. Definition:

Review the combustible gas control system to determine the capability of the system to monitor the combustible gas concentration in the containment; to mix combustible gases within the containment atmosphere; and to maintain combustible gas concentrations below the combustion limits (e.g., by recombination, dilution, or purging). For facilities which share recombiners (portable) between units or sites, determine that the recombiners can be made available within a suitable time. For facilities which utilize purging as a primary means of combustible gas control, determine the radiological consequences of the system operation. Reevaluate hydrogen production and accumulation analysis to consider (1) reduction of Zr/water reaction on the basis of five times the Appendix K calculation amount and (2) potential increases in hydrogen production from corrosion of metals inside containment.

2. Safety Objective:

To prevent the formation of combustible gas explosive concentrations in the containment or in localized regions within containment, following a postulated accident; to assure that the radiological consequences of the system operation are acceptable.

3. Status:

Proposed 10 CFR 50.44 would permit a BWR licensee to propose an alternate combustible gas control system in lieu of inerting. Four such proposals for containment atmosphere dilution (CAD) systems are currently under review, and the COGAP II computer code is being revised to perform the system evaluations.

4. References:

1. Proposed Rule 10 CFR 50.44
2. DOR Technical Activities, Category A, Item 8, "Containment Purge During Normal Operation", April 1977
3. DOR Technical Activities, Category A, Item 14, "Inerting Requirements/CAD", April 1977
4. Branch Technical Position CSB 6-2
5. Standard Review Plan 6.2.5

DEFINITION

TOPIC: VI-6 Containment Leak Testing

1. Definition:

   Certain requirements of primary reactor containment leakage testing for water-cooled power reactors as described in Appendix J to 10 CFR Part 50 (issued February 1973) have been found to be conflicting, impractical for implementation, or subject to a variety of interpretation. Review the primary reactor containment leak testing program for operating nuclear plants.

2. Safety Objective:

   To assure that the containment leak testing program provides a conservative assessment of the leakage rate through individual leakage barriers and to assure that proper maintenance and repairs are conducted during the service life of the containment. The testing acceptance criteria are established to ensure that containment leakage following a postulated accident will not result in off-site doses exceeding 10 CFR 100 guidelines.

3. Status:

   A generic review for compliance with Appendix J and the review of requested exemptions to the regulation is currently underway. Proposed revisions to Appendix J to improve the testing requirements are under development.

4. References:

   1. 10 CFR 50, Appendix J
   2. 10 CFR 50, Appendix A, Criteria 52 and 53
   3. Pink Book - Generic Issue 3-10, "Containment Leak Testing - Appendix J", April 1977
   4. DOR Technical Activities, Category B, Item 33, "Containment Leak Testing Requirements", May 1977
   5. DSS Technical Activities, Category A, Item 30, "Containment Leak Testing". April 1977

DEFINITION

TOPIC: VI-7.A.1 ECCS Re-evaluation to Account for Increased Reactor Vessel Upper Head Temperature

1. Definition:

LOCA analyses for all Westinghouse reactors were conducted assuming that the water in the upper head region of the reactor vessel was the same as the inlet water temperature because of a bypass flow from the downcomer to the upper head. Temperature measurements made by Westinghouse indicate that the actual temperature of the upper head fluid exceeds cold leg temperature by 50 to 75% of the difference between hot leg and cold leg (inlet) temperature. All operating reactors were required to resubmit LOCA analyses using hot leg temperature for the upper head volume.

2. Safety Objective:

To provide revised LOCA analyses with correct upper head temperatures to assure that peak clad temperature limits are not exceeded.

3. Status:

Revised analyses have been received from all Westinghouse plants. All but three have been reviewed and approved.

4. References:

DEFINITION

TOPIC: VI-7-A-2 Upper Plenum Injection

1. Definition:

ECCS evaluation of Westinghouse two-loop plants was performed assuming that low pressure pumped injection is delivered directly to the lower plenum. However, ECC coolant is delivered directly into the upper plenum. Interaction of the cold injection water with the steam exiting from the core during refill and reflood and the heat transfer effects during the downward passage to the lower plenum have not been adequately considered.

2. Safety Objective:

To provide assurance that existing analyses with Westinghouse two-loop plants are acceptable either by showing that the present analyses are conservative, or by developing a new ECCS model which considers upper plenum injection.

3. Status:

The staff met with the Licensees and Westinghouse on January 11 and 26, 1977. The staff requested that the Licensees formally submit the information presented at the January 26, 1977 meeting. Two Westinghouse reports have been received to date. The staff is continuing to evaluate the problem. Research requested by NRR and performed by RES in the semiscale facility provided basis for evaluation.

4. References:

DEFINITION

TOPIC: VI-7.A.3 ECCS Actuation System

1. Definition:

   Review the ECCS actuation system with respect to the testability of operability and performance of individual active components of the system and of the entire system as a whole under conditions as close to the design condition as practical.

2. Safety Objective:

   To assure that all ECCS components (e.g. valves and pumps) are included in the component and system test. To assure that the frequency and scope of the periodic testing is adequate and meets the requirements of GDC 37.

3. Status:

   New applications (CP and OL) are reviewed in accordance with the Standard Review Plan and the references listed below. No specific activity for operating reactors is in progress.

4. References:

   1. R. G. 1.22
   2. Branch Technical Position EICSB 25, "Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole"
   3. 10 CFR 50, Appendix A, GDC 37

DEFINITION

TOPIC: VI-7.A.4 Core Spray Nozzle Effectiveness

1. Definition:

Core Spray systems are designed with a nozzle or a set of nozzles arranged above the core in such a way that, following a LOCA, a spray of water will be distributed over the top of the core so that each fuel bundle will receive a specified minimum flow which will provide adequate core cooling. Recent test data for a single nozzle in a steam environment noted partial or complete collapse of the spray cone and/or a shift in the direction of spray. These effects were not included in earlier full scale spray tests in air.

2. Safety Objective:

To assure adequate spray cooling following a LOCA.

3. Status:

The NRC has reviewed and accepted spray system performance for multiple nozzle spray systems, but has not accepted spray systems with a single overhead spray nozzle. Recent tests in Florida on the Big Rock Point spray nozzle indicates incomplete core coverage. As a result of these tests, NRC is requesting further testing by GE of multiple spray nozzles.

4. References:

1. Letter, K. Goller to OR BCs, "Generic Issue - Effects of Steam Environment on Core Spray Distribution for Non-jet Pump BWRs", dtd. December 7, 1976.
2. GE Topical Report, "BWR Core Spray Distribution". NEDO-10846

DEFINITION

TOPIC: VI-7-B ESF Switchover From Injection to Recirculation **Mode (Automatic ECCS Realignment)**

1. Definition:

   Most PWRs require operator action to realign ECCS systems for the recirculation mode following a LOCA.

   We have been requiring, on an ad hoc basis, some automatic features to realign the ECCS from the injection to the recirculation mode of operation.

2. Safety Objective:

   To increase the reliability of long term core cooling by not requiring operator action to change system realignment to the recirculation mode.

3. Status:

   A draft Branch Technical Position has been prepared which covers both ECC and containment spray systems. The proposed position is awaiting review by the RRRC.

4. References:

   1. Draft ANSI Standard N 660.

DEFINITION

TOPIC: VI-7.C ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Value

1. Definition:

The physical locking out of electrical sources to specific motor-operated valves required for the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e. g., closed rather than open, or opened rather than closed. There is some concern that interlock circuitry on ECCS valves may not be independent such that a single failure of an interlock due to equipment malfunction or operator error could defeat more than one interlock and cause the valves to be cycled to the wrong position.

2. Safety Objective:

To ensure that all power operated valves which could affect ECC system performance by being in the wrong position have power removed except when in use. This will ensure that ECC systems are not defeated by having a valve in the wrong position.

3. Status:

The staff plans to reconsider EICSB BTP-18 and RSB BTP-6-1.

4. References:

ACRS Generic concern II C-1.

DEFINITION

TOPIC: VI-7.C.1 Appendix K - Electrical Instrumentation and Control (EIC) Re-reviews

1. Definition:

During the Appendix K reviews of some facilities initially considered, a detailed EIC review was not performed. Re-review the modified ECCS of these facilities to confirm that it is designed to meet the most limiting single failure.

2. Safety Objective:

To assure that the modified ECCS is designed to meet the most limiting (design basis) single failure.

3. Status:

No current activity in DOR.

4. References:

1. R. G. 1.6
2. IEEE 308

DEFINITION

TOPIC: VI-7.C.2 Failure Mode Analysis ECCS

1. Definition:

   Failure modes and effects criticality analyses (FMECA) would be conducted for the purpose of systematically determining potential single failures in ECCS systems.

2. Safety Objective:

   To determine if single failures exist in ECC system as an aid in assessing overall plant safety.

3. Status:

   FMECA analyses has been conducted on the hydraulic portion of ECC systems of representative plant types. In addition single failure analyses were performed on each plant as a part of the required Appendix K analysis except for those plants with stainless steel clad cores.

4. References:

DEFINITION

TOPIC: VI-7.C.3 The Effect of PWR Loop Isolation Valve Closure During a LOCA on ECCS Performance

1. Definition:

Some PWR's are equipped with loop isolation valves. The effect of spurious closure of a loop isolation valve during a LOCA has never been analyzed. To ensure ECCS performance, power in some cases has been removed from loop isolation valves to prohibit spurious closure.

2. Safety Objective:

To assure that all plants with loop isolation valves have power removed during operation, or that other acceptable measures are taken to preclude inadvertent closing.

3. Status:

In most cases power has been removed from loop isolation valves, and this is confirmed as part of staff ECCS performance evaluations. This has not been confirmed for all plants with loop isolation valves.

DEFINITION

TOPIC: VI-7 D Long Term Cooling Passive Failures (e.g., Flooding of Redundant Components)

1. Definition:

The General Design Criteria require that the Emergency Core Cooling Systems (ECCS) shall be capable of providing adequate core cooling following a Loss-of-Coolant Accident, assuming a single failure in Emergency core Cooling Systems. The staff assumes the single failure to be either an active failure during the injection phase, or an active or passive failure during the long-term recirculation phase. The physical layouts of engineered safety feature pumps and components on some pressurized water reactors makes them vulnerable to flooding that might result from passive failures in system piping. Protection for pipe cracks or ruptures is not required because of the low probability of occurrence during the ECCS recirculation mode.

2. Safety Objective:

To provide for increased reliability of ECC systems by assuring that passive failures will not cause flooding and failure of ECCS valves and equipment.

3. Status:

Issue identified by Fluegge in letter to Rowden October 24, 1976. Staff response was prepared which concluded that " . . . consideration of this issue does not warrant revisions to any existing licenses or changes in present priority for addressing the treatment of passive failures subsequent to a LOCA. ECCS passive failure criteria being implemented by the staff requires consideration of additional leakage but no pipe breaks beyond the initiating LOCA.

4. References:

NUREG 0138 Issue No. 7

DEFINITION

TOPIC: VI-7 E ECCS Sump Design and Test for Recirculation Mode Effectiveness

1. Definition:

Following a LOCA in a PWR an emergency core cooling system (ECCS) automatically injects water into the system to maintain core cooling. Initially, water is drawn from a large supply tank. Water discharging from the break and containment spray collects in the containment building sump. When the supply tank has emptied to a predetermined level, the ECCS is switched from the "injection" mode to the "recirculation" mode. Water is then drawn from the containment building sump.

ECC systems are required to operate indefinitely in this mode to provide decay heat removal. Certain flow conditions could occur in the sump, which could cause pump failures. These include entrained air, prerotation or vortexing and losses leading to deficient NPSH.

2. Safety Objective:

To confirm effective operation of ECC systems in the recirculation mode.

3. Status:

Confirmation through pre-operational testing is now required on all CPs. Staff has been accepting scaled tests in lieu of pre-op tests at OL stage. Some plants have required modification to achieve vortex control.

4. References:

1. RFP - Vortex Technology (PWR)
2. Reg. Guide 1.79 para. b(2)

DEFINITION

TOPIC: VI-7.F Accumulator Isolation Valves Power and Control System Design

1. Definition:

For many loss-of-coolant accidents the performance of the ECCS in PWR plants depends upon the proper functioning of the accumulators. The motor-operated isolation valve, provided between the accumulator and the primary system, must be considered to be "operating bypass" (IEEE 279-1971) because, when closed, it prevents the accumulator from performing the intended protective function. The motor-operated isolation valve should be designed against a single failure that can result in a loss of capability to perform a safety function.

2. Safety Objective:

To assure that the accumulator isolation valve meets the "operation bypass" requirements of IEEE 279-1971 which states that the bypass of a protective function will be removed automatically whenever permissive conditions are not met. To assure that a single failure in the electrical system or single operator error cannot result in the loss of capability of an accumulator to perform its safety function.

3. Status:

Staff positions listed below are implemented on new applications. No systematic review program for operating reactors exists.

4. References:

1. IEEE Std. 279-1971
2. Branch Technical Position EICSB-4, "Requirements on Motor-Operated Valves in the ECCS Accumulator Lines"
3. Branch Technical Position EICSB-18, "Application of Single Failure Criteria to Manually-Controlled Electrically Operated Valves"

DEFINITION

TOPIC: VI-8 Control Room Habitability

1. Definition:

Control rooms in operating plants may not fully comply with General Design Criterion 19. This review should include, but not be limited to, analysis of the control room air infiltration rate, ventilation system isolability and filter efficiency, shielding, emergency breathing apparatus, short distance atmospheric dispersion, operator radiation exposure, and on-site toxic gas storage proximity.

2. Safety Objective:

To assure that the plant operators can safely remain in the control room to manipulate the plant controls after an accident.

3. Status:

DOR now reviews control room habitability in operating plants when related licensing actions (e.g., assessment of BWR Containment Air Dilution system post-LOCA radiological impact) require it. DSE has a technical assistance contract with the National Bureau of Standards to measure the control room air infiltration rate at a few operating plants. These measurements will be used to gauge the conservatism of the assumed air infiltration rates currently used by NRC. Some reviews are now in progress for plants we have reason to believe do not meet G. D. Criterion 19 (SONGS-1, Vermont Yankee, St. Lucie).

4. References:

1. SRP 6.4
2. 10 CFR 50, Appendix A, GDC 19
3. "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19", by K. G. Murphy and Dr. K. M. Campe, Proceedings of the Thirteenth AEC Air Cleaning Conference
4. R. G. 1.78
5. R. G. 1.95, Rev. 1

DEFINITION

TOPIC: VI-9.A Main Steam Line Isolation Seal System - BWR

1. Definition:

   Operating experience has indicated that there is a relatively high failure rate and variety of failure modes for components of the main steam isolation valve leakage control system (MSIV-LCS) in certain operating BWRs.

2. Safety Objective:

   To assure that leakage rate limits are not exceeded and the resulting calculated offsite doses do not exceed 10 CFR Part 100 guidelines using the staff's assumptions.

3. Status:

   Experience from surveillance testing as reported in recent Licensee Event Reports is compiled by DOR to serve as a basis for identifying design improvements and for preparing recommendations for future revisions to Regulatory Guide 1.96.

4. References:

   1. DOR Technical Activities, Category B, "Main Steam Line Leakage Control System", May 1977
   2. R. G. 1.96
   3. SRP 6.7

DEFINITION

TOPIC: VI-10.A Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing

1. Definition:

Review the reactor trip system (RTS) and engineered safety features (ESF) test program to verify RTS and ESF operability on a periodic basis and to verify RTS and ESF response time.

2. Safety Objective:

To assure the operability of the RTS and ESF, on a periodic basis, including verification of sensor response times. To ensure that the RTS and ESF test program demonstrates a high degree of availability of the systems and the response times assumed in the accident analyses are within the design specifications.

3. Status:

The test program of the RTS and ESF of new license applications is reviewed in accordance with the Standard Review Plan, including applicable Branch Technical Positions. Some licensees have agreed to perform response time measurements. Operability testing is probably performed, in one form or another, for most licensees of operating reactors.

4. References:

1. EICSB Branch Technical Position 24, "Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times"
2. Memorandum to V. A. Moore from V. Stello, October 12, 1973 (GESSAR Second Round of Questions No. 2 and No. 9)
3. R. G. 1.22, 1.105, 1.118

DEFINITION

TOPIC: VI-10.B Shared Engineered Safety Features, On-site Emergency Power, and Service Systems for Multiple Unit Stations

1. Definition:

The sharing of engineered safety features systems (ESF) systems, including on-site emergency power systems, and service systems for a multiple unit facility can result in a reduction of the number and of the capacity of on-site systems to below that which normally is provided for the same number of units located at separate sites. Review these shared systems for multiple unit stations.

2. Safety Objective:

To assure that: (1) the interconnection of ESF, on-site emergency power, and service systems between different units are not such that a failure, maintenance or testing operation in one unit will affect the accomplishment of the protection function of the system(s) in other units, (2) the required coordination between unit operators can cope with an incident in one unit and safe shutdown of the remaining unit(s), and (3) system overload coditions will not arise as a consequence of an accident in one unit coincident with a spurious accident signal or any other single failure in another unit.

3. Status:

A systematic review of shared ESF, on-site emergency power, and service systems for operating multiple unit stations is not being conducted. The EICSB Branch Technical Position is applied in the review of new licensee applications.

4. References:

1. EICSB Branch Technical Position 7, "Shared Onsite Emergency Electric Power Systems for Multi-Unit Stations"
2. R. G. 1.81

DEFINITION

TOPIC: VII-1.A Isolation of Reactor Protection System From Non-Safety Systems, Including Qualification of Isolation Devices

1. Definition:

Non-safety systems receive generally control signals from the reactor protection system (RPS) sensor current loops. The non-safety sensor circuits are required to have isolation devices to insure the independence of the RPS channels. Requirements for the design and qualification of isolation devices are quite specific. Recent operating experience has shown that some of the earlier isolation devices or arrangement at operating plants may not be effective.

2. Safety Objective:

To verify that operating reactors have RPS designs which provide effective and qualified isolation of non-safety systems from safety systems to assure that safety systems will function as required.

3. Status:

A limited generic review of isolation devices is being performed by DOR as part of a followup on LER No. 76-42/IT for Calvert Cliffs Unit 1 (TAC 6696). This limited generic review should be complete by August 1, 1977.

4. References:

1. LER No. 76-42/IT, Calvert Cliffs Unit 1 (TAC 6696)
2. SRP 7.2

DEFINITION

TOPIC: VII-1.B Trip Uncertainty and Setpoint Analysis Review of Operating Data Base

1. Definition:

As a result of Issue No. 13 in NUREG-0138 (Ref. 1) the staff is conducting a survey of plants at the OL stage of review to more specifically identify the margin between actual allowable trip parameter limits (from safety analyses standpoint) and actual reactor protection system (RPS) setpoints specified in the technical specifications. To clearly identify the setpoint margins, both the ultimate allowable and the specified nominal setting will be identified in the technical specifications.

2. Safety Objective:

To assure that the margins between the allowable trip parameters and the actual RPS setpoints are adequate and properly identified.

3. Status:

Implementation letters have been sent to the current applicants for operating licenses. The technical specifications for operating reactors are only being changed to include both values if a particular plant is converting to standard technical specifications.

4. References:

1. NUREG-0138, November 1976, Issue No. 13, "Instrument Trip Setpoints in Standard Technical Specifications"
2. Memo V. Stello to R. Boyd, dated February 18, 1977, Subject - Instrument Trip Setpoint Values
3. DOR Technical Activities, Category B, Item 29, "Instrument Trip Setpoints on Standard Technical Specifications", May 1977

DEFINITION

TOPIC: VII-2 Engineered Safety Features (ESF) System Control Logic and Design

1. Definition:

During the staff review of the Safety Injection System (SIS) reset issue (Ref. 1) the staff determined that the Engineered Safety Features Actuation Systems (ESFAS) at both PWRs and BWRs may have design features that raise questions about the independence of redundant channels, the interaction of reset features and individual equipment controls, and the interaction of the ESFAS logic that controls transfers between on-site and off-site power sources. Review the as-built logic diagrams and schematics, operator action required to supplement the ESFAS automatic actions, the startup and surveillance testing procedures for demonstrating ESFAS performance.

Several specific concerns exist with regard to the manual SIS reset feature following a LOCA: (1) If a loss of offsite power occurs after reset, operator action would be required to remove normal shutdown cooling loads from the emergency bus and re-establish emergency cooling loads. Time would be critical if the loss of offsite power occurred within a few minutes following a LOCA. (2) If loss of offsite power occurs after reset, some plants may not restart some essential loads such as diesel cooling water. (3) The plant may suffer a loss of ECCS delivery for some time period before emergency power picks up the ECCS system.

Review the ESF system control logic and design, including bypasses, reset features and interactions with transfers between onsite and offsite power sources.

2. Safety Objective:

To assure that the ESFAS's are designed and installed such that the necessary automatic control of engineered safety features equipment can be accomplished when required.

3. Status:

A review of ESFAS's of operating PWRs is being performed by DOR as part of the followup action to Reference 1 (to be completed end of 1977).

4. References:

1. NUREG-0138, November 1976, Issue No. 4, "Loss of Off-site Power Subsequent to Manual Safety Injection Reset Following a LOCA"
2. DOR Technical Activities Category A, April 1977, Item 22, "Loss of Off-site Power Subsequent to Manual Reset"
3. R. G. 1.41

DEFINITION

TOPIC: VII-3 Systems Required for Safe Shutdown

1. Definition:

Review plant systems that are needed to achieve and maintain a safe shutdown condition of the plant, including the capability for prompt hot shutdown of the reactor from outside the control room. Included also, a review of the design capability and method of bringing a PWR from a high pressure condition to low pressure cooling assuming the use of only safety grade equipment.

2. Safety Objective:

(1) To assure the design adequacy of the safe shutdown system to (i) initiate automatically the operation of appropriate systems, including the reactivity control systems, such that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences or postulated accidents and (ii) initiate the operation of systems and components required to bring the plant to a safe shutdown.

(2) To assure that the required systems and equipment, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown are located at appropriate locations outside the control room and have a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

(3) To assure that only safety grade equipment is required for a PWR plant to bring the reactor coolant system from a high pressure condition to a low pressure cooling condition.

3. Status:

A survey of remote shutdown capability of operating plants was performed some time ago by DOR. A technical activity has been proposed by DPM (see reference below) regarding safety objective (3). No other activities are in progress.

4. References:

1. DPM Technical Activities, Category A, Item 7, "Isolating Low Pressure Systems Connected to the RCPB", April 1977

DEFINITION

TOPIC: VII-4 Effects of Failure in Non-Safety Related Systems on Selected Engineered Safety Features

1. Definition:

Potential combinations of transients and accidents with failures of non-safety-related control systems were not specifically evaluated in the original safety analysis of currently operating reactor plants. Review the effects of control system malfunctions as initiating events for anticipated transients and also as failures concurrent with or subsequent to anticipated events or postulated accidents initiated by a different malfunction (e.g., the effect of the loss of the plant air system on the plant control and monitoring system). A complete discussion is provided in reference 1.

2. Safety Objective:

To assure that any credible combination of a non-safety-related system failure with a postulated transient or accident will not cause unacceptabl consequences.

3. Status:

A technical assistance contract with ORNL for failure mode analyses of control systems was initiated to determine sensitive areas of the plant designs. The results of this program in conjunction with the results of the failure mode and effects analyses for transients and accidents being performed under contract by INEL should provide a basis for any new review and safety requirements.

4. References:

1. NUREG-0153, Item 22, "Systematic Review of Normal Plant Operation and Conttrol System Failures", December 1976
2. Memorandum from V. Stello to R. J. Hart; dated 12/23/76, NRR letter No. 46.
3. DOR Task Force Report on SEP, Appendix B (TFL 118), November 1976
   a. Item 33 "Safety Related Control Power"
   b. Item 34 " Safety Related Instrumentation Power"
   c. Item 56 "Effect of Failure in Non-Safety Related Systems During Design Basis Events"
   d. Item 57 "Loss of Plant Air System (Effect on Plant Control and Monitoring)"
   e. Item 77 "Safety Related Control and Instrument Power"
4. DOT Recommended List of SEP Subjects, Spring 1977 C DOT 102, Item 100z, "Loss of Plant Air System (Effect on Plant Control and Monitoring)

DEFINITION

TOPIC: VII-5 Instruments for Monitoring Radiation and Process Variables During Accidents

1. Definition:

The adequacy of the instruments for monitoring radiation and process variables during accidents has not been reviewed for conformance with Regulatory Guide 1.97. A generic review is planned to assess the licensee's existing or proposed monitoring instruments during and following accidents to determine the adequacy of their range, response and qualifications, and to determine the sufficiency of the variables to be monitored. Certain instruments to monitor conditions beyond the design basis accidents will also be required in accordance with an RRRC determination (Reference 3).

2. Safety Objective:

To assure that plant operators and emergency response personnel have available sufficient information on plant conditions and radiological releases to determine appropriate in-plant and offsite actions throughout the course of any accident. The instrumentation should also provide recorded transient or trend information necessary for post-accident evaluation of the event. The ability to follow the course of accidents beyond the design basis accidents is also required.

3. Status:

Generic review of instrumentation to follow the course of accidents in operating plants and in all plants now under construction or seeking a construction permit will begin with the issuance of Regulatory Guide 1.97, Revision 1, this year. Submittals describing the facilities' post-accident instrumentation will be obtained from all operating licensees and reviewed by the end of 1978. The implementation of Regulatory Guide 1.97, Revision 1 on operating plants is proceeding independent of the SEP. RRRC has determined that Revision 1 to Regulatory Guide 1.97 should be treated as a Category 2 item (backfit on operating plants on a case by case basis).

4. References:

1. H. G. Mangelsdorf (ACRS) memo of 8/14/73 to L. M. Muntzing (Regulation)
2. L. M. Muntzing (Regulation) memo of 11/1/73 to H. G. Mangelsdorf (ACRS)
3. Proposed Revision 1 to Regulatory Guide 1.97, the Enclosure with the 4/4/77 memo R. B. Minogue (SD) to E. G. Case (NRR)
4. SRP 7.5
5. SRP 7.6
6. SRP 11.5
7. T. A. Ippolito (EICSB) memo of 8/12/74 to Emergency Instrumentation Task Force Members
8. Issue 21, NUREG-0153
9. RRRC Meeting Minutes (January 28, 1977)

DEFINITION

TOPIC: VII-6 Frequency Decay

1. Definition:

   In an issue of Reference 1 it is stated that the staff should require that a postulated rapid decay of the frequency of the offsite power system be included in the accident analysis and that the result be demonstrated to be acceptable. Alternatively, the reactor coolant pump (RCP) circuit breakers should be designed to protection system criteria and tripped to separate the pump motors from the offiste power system. Rapid decay of the frequency of the offsite power system has the potential for slowing down or breaking the RCP thereby reducing the coolant flow rates to levels not considered in previous analyses.

2. Safety Objective:

   To assure that the reactor coolant flow rate will not decrease below those assumed for a flywheel coastdown.

3. Status:

   Oak Ridge National Laboratory (ORNL), under a technical assistance program, is currently reviewing the frequency decay rate and its effects on RCP's. This program should be completed before the end of this year and this issue resolved.

4. References:

   1. NUREG-0138, November 1976, Item 9, "Frequency Decay"
   2. DOR Technical Activities Category B, May 1977, Item 27, "Frequency Decay"

DEFINITION

TOPIC: VII-7 Acceptability of Swing Bus Design on BWR-4 Plants

1. Definition:

The swing bus in the original BWR-4 design was used to provide power from either of two redundant electric sources to the LPCI valves by means of an automatic transfer scheme. A single failure in the transfer circuitry could result in paralleling the two redundant electric power sources thereby degrading their functional capabilities. Review licensee's swing bus automatic transfer circuitry to verify that it is immune to single failures which could lead to paralleling the two electric power sources.

2. Safety Objective:

To assure that the swing bus design will not propagate an electrical failure between two redundant power sources due to a single failure in the automatic transfer circuit at the BWR-4 swing bus.

3. Status:

During the course of generic review for compliance with ECCS criteria 10 CFR 50.46 and Appendix K some licensees have elected to modify the LPCI system to take credit for a portion of the LPCI flow. These facilities have replaced the swing bus design with a split bus configuration which complies with the requirements of Regulatory Guide 1.6. Not all facilities required a modification of the LPCI to meet the criteria and have retained the swing bus design.

The issue of the swing bus design was identified in Reference 1 and in addition in a letter from the ACRS dated December 12, 1976.

4. References:

1. NUREG-0138, November 1976, Item 3, "Acceptability of Swing Bus Design fo BWR-4 Plants"
2. DOR Technical Activities Category B, Item 26, "Acceptability of Swing Bus Design for BWR-4 Plants"
3. R. G. 1.6
4. General Design Criteria 17
5. IEEE 308

DEFINITION

TOPIC: VIII-1.A Potential Equipment Failures Associated With Degraded Grid Voltage

1. Definition:

A sustained degradation of the off-site power source voltage could result in the loss of capability of redundant safety loads, their control circuitry and the associated electrical components required to perform safety functions.

2. Safety Objective:

To assure that a degradation of the off-site power system will not result in the loss of capability of redundant safety-related equipment and to determine the susceptibility of such equipment to the interaction of on-site and off-site emergency power sources.

3. Status:

A program plan has been developed which includes a short-term program for the review of the emergency power systems of operating reactors and a long-term program to identify those conditions affecting the off-site power sources which may require that additional safety measures be taken.

4. References:

1. NUREG-0090-5, Report to Congress, "Abnormal Occurrences at Millstone 2", July-September 1976.
2. Status Report, "Review of Emergency Power Systems and Off-site Power Studies", February 23, 1977.
3. Memo, D. G. Eisenhut to K. R. Goller, April 20, 1977, Staff Positions (Short-Term Program)
4. Letters to Licensees, August 12 and 13, 1976
5. DOR Technical Activities, Category A, Item 9, "Potential Equipment Failures Associated with a Degraded Off-site Power Source", April 1977

DEFINITION

TOPIC: VIII-2 Onsite Emergency Power Systems - Diesel Generator

1. Definition:

Diesel generators, which provide emergency standby power for safe reactor shutdown in the event of total loss of offsite power, have experienced a significant number of failures. The failures to date have been attributed to a variety of causes, including failure of the air startup, fuel oil, and combustion air systems. In some instances the malfunctions were due to lockout. The information available to the control room operator to indicate the operational status of the diesel generator was imprecise and could lead to misinterpretation. This was caused by the sharing of a single annunciator station by alarms that indicate conditions that render a diesel generator unable to respond to an automatic emergency start signal and alarms that only indicate a warning of abnormal, but not disabling, conditions. Another cause was the wording on an annunciator window which did not specifically say that the diesel generator was inoperable (i.e., unable at the time to respond to an automatic emergency start signal) when in fact it was inoperable for that purpose. The review includes the qualification, reliability, operation at low loads, lockout, fuel oil and testing of diesel generators.

2. Safety Objective:

To assure that the diesel generator meets the availability requirements for providing emergency standby power to the engineered safety features.

3. Status:

Under a technical assistance request (in preparation) a thorough evaluation of all reported failures, including a comprehensive evaluation of diesel manufacturer and utility procedures for inspection, maintenance and operation will be performed. Letters were sent on March 29, 1977, to all of the affected licensees requesting additional information about diesel generator status indication in the control room. Our intention is to require that at least one annunciation be provided in the control room which will alarm whenever the diesel generator is unavailable due to any lockout condition.

4. References:

1. DOR Technical Activities, Category B, Item 35, "Diesel Reliability"
2. DOR Technical Activities, Category B, Item 20, "Diesel Generator Lockout, Reset and Annunciation"
3. R. G. 1.108
4. Pink Book, Generic Issue 3-11, "Diesel Generator Lockout", April 1977

DEFINITION

TOPIC: VIII-3.A Station Battery Capacity Test Requirements

1. Definition:

Revi w the Technical Specification, including the test program, with regard to the requirement for periodic surveillance testing of onsite Class IE batteries and the extent to which the test meets Section 5.3.6 of IEEE Std. 308-1971, to determine battery capacity.

2. Safety Objective:

To assure that the onsite Class IE battery capacity is adequate to supply d-c power to all safety related loads required by the accident analyses and is verified on a periodic basis. This effort is needed to ensure that the test to determine battery capacity includes (1) an acceptance test of battery capacity performed in accordance with Section 4.1 of IEEE Std. 450-1975, (2) a performance discharge test listed in Table 2 of IEEE Std. 308-1971, performed according to Sections 4.2 and 5.4 of IEEE Std. 450-1975; and (3) a battery service test described in Section 5.6 of IEEE Std. 450-1972, to be performed during each refueling operation.

3. Status:

The review of station battery capacity test requirements is applicable to all operating reactors. There is no ongoing effort on this subject for operating reactors except for those reactors converting to Standard Technical Specifications.

4. References:

1. SRP, Appendix 7-A, BTP EICSB 6
2. IEEE Std. 380-1971, 1974
3. IEEE Std. 450-1975
4. Memorandum to R. H. Vollmer from J. G. Keppler, March 20, 1972
5. Memorandum to R. Carlson from V. D. Thomas, January 18, 1972

DEFINITION

TOPIC: VIII-3.B DC Power System Bus Voltage Monitoring and Annunciation

1. Definition:

Review the d-c power system battery, battery charger, and bus voltage monitoring and annunciation design with respect to d-c power system operability status indication to the operator. This information is needed so that timely corrective measures can be taken in the event of loss of an emergency d-c bus.

2. Safety Objective:

To assure the design adequacy of the d-c power system battery and bus voltage monitoring and annunciation schemes such that the operator can (1) prevent the loss of an emergency d-c bus; or (2) take timely corrective action in the event of loss of an emergency d-c bus.

3. Status:

The review of the d-c power system battery and bus voltage monitoring and annunciation adequacy as it relates to the loss of an emergency d-c bus is applicable to all operating reactors. This topic is included in the NRR Technical Activity "Adequacy of Safety Related DC Power Supplies".

4. References:

1. SRP 8.3.2

DEFINITION

TOPIC: VIII-4 Electrical Penetrations of Reactor Containment

1. Definition:

Review the electrical penetration assembly with respect to the capability to maintain containment integrity during short- circuit current conditions and mechanical integrity during the worst expected fault current vs. time conditions resulting from single random failures of circuit overload protection devices.

2. Safety Objective:

To assure that all electrical penetrations in the containment structure, whether associated with Class IE circuits or non-class IE circuits, are designed not to fail from electrical faults during a LOCA.

3. Status:

The subject of electrical cable penetrations was identified in Reference 1 and has been proposed as a Technical Activity Category A item by DSS (Reference 2). The purpose of that activity is a re-evaluation of the penetrations to clarify and augment the design safety margin.

4. References:

1. NUREG-0153, Issue 18, "Electrical Cable Penetration of Reactor Containment", December 1976
2. DSS Technical Activity, Category A, Item 36, "Electrical Cable Penetrations of Reactor Containment", April 1977
3. R. G. 1.63
4. IEEE Std. 317 - 1976

DEFINITION

TOPIC: IX-1 Fuel Storage

1. Definition:

Review the storage facility for new and irradiated fuel, including the cooling capability and seismic classification of the fuel pool cooling system of the spent fuel storage pool. Specifically review the expansion of the on-site spent fuel storage capacity, including the structural response of the fuel storage pool and the racks, the criticality analyis for the increased number of stored fuel assemblies at reduced spacing, and the capability of the spent fuel cooling system to remove the additional heat load.

2. Safety Objective:

To assure that new and irradiated fuel are stored safely with respect to criticality ($k_{eff} < 0.95$) cooling capability (outlet temperature < 150°F), shielding, and structural capability.

3. Status:

Approximately two thirds of the operating reactor plants have requested authorization to increase the storage capacity of their fuel storage pool. The applications are reviewed on a case-by-case basis. New or modified storage rack designs are reviewed against current design criteria; however, the existing pool structure is based on original design criteria.

4. References:

1. DOR Technical Activities, Category A, item 27, "Increase in Spent Fuel Storage Capacity", April 1977
2. ANSI-210, "Design Objectives for Spent Fuel Storage Facilities"

DEFINITION

TOPIC: IX-2 Overhead Handling Systems - Cranes

1. Definition:

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of PWR and BWR spent fuel storage facilities and inside the reactor building. If a heavy object (e. g., a shielded cask) were to drop on the spent fuel or on the reactor core during refueling, there could be a potential for overexposure of plant personnel and for release of radioactivity to the environment. Review the overhead handling system, including sling and other lifting devices, and the potential for the drop of a heavy object on spent fuel including structural effects.

2. Safety Objective:

To assess the safety margins, and improve margins where necessary, of the overhead handling systems to assure that the potential for dropping a heavy object on spent fuel is within acceptable limits and that the potential radiation dose to an individual does not exceed the guidelines of 10 CFR Part 100.

3. Status:

Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants" was issued for comment in February 1976 and references various industry standards. New applications (CP and OL) are reviewed in accordance with the APCSB Branch Technical Position 9-1 which is identical to Regulatory Guide 1.104.

The review of overhead handling systems of operating reactor facilities is performed on a generic basis and has also been identified as a DOR Technical Activity Category A.

4. References:

1. R. G. 1.104
2. APCSB Branch Technical Position 9-1, "Overhead Handling Systems for Nuclear Power Plants"
3. Pink Book - Generic Issue 3-22, "Fuel Cask Drop Analysis", April, 1977.
4. DOR Technical Activities, Category A Item 50, "Control of Heavy Loads Over Spent Fuel", April 1977

DEFINITION

TOPIC: IX-3 Station Service and Cooling Water systems

1. Definition:

Review the station service water and cooling water systems that are required for safe shutdown during normal, operational transient, and accident conditions, and for mitigating the consequences of an accident, or preventing the occurrence of an accident. These include cooling water systems for reactor system components (component cooling water system), reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system (ECCS). These systems also include the station service water system, the ultimate heat sink and the interaction of all of the above systems.

The review of these systems includes the pumps, heat exchangers, valves and piping, expansion tanks, makeup piping, and points of connection or interfaces with other systems. Emphasis is placed on the cooling systems for safety-related components such as ECCS equipment, ventilation equipment, and reactor shutdown equipment.

The following specific aspects of those systems will be considered in the review:

a. physical separation of redundant cooling water systems that are vital to the performance of engineered safety systems components,

b. availability of cooling water to primary reactor coolant pumps,

c. requirements for makeup water of cooling water systems,

d. effect of water overflow from tanks,

e. circulating water system barrier failure protection.

2. Safety Objective:

To assure that the station service and cooling water systems have the capability, with adequate margin, to meet their design objective. To assure, in particular, that

a. systems are provided with adequate physical separation such that there are no adverse interactions among those systems under any mode of operation;

b. cooling water is provided to the bearings of the primary reactor coolant pumps by two independent essential service water systems for PWR plants that take credit for core cooling by pump coast down. In addition, it should be demonstrated that the possibility of simultaneous loss of water in both essential service water systems by valve closure is sufficiently small;

c. sufficient cooling water inventory has been provided or that adequate provisions for makeup are available;

d. tank overflow cannot be released to the environment without monitoring and unless the level of radioactivity is within acceptable limits.

e. vital equipment necessary for achieving a controlled and safe shutdown is not flooded due to the failure of the main condenser circulating water system.

3. Status:

The station service and cooling water systems of application presently under review are evaluated in accordance with the Standard Review Plan (Sections 9.2.2 and 10.4.5). Some of the specific concerns identified above are under generic review or have been proposed for a Technical Activity in NRR in accordance with the references below.

4. References:

1. ACRS letter, Fraley to Gossick, "Analysis of Systems Interactions", November 1, 1976.
2. Memorandum, Rusche to Gossick, ACRS Subcommittee on Systems Interactions, January 1977.
3. DPM Technical Activities, Category A, Item DPM-15, "Systems Interactions in Nuclear Power Plants", April 1977
4. Memo to Tedesco to Vassallo, Auxiliary Systems Branch Q2 on Yellow Creek Nuclear Plant, Item 010.42, January 31, 1977 (cooling water for RCP)
5. DSS Technical Safety Activities Report, "Cooling Water System Makeup Water Requirements (For Safety Systems)", December 1975
6. Pink Book - Generic Issue 3-20, "Flood of Equipment Important to Safety (Generic)", April 1977
7. DOR Technical Activities, Category A, Item 15, "Flood of Equipment Important to Safety", April 1977

DEFINITION

TOPIC: IX-4 Boron Addition System (PWR)

1. Definition:

Review the boron addition system (PWR), in particular with respect to boron precipitation during the long term cooling mode of operation following a loss of coolant accident.

2. Safety Objective:

To assure that boron precipitation will not impair the operability of valves or components in the boron addition system which could compromise its capability to control core reactivity during normal, transient, or emergency shutdown conditions or that would result in flow blockage through the core during the long term core cooling mode following a loss of coolant accident.

3. Status:

Operating PWR reactors, with the exception of the Combusiton Engineering reactors, have been reviewed and found to be acceptable in regard to boron precipitation following a loss of coolant. There are still certain outstanding issues that need to be resolved on this issue for Combustion Engineering reactors. In regard to the precipitation of boron in the boron addition system in both BWRs and PWRs, certain older plants may not have been reviewed in sufficient detail to assure that system reliability is adequate.

4. References:

1. SRP 9.3.4

DEFINITION

TOPIC: IX-5 Ventilation Systems

1. Definition:

Review the design and operation of ventilation systems whose function is to maintain a safe environment for plant personnel and engineered safety features equipment. For example, the function of the spent fuel pool area ventilation system is to provide ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational transients, and following postulated fuel handling accidents. The function of the engineered safety feature ventilation system is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design basis accidents.

2. Safety Objective:

To assure that the ventilation systems have the capability to provide a safe environment, under all modes of operation, for plant personnel (10 CFR Part 20) and for engineered safety features (e.g., to assure that the diesel room has redundant outside air intakes and removed from the exhaust discharge).

3. Status:

Ventilation systems of plants under current review (CP and OL application) are currently evaluated in accordance with the Standard Review Plan. No specific issues or concerns have been identified for operating reactor plants.

4. References:

1. SRP 9.4.1 through 9.4.5

DEFINITION

TOPIC: IX-6 Fire Protection

1. **Definition**:

Review the fire protection program of operating reactor plants to determine whether improvements are required in accordance with the APCSB Technical Position 9.5-1, Appendix A, (Reference 2). The fire protection program encompasses the components, procedures and personnel utilized in carrying out all activities of fire protection and includes such things as fire prevention, detection, annunciation, control, confinement, suppression, extinguishment, administrative procedures, fire brigade organization, inspection and maintenance, training, quality assurance, and testing. The review includes such items as: (1) the use of insulation inside the containment and (2) the consequences of the inadvertent release of hydrogen into the plant.

2. Safety Objective:

To assure that, in case of a fire within the plant, the integrity of the engineered safety features is not compromised and that the safe shutdown capability and control of the plant is not lost.

3. Status:

A generic review of fire protection for operating plants is underway. All licensees were requested by letter (May 11, 1976) to submit an evaluation of their fire protection program for that plant in comparison with the APCSB Technical Position 9.5-1. Subsequently, in September 1976 the licensees were provided with Appendix A to the BTP 9.5-1 which presents acceptable alternatives for operating plants.

4. References:

1. NUREG 0050, "Recommendations Related to Browns Ferry Fire", February 1976
2. APCSB BTP 9.5-1, Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976"
3. R. G. 1.120
4. Pink Book - Generic Issue 3-18, "Fire Protection", April 1977
5. DOR Technical Activities, Category A, Item 28, "Fire Protection", April 1977
6. DSS Technical Activities, Category A, Item 32, "Fire Protection", April 1977
7. ACRS Letter, R. F. Fraley to L. V. Gossick, (Analysis of Systems Interactions - Item 6), November 1, 1976

DEFINITION

TOPIC: X Auxiliary Feedwater System

1. Definition:

   Review the auxiliary feedwater system, associated instrumentation, and connection between redundant systems. The review includes the aspects of pump drive and power supply diversity (e.g., electrical and steam-driven sources), and the water supply sources for the auxiliary feedwater system.

2. Safety Objective:

   To assure that the auxiliary feedwater system can provide an adequate supply of cooling water to the steam generators for decay heat removal in the event of a loss of all main feedwater. Older PWR plants may not meet the requirement for pump drive and power supply diversity.

3. Status:

   Reviews for new license applications are performed in accordance with the SRP. This topic is not under active review for operating plants.

4. References:

   1. SRP, 10.4.9
   2. APCSB BTP 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWR Plants"

DEFINITION

TOPIC: XI-1 Appendix I

1. Definition:

A generic review of all operating plants to determine their capability to comply with Appendix I, 10 CFR 50, and to prevent explosions in the gaseous radwaste system is currently underway.

2. Safety Objective:

To provide assurance that radioactive gaseous effluents from the facility can be kept "as low as reasonably achievable" as defined in Appendix I, 10 CFR Part 50, and to assure adequate control of the mixture of gases in the gaseous radwaste system to prevent explosions.

3. Status:

A generic review of all ORs for their capability to conform with Appendix I, 10 CFR Part 50, is currently underway by DSE. Upon the completion of this review, new gaseous and liquid radiological effluent and monitoring technical specifications will be issued to all ORs. This will include new technical specifications on gaseous radwaste systems which may contain explosive gas mixtures to meet present criteria. The estimated completion date of this review is 1979.

4. References:

1. 10 CFR Part 20
2. 10 CFR Part 50, Appendix I
3. 10 CFR Part 50, Appendix A
4. GDC 60, 61, 63 and 64
5. SRP 11.3

DEFINITION

TOPIC: XI-2 Radiological (Effluent and Process) Monitoring Systems

1. Definition:

Onsite radiological monitoring systems are used to:

a. assess the proper functioning of the process and waste treatment systems,
b. assure that radioactive releases do not exceed the appropriate guidelines, and
c. measure actual releas o evaluate their environmental impact.

There is concern about the adequacy of radiation monitoring systems. A survey of 12 plants has been initiated. The results of this survey will indicate whether this area needs to be reviewed for all operating plants. Re-review would include the monitor's sensitivity, range, location, and calibration techniques.

2. Safety Objective:

To provide reasonable assurance that the licensee adequately monitors the releases of radioactive materials in liquid and gaseous effluent and that the releases are properly restricted. To provide assurance that the licensee adequately monitors the operation of equipment that contains or may contain radioactive material.

3. Status:

A technical assistance program has been initiated at Brookhaven National Laboratory with the scope including the above safety objectives.

4. References:

1. 10 CFR 20.106
2. 10 CFR 50.36a
3. 10 CFR 50, Appendix I
4. 10 CFR 50, Appendix A, Criteria 60, 61, 63, and 64
5. SRP 11.5

DEFINITION

TOPIC: XII (Section on RADIATION PROTECTION Intentionally Left Blank)

DEFINITION

TOPIC: XIII-1 Conduct of Operations

1. Definition:

The organization, administrative controls and operating experience will be reviewed. The existing organization and administrative controls will be compared with standard technical specifications and guidance provided in Regulatory Guide 1.8 and 1.33 to determine the adequacy of the staff to protect the plant and to operate safely in routine, emergency, and long-term post-accident circumstances. The plant operating history will be reviewed to assess the combination of staff, operating controls and alarms, and administrative controls, in particular plant procedures, emergency planning and offsite preparedness, to determine whether additional staff, qualifications, or administrative controls will be required for continued safe operation.

2. Safety Objective:

To obtain reasonable assurance that the plant has enough people, with sufficient training and experience, and has administrative controls adequate to specify proper operation in routine, emergency and post-accident conditions.

3. Status:

Most of the older plants have staff members that meet the experience and educational requirements given in ANSI N18.1 - 1971 (endorsed by Regulatory Guide 1.8); however, a comparison against current criteria for the composite staff has not been made. These plants have provided training for subsequent plant staffs and plant experience has in general demonstrated safe design and operation. Operating experience review is ongoing; and has been, in general, favorable. However, an analysis of this experience for trends, common elements, and potential hidden problems has not been systematically performed.

A review of Section VI of operating reactor licensees technical specifications was begun in 1974 using Section VI of STS as a model. As of September 1975 these reviews had been completed and the plants licensed prior to this time had been found to: (1) be acceptable and upgrading was not required, (2) require upgrading of only the reporting requirements, or (3) require improvement to be comparable to the STS model. Plants licensed after September 1975 have been reviewed against the STS model. Further review of Section VI, therefore will not be required.

Emergency plans submitted at the OL stage complied with 10 CFR 50 Appendix E 1970; however, these plans are not consistent with the guidance given in new Regulatory Guide 1.101 Rev. 1 1977.

4. References:

1. R. G. 1.8 and 1.33
2. ANSI N18.1 - 1971
3. ANSI N18.7 - 1972 Revised
4. Standard Technical Specifications, Section VI
5. 10 CFR 50, Appendix E
6. R. G. 1.101 Rev. 1 1977
7. SRP 13.3
8. NUREG 75/111, Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans In Support of Fixed Nuclear Facilities
9. EPA Manual of Protective Action Guides and Protective Action for Nuclear Incidents, September 1975
10. Memorandum of Understanding, NRR and OSP on State and Local Preparedness, March 10, 1977

DEFINITION

TOPIC: XIII-2 Safeguards/Industrial Security

1. Definition:

Industrial security will be included under the scope of the operations review. Design features to assess the plant's capability to prevent sabotage and protect the operating unit(s) at dual or three unit sites with unit(s) under construction will be included. Protective measures will be balanced against the sabotage threat. Fuel accountability will also be reviewed to assure that adequate inventory control procedures exist and the required records are kept.

2. Safety Objective:

To determine that the plant has adequate security forces, design features, procedures and plans, and other administrative controls to meet the postulated sabotage threat. To assure that the fuel is adequately accounted for, that proper records are maintained, and the required reports are made.

3. Status:

Each licensee presently has a security program and a fuel accountability program. Revised 10 CFR 73.55 has been published and submittals in accordance with its provisions were due May 25, 1977. These submittals are presently being evaluated.

4. References:

1. 10 CFR 70
2. 10 CFR 73
3. Standard Technical Specifications, Section VI

DEFINITION

TOPIC: XIV (Section on STARTUP TESTS AND CRITICALITY Intentionally Left Blank)

DEFINITION

TOPIC: XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve.

1. Definition:

Review the assumptions, calculational models used and consequences of postulated accidents which involve an unplanned increase in heat removal. An excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. If clad failure is calculated to occur, determine that off-site dose consequences are acceptable.

2. Safety Objective:

To assure that pressures in the reactor coolant and main steam systems are limited in order to protect the reactor coolant pressure boundary from overpressurization and that fuel rod cladding failure as a result of DNBR is limited.

3. Status:

During each reload review by the staff, the previously determined limiting transient is reviewed to determine if new core parameters are more restrictive than the reference analysis parameter values.

4. References:

SRP 15.1.1 through 15.1.4

DEFINITION

TOPIC: XV-2 Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)

1. Definition:

Review the assumptions, including use of non-safety grade equipment and concurrent steam generator or tube failure or blowdown of more than one steam generator, calculational models used and consequences of postulated accidents which cause an increase in steam flow. The excessive steam flow reduces system temperature and pressure which increases core reactivity and can lead to a decrease of shutdown margin and DNBR.

2. Safety Objective:

To assure that (1) pressure in the reactor coolant and main steam lines are limited in order to protect the reactor coolant pressure boundary from overpressurization, (2) fuel damage is sufficiently limited so that the core will remain in place and intact with no loss of core cooling capability, (3) doses at the nearest exclusion area boundary are a small fraction of 10 CFR Part 100 guidelines, (4) ambient conditions do not exceed equipment qualification conditions (particularly non-safety grade quipment used to mitigate the accident), (5) the thermal and stress transients do not damage the reactor vessel and (6) systems necessary for safe shutdown are not damaged by the accident.

3. Status:

Investigation of the effects of high energy line failures outside containment on other equipment was initiated as a generic issue in 1971 and all but a few facilities have been completed. New acceptance criteria has evolved during the review period. There was no similar investigation for failures inside containment. No reviews on operating plants of the effects on the reactor of concurrent steam generator or tube failure, or of blowdown of more than one steam generator have been performed.

4. References:

SRP 15.1.5

DEFINITION

TOPIC: XV-3 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulatory Failure (Closed)

1. Definition:

Review the assumptions, calculational models used and consequences of postulated accidents which involve a decrease in secondary heat removal. The decrease in heat removal causes a sudden increase in system pressure and temperature.

2. Safety Objective:

To assure that pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained.

3. Status:

The consequences associated with these transients are compared during each reload review to the consequences found to be acceptable during previous reload reviews.

4. References:

SRP 15.2.1 through 15.2.5

DEFINITION

TOPIC: XV-4 Loss of Non-Emergency A-C Power to the Station Auxiliaries

1. Definition:

Review the assumptions, calculational models used, and consequences of postulated accidents which involve the loss of non-emergency AC power (loss of offsite power or onsite a-c distribution system) to station auxiliaries (e.g., reactor coolant circulation pumps). This power loss will, within a few seconds, cause the turbine to trip and reactor coolant system to be isolated, which in turn causes the coolant pressure and temperature to increase.

2. Safety Objective:

To assure that the pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained.

3. Status:

During each reload review by the staff, the previously determined limiting transient is reviewed to determine if new core parameters are more restrictive than the reference analysis parameter values.

4. References:

SRP 15.2.6

DEFINITION

TOPIC: XV-5 Loss of Normal Feedwater Flow

1. Definition

   Review the assumptions, calculational models used, and consequences of the postulated loss of feedwater flow accidents, which cause an increase in coolant pressure and temperature.

2. Safety Objective:

   To assure that pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained.

3. Status:

   The consequences associated with these transients are compared during each reload review to the consequences found to be acceptable during previous reload reviews.

4. References:

   SRP 15.2.7

DEFINITION

TOPIC: XV-6 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

1. Definition:

Review the assumptions, calculational models used and consequences of postulated accidents which involve feedwater line breaks of different sizes. A feedwater line break, depending on size, may cause reactor system heatup (by reducing feedwater flow to the steam generator), or cooldown (by excessive energy discharge through the break).

2. Safety Objective:

To assure that pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained and that any radioactivity release would result in doses at the site boundary well within 10 CFR Part 100 guidelines.

3. Status:

The identification of the most limiting transients and the consequences associated with these transients is evaluated during each reload review by the staff.

4. Reference:

SRP 15.2.8

DEFINITION

TOPIC: XV-7 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

1. Definition:

Review the assumptions, calculational models, and consequences of seizure of the rotor or break of the shaft of a reactor coolant pump in a PWR or recirculation pump in a BWR. These accidents result in a sudden decrease in core coolant flow and corresponding degradation of core heat transfer and, in a PWR, an increase in primary system pressure. If clad failure is calculated, determine that off-site consequences are acceptable.

2. Safety Objective:

To assure that the consequences of a reactor coolant pump rotor seizure or reactor coolant pump shaft break are acceptable; i.e., that no more than a small fraction of the fuel rods fail, that the radiological consequences are a small fraction of 10 CFR Part 100 guidelines and that the system pressure is limited in order to protect the reactor coolant pressure boundary from overpressurization.

3. Status:

Reviewed during each reload only if there is reason to believe that results would be different from the reference analysis; i.e., only if a change in core parameters invalidates previous analyses.

4. References:

1. SRP 15.3.3

DEFINITION

TOPIC: XV-8 Control Rod Misoperation (System Malfunction or Operator Error)*

1. Definition:

Review the licensee's description of rod position, flux, pressure, and temperature indication systems and the actions initiated by those systems which can mitigate the effects or prevent the occurrence of various misoperations. Review the descriptions of the input calculations and the calculational models used and the justification of their validity and adequacy. A transient of this type can result in achieving fuel melt temperatures and potential fuel damage.

2. Safety Objective:

To assure that the consequences of this event do not exceed specified fuel design limits and that the protection system action be initiated automatically.

3. Status:

Reviewed during reload, technical specifications revised to compensate for changes in analytical results.

4. References:

1. SRP 15.4.3

*Reviewed for PWRs only, SRP 15.4.1 and 15.4.2 cover BWRs and no additional areas considered.

DEFINITION

TOPIC: XV-9 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

1. Definition:

Review BWRs for (1) startup of an idle recirculation pump, and (2) a flow controller malfunction causing increased recirculation flow. Review PWRs with loop isolation valves for startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which isolation valves open. For PWRs without loop isolation valves review startup of a pump in any inactive loop. If clad failures are calculated, determine that offsite consequences are acceptable.

2. Safety Objective:

To verify that the plant responds in such a way that the criteria regarding fuel damage and system pressure are met (i.e., no more that a small fraction of the fuel rods fail, that radiological consequences are small fraction of 10 CFR Part 100 guidelines, and that the system pressure is limited in order to protect the reactor coolant pressure boundary from overpressurization.)

3. Status:

PWRs reviewed against FSAR, BWR reviewed at each reload, technical specifications required to preclude exceeding safety limits during transients.

4. References:

1. SRP, 15.4.4, 15.4.5

DEFINITION

TOPIC: XV-10 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)

1. Definition:

   Review the assumptions, calculational models used and consequences of moderator dilution. An accident of this type could result in a departure from nucleate boiling and a loss of shutdown margin.

2. Safety Objective:

   Confirm that the plant responds to the events in such a way that the criteria regarding fuel damage and system pressure are met and adequate time allowed for the operator to terminate the dilution before the shutdown margin is reduced. (Reactor coolant pressure and main steam pressure should be limited in order to protect the reactor coolant pressure boundary from overpressurization.) (Operator action must be initiated within 30 minutes following this event if refueling, and within 15 minutes during other modes of operation.)

3. Status:

   Only reviewed during initial OL review and not thereafter. The consequences may not have been calculated in accordance with current practice.

4. References:

   1. SRP 15.4.6

DEFINITION

TOPIC: XV-11 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (BWR)

1. Definition:

   Review the spectrum of misloading events analyzed to verify that the worst situation undetectable by incore instrumentation has been identified. This review will include an assessment of the plant's offgas and steam line radiation monitors to detect fuel damage and their capability to automatically isolate the offgas system when necessary.

2. Safety Objective:

   To assure that a misloaded assembly is detected and if undetected will not result in exceeding fuel safety limits, or radioactive releases.

3. Status:

   Reviewed during reloads, technical specifications developed to limit consequences of worst misloaded assembly to small fraction of 10 CFR Part 100 guidelines. Technical Specifications set points for radiation monitors alarm/isolation signals have been found deficie and have been updated on a case-by-case basis for several plants.

4. References:

   1. SRP 15.4.7

DEFINITION

TOPIC: XV-12 Spectrum of Rod Ejection Accidents (PWR)

1. Definition:

Review the assumptions, calculational models used and consequences, including radiological consequences, of PWR control rod ejection accidents and review the technical specifications regarding control of reactivity worth and technical specifications on primary to secondary leakage. Ejection of a control element assembly from the core can occur if the control element drive mechanism housing or the nozzle on the reactor vessel head breaks off circumferentially. The ejection of a control element assembly by the reactor coolant system pressure can cause a severe reactivity excursion. This accident may result in high doses for those plants where fuel failures are postulated to occur as a result of the accident. This accident usually determines the maximum allowable steam generator leak rate.

2. Safety Objective:

To ensure that, if a control element assembly ejection occurs, core damage is minimal, no additional RCPB failures occur, the calculated radial average energy density is limited to 280 cals/gm at any axial fuel location in any fuel rod, and that the radiological consequences will not exceed appropriate limits.

3. Status:

Releases through the containment and/or steam generator leaks are analyzed for current plants, but were not reviewed routinely for older plants. Many of the operating plants have no leak technical specifications or they are excessively high. During each reload by the staff, the previously determined limiting transient is reviewed to determine if the new ejected rod worth is more restrictive than the reference analysis values.

4. References:

SRP 15.4.8 and R. G. 1.77

DEFINITION

TOPIC: XV-13 Spectrum of Rod Drop Accidents (BWR)

1. Definition:

Review the assumptions, calculational models used and consequences of BWR control rod drop accidents and review the technical specifications regarding control of rod reactivity worth. An uncoupled rod may hang up in the core when the control rod drive is withdrawn and drop later when the consequences of a rapid control rod withdrawal are most severe. An analysis of the radiological consequences from this accident will be included.

2. Safety Objective:

To limit the effects of a postulated control rod drop to the extent that RCPB stresses are not exceeded and core damage is minimal. To assure that the radial average fuel rod enthalpy at any axial location in any fuel rod is limited to less than 280 cals/gm following the worst reactivity excursion and to assure that the radiological consequences do not exceed appropriate guidelines.

3. Status:

The potential for and reactivity consequences of an accidental control rod drop are now routinely evaluated prior to issuance of an operating license and any time thereafter when changes could affect the accident results or probability of occurrence. Radiological consequences may not have been calculated in accordance with present practice.

4. Reference:

SRP 15.4.9

DEFINITION

TOPIC: XV-14 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

1. Definition:

Review the assumptions, calculational models used and consequences of actuation of the high pressure coolant injection system or faulty operation of the volume control system. The chemical and volume control system regulates both the chemistry and the quantity of coolant in the reactor coolant system. Changing the boron concentration in the reactor coolant system is a part of normal plant operation compensating for long term reactivity effects. Actuation of these systems could increase the volume of coolant within the RCPB causing a high water level, possibly high power level, and high or low pressure. If clad failure is calculated, determine that off-site consequences are acceptable.

2. Safety Objective:

To assure that water added to the RCPB does not cause transients that exceed RCPB pressure limits or result in unacceptable fuel damage. No activity is released during the transient but the transient may subsequently result in increased radioactivity in gaseous releases during normal operation.

3. Status:

This transient is now routinely analyzed prior to issuance of an operating license and any time thereafter when proposed changes would affect the transient results. Radiological consequences may not have been calculated in accordance with current practice.

4. Reference:

SRP 15.5.1

DEFINITION

TOPIC: XV-15 Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

1. Definition:

Review the assumptions, calculational models used and consequences of inadvertent opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve. Loss of reactor coolant inventory and depressurizing action of the reactor coolant system can occur if the PWR pressurizer safety/relief valve or the BWR Safety/Relief valves open spuriously, or open when required but fail to reclose properly.

2. Safety Objective:

To preserve fuel cladding integrity during reactor coolant system depressurization transients resulting from faulty operation of a relief or safety valve while at rated power.

3. Status:

The transient is now evaluated prior to issuance of an operating license and any time thereafter when proposed changes could affect the transient results.

4. Reference:

SRP 15.6.1 and R.G. 1.70

DEFINITION

TOPIC: XV-16 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

1. Definition

Review the assumption, calculational models used and radiological consequences of failure of small lines carrying primary coolant outside containment and review the technical specifications associated with primary coolant radioactivity concentrations, isolation valve closure times and isolation valve leakage limits. In the event of a rupture of any component in the instrument lines outside primary containment, primary coolant and any radioactivity contained in the coolant or released to the coolant during the transient will be released if the instrument lines are connected to the RCPB. Primary coolant sample lines if broken outside primary containment can also allow coolant and radioactivity in the coolant to escape in the same manner. When these lines discharge to secondary containment, the integrity of the secondary containment and the efficiency of the filtration systems must be determined.

2. Safety Objective:

To assure that any release of radioactivity to the environment is substantially below the guidelines of 10 CFR 100.

3. Status:

The radiological consequences of small line breaks outside of primary containment have been evaluated routinely since 1970 prior to issuance of operating licenses, but have not always included the effects of iodine spikes during the depressurization transient.

4. Reference:

R. G. 1.11, GDC 55 and 56, SRP 15.6.2

DEFINITION

TOPIC: XV-17 Radiological Consequences of Steam Generator Tube Failure (PWR)

1. Definition:

Review the assumptions, calculational models used and consequences of a steam generator tube failure with and without loss of off-site power and review the technical specifications associated with coolant activity concentrations. Steam generator tube failures allow escape of reactor coolant into the main steam system and to the environment. An analysis of the radiological consequences of this accident will be included.

2. Safety Objective:

To assure that the plant responds in a proper manner to this accident, including appropriate operator actions and to assure that radioactivity released following steam generator tube failure(s) is a small fraction of the 10 CFR 100 guidelines and within 10 CFR 100 for the case of a coincident iodine spike.

3. Status:

The iodine release mechanism may not have been analyzed in accordance with present assumptions and methods for some of the older PWRs. Some operating plants do not have iodine activity limits in their technical specifications or have inappropriately high limits.

4. Reference:

SRP 15.6.3, R. G. 1.5

DEFINITION

TOPIC: XV-18 Radiological Consequences of Main Steam Line Failure Outside Containment [redacted]

1. Definition:

Review the assumptions, calculational models used and consequences of failure of a main steam line outside containment and review the technical specifications associated with primary coolant activity concentrations and main steam isolation valve closure times.

2. Safety Objective:

A steam line break outside containment allows radioactivity to escape to the environment. To limit the release of radioactivity to the environment to well within the guidelines of 10 CFR 100 in the event of a large steam line break the primary coolant radioactivity must be appropriately limited by technical specifications.

3. Status:

Some operating plants do not have appropriate coolant activity technical specifications.

4. Reference

SRP 15.6.4

ref VI-9A

DEFINITION

TOPIC: XV-19 Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

1. Definition:

Review the licensee's analyses of the spectrum of loss of coolant accidents including break locations, break sizes and initial conditions assumed, the evaluation model used, failure modes, radiological consequences, acceptability of auxiliary systems, functional capability of the containment and the effects of blowdown loads. LOCAs are postulated breaks in the reactor coolant pressure boundary resulting in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. LOCAs result in excessive fuel damage or melt unless coolant is replenished.

2. Safety Objective:

To assure that the consequences of loss of coolant accidents are acceptable; i.e., that the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 are met, that the radiological consequences of a design basis loss of coolant accident from containment leakage and the radiological consequences of leakage from engineered safety features outside containment are acceptable and the structural effects of blowdown are acceptable.

3. Status:

ECCS evaluation is a generic item which is presently under review or is complete for all operating reactors (La Crosse and San Onofre have stainless steel cores and have analyses completed to show conformance with the Interim Acceptance Criteria). Related generic items presently under review are reevaluations for increased vessel head fluid temperatures in W PWRs, effects of core flow on BWR LOCA analyses, GE ECCS input errors and non-jet pump BWR Core spray cooling coefficients. Radiological consequences are not routinely rereviewed.

4. References:

1. Technical Safety Activities Report - December, 1975
2. SRP Section 15.6.5 and its Appendices
3. Cross References - Comp List 1 items IV-1 G, VI-2, VI-3, VI-6A, VI-8

DEFINITION

TOPIC: XV-20 Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment)

1. Definition:

Review the assumptions, calculational models used and consequences of postulated fuel damaging accidents inside and outside containment and review technical specifications associated with fuel handling and ventilation system and filter systems, including interlocks on fuel movement and damage from fuel cask drop and tipping. Include in the review, the assumed activity available for release, decontamination factors, filter efficiencies, activity transport mechanisms and rates, ventilation system potential release pathways, and calculated doses.

2. Safety Objective:

To assure that offsite doses resulting from fuel damaging accidents, resulting from fuel handling, or dropping a heavy load on fuel are well within the guideline values of 10 CFR Part 100.

3. Status:

The radiological consequences of fuel handling accidents inside containment is presently being performed as a generic review for PWRs. The radiological consequences of fuel damaging accidents outside containment of operating plants is only evaluated if technical specifications are reviewed.

4. References:

1. SRP Section 15.7.4
2. R. G. 1.25

DEFINITION

TOPIC: XV-21 Spent Fuel Cask Drop Accidents

1. Definition:

Review the potential for spent fuel cask drops, the damage which could result from cask drops, and the radiological consequences of a cask drop from fuel damaged within the cask under conditions exceeding the design basis impact on the cask.

2. Safety Objective:

To assure that the damage to fuel within the casks and radiological consequences resulting from a cask drop are acceptable or that acceptable measures have been taken to preclude cask drops.

3. Status:

Fuel cask drop analysis is a generic item which has been completed on some plants or is presently under review for all other operating reactors.

4. References:

1. SRP Section 15.7.5
2. R. G. 1.25
3. Pink Book

DEFINITION

TOPIC: XV-22 Anticipated Transients Without Scram

1. Definition:

Review the postulated sequences of events, analytical models, values of parameters used in the analytical models and the predicted results and consequences of events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram). Analyses of the radiological consequences for these transients will be included. Failure of the reactor to shutdown quickly during anticipated transients can lead to unacceptable reactor coolant system pressures and to fuel damage.

2. Safety Objective:

To assure that the reliability of the reactor shutdown systems is high enough so that ATWS events need not be considered or to assure that the consequences of ATWS events are acceptable, i.e., that the reactor coolant system pressure, fuel pressure, fuel thermal and hydraulic performance, maximum containment pressure and radiological consequences are within acceptable limits.

3. Status:

ATWS is a generic topic currently under review to determine a position for all power reactors. BWR licensees have been requested to install reactor coolant pump trips as a short term program measure. All licensees have submitted descriptions of the applicability of vendor generic ATWS reports for their plants. The schedule for review of class C plants, which includes those plants designated for Phase II of SEP, has not yet been developed.

4. References:

1. Pink Book
2. WASH 1270
3. ACRS
4. TSAR
5. SRP Section 15.8 and Appendix

DEFINITION

TOPIC: XV-23 Multiple Tube Failures in Steam Generators

1. Definition:

Assess the effects of multiple steam generator tube failures (ranging from leaks to double ended ruptures) as a result of pressure differentials that may occur following a LOCA, steam line break or ATWS events.

2. Safety Objective:

Assure that the reflood of the core following a LOCA is possible and that the radiological consequences following these accidents are within the 10 CFR Part 100 guidelines.

3. Status:

The consequences of multiple tube failures have not been analyzed for any plant at the licensing stage. Work has been done for some operating plants, but ultimate goals have yet to be set.

4. References:

1. Prairie Island Docket
2. Turkey Point Docket
3. Surry #1 and #2 Docket
4. ATWS Report

DEFINITION

TOPIC: XV-24 Loss of All AC Power

1. Definition:

   Review plant systems to determine that following loss of all AC power (on and offsite) the reactor is shutdown and core cooling can be initiated. Loss of all AC power causes loss of most emergency equipment and instrumentation.

2. Safety Objective:

   To assure that with only DC power, i.e., equipment design, diversity, and operator action are sufficient to initiate core cooling within a short time period (typically 20 minutes).

3. Status:

   Not an explicit SRP topic. Availability of some AC power is assumed in all accident/transient analyses. Topic may be considered as an auxiliary fuel pump or RCIC pump diversity spinoff.

4. References:

DEFINITION

TOPIC: XVI Technical Specifications

1. Definition:

   The existing technical specifications, associated with SEP topics, will be compared with the standard technical specifications for deviations. Where significant differences exist, they will be identified and considered for upgrading. The bases for the specifications will be examined including trip setpoints and accounting for nuclear uncertainty. Where significant voids occur in existing specifications, appropriate values will be identified and considered for upgrading.

2. Safety Objective:

   To assure that the safety limits and operational safety measures are sufficiently specified for the plant to minimize the probability of accidents that could result from equipment failure, misoperation or human error.

3. Status:

   See Topic XIII-1 Conduct of Operation for Section 6 status. The other sections of the Technical Specifications are reviewed only to the extent that reloads, license amendments or generic problem require.

4. References:

   1. Standard Technical Specifications R. G. 1.8 and 1.33
   2. Standard Review Plan
   3. Regulatory Guide 1.70 Chapter 16
   4. 10 CFR 50.36

DEFINITION

TOPIC: XVII Operational QA Program

1. Definition:

   Review the Quality Assurance Program with respect to safe and reliable operation of the plant.

2. Safety Objective:

   Since 1973 significant new guidance for operational QA programs in the form of Regulatory Guides and WASH documents has been issued describing how to meet the criteria at 10 CFR Appendix B. The objective of this guidance is to assure that operation, maintenance, modification, and test activities do not degrade the capability of safety related items to perform their intended function.

3. Status:

   Generic review for compliance with current standards is underway. As of May 1977 50 of the 63 operating plants have QA programs which meet current criteria. The 13 remaining plants are currently under review, with an estimated completion date of July 1977.

4. References:

   1. 10 CFR 50, Appendix B
   2. WASH 1283 - Rev. 1 (5/24/74)
   3. WASH 1284 (10/26/73)
   4. WASH 1309 (5/10/74)
   5. ANSI N18.7 - 1976 (2/19/76)